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# MATERIALS TESTING REACTOR PROJECT HANDBOOK

J. H. Buck
C. F. Leyse

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# MATERIALS TESTING REACTOR PROJECT HANDBOOK

Prepared Jointly by Argonne National Laboratory
and
Oak Ridge National Laboratory


John H. Buck . . . . . Editor
Carl F. Leyse . . . . Associate Editor


Prepared at the request of the MTR Steering Committee

DATE ISSUED SEP 12 1952


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## Chapter 7*

# PROCESS-WATER SYSTEM

## 7.1 INTRODUCTION

The process-water system is the primary cooling system of the Materials Testing Reactor. The system provides a flow of water through the active lattice and other parts of the reactor that require water cooling. This chapter discusses the design and some of the problems encountered in the design of the process-water system. Although the active lattice and beryllium reflector form an integral part of the complete system, they are discussed only in a very general way in regard to some of the design problems. Detailed discussions of these two parts of the reactor are given in previous chapters.

During 1947 a design of the process-water system was conceived and developed for construction of the MTR at Oak Ridge National Laboratory. In the middle of 1949 the U. S. Navy Proving Ground in the Snake River Plains of Idaho was chosen as the Reactor Testing Station. With these fundamental data available, the design was redeveloped to suit the Idaho locality.

## 7.2 THE SYSTEM IN GENERAL

**7.2.1 General Description of the System.** Figure 7.2.A, schematic diagram of the MTR process-water system, presents an overall picture of all the vessels, pumps, pipe lines, and auxiliary equipment involved in the MTR process-water system.

Water at 100°F is pumped up to the working reservoir from the 60,000-gal sump tank by means of two 10,000-gpm 225-ft-head horizontal centrifugal pumps. A third pump is available in the event that either of the operating pumps fails or is shut down for maintenance. The water from the 150,000-gal working reservoir flows, under a normal head of 170 ft, directly to the reactor tank through a 30-in. stainless steel pipe outside the Reactor Building and a 36-in. stainless steel pipe beneath the Reactor Building. A flow instrument, strainer, and electric motor-operated flow control valve are located between the working reservoir and the reactor. These three items are blocked off in

* ANL contribution.

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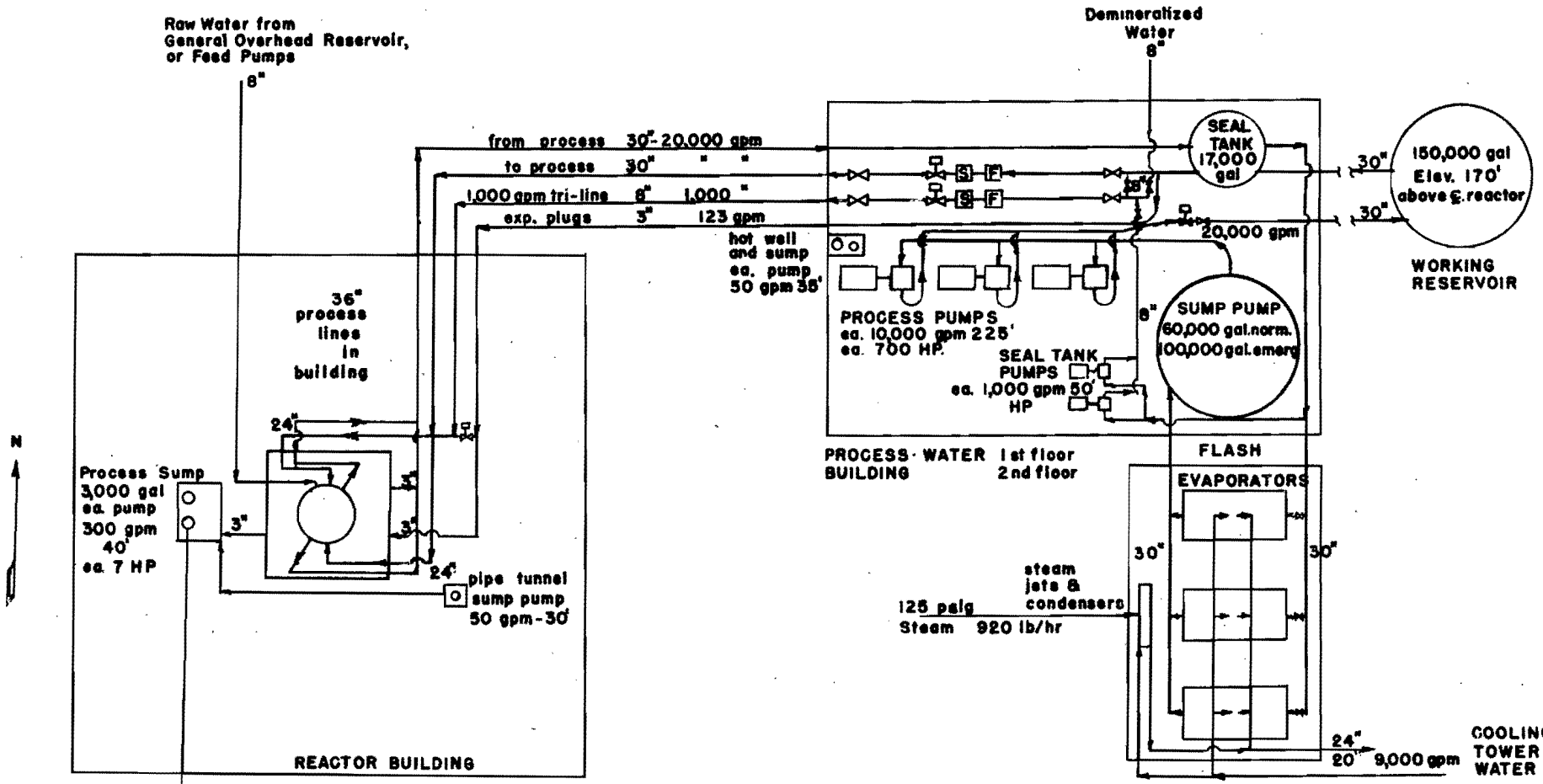


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| FRACTIONAL TOL. ± 1/64 | CHIEF ENG | | | This drawing, [redacted] is the property of the Argonne National Laboratory. [redacted] | PART | SCHEMATIC DIAGRAM OF | | |
| DECIMAL TOL. ± .005 | CHIEF DRAFTSMAN | | | | ASSEMBLY | MTR PROCESS WATER SYSTEM | | |
| ANGLE ± 1/4° | CHECKED | | | | | | | |
| ALL SURFACES TO HAVE AVERAGE 125 MICRO INCH OR AS RECEIVED FINISH. UNLESS OTHERWISE NOTED | DRAWN | 12-21-56 | J.W.M. | SCALE NONE / MATL. / NO. REQD. | SHEET OF | DRWG. NO. FIGURE 7.2.A | | ITEM NO. |
| | GROUP | REPRESENTATIVE | | | | | | |

a pit by gate valves so that the water may be drained from them and they may be inspected or repaired. The 36-in. process-water line is split into two 24-in. stainless steel lines below the basement floor. These lines go up through diagonally opposite corners of the reactor biological shield and connect to the top section of the reactor tank. The water descends through the reactor tank sections, the active lattice, and beryllium reflector, and emerges from the bottom of the reactor tank through two 24-in. lines. The two exit lines are joined to a 36-in. line at the same location as that where the inlet line diverges from a 36-in. line into two 24-in. lines. The exit 36-in. line carries the water to the 17,000-gal seal tank in the process-water building.

The head losses between the working reservoir and the exit of the reactor tank are due almost entirely to the velocity head and friction losses across the active and beryllium sections, with relatively small losses contributed by the conveying pipe line. There are block valves and a flow control valve in the inlet line to the reactor, but there are no obstructions in the pipe line from the reactor to the seal tank.

There is an 8-in. line connected to the 30-in. inlet line at a point upstream from the valve pits and to the 36-in. line near the reactor. This 8-in. line provides a flow parallel to the main flow stream and contains two block valves, a flow instrument, a strainer, and a control valve. It can carry 1000 gpm and is used in the event that the main water line valve is closed off, thereby assuring a continuous water supply to the reactor at all times, particularly at shutdown. This same line, which allows 1000 gpm of process water to by-pass the main stream, can also be used to carry 1000 gpm of fresh demineralized water during periods of reactor flushing, or 1000 gpm of seal-tank water for recirculation through the seal tank and the reactor only.

The pipe tunnel beneath the Reactor Building is about 55 ft below the centerline of the active section and lies on bed rock. At the low point in the pipe tunnel there is a single 50-gpm sump pump which pumps out any water that leaks into the tunnel. This water is pumped to the 3000-gal process water sump, which is also located beneath the Reactor Building. The two 36-in. lines in the pipe tunnel lie one above the other; the inlet line is on top.

Water in the seal tank is drawn into the flash evaporators by means of the partial vacuum existing there. The water, in being flashed at a pressure that yields 100°F effluent, falls from the flash evaporators into the sump tank below. From the sump tank the water is once more pumped up to the working reservoir to complete the process-water cooling cycle. The reactor heat, liberated in the flash evaporators, is carried away by the cooling-tower water circulated through the condenser tubes in the evaporators.

**7.2.2 Stand-by and Emergency Features.** Certain provisions are made to ensure satisfactory operation of the process-water system under emergency conditions. The primary requirement in emergencies is that there always be some water flow through the reactor.

In addition to the water available in the working reservoir, raw water from the general overhead reservoir is available for cooling the reactor by a direct 8-in. line to the reactor tank. In the event both overhead tanks are destroyed, water can be pumped directly to the reactor from the two 500,000-gal ground-level reservoirs by means of two 850-gpm electric motor-driven or a 1000-gpm gasoline engine-driven centrifugal pump. If the motor-driven pumps in the demineralized-water line can be used, the reactor can be supplied through the two 24-in. inlet process-water lines with water from the 100,000-gal demineralized water storage tank. Within 30 sec of commercial electrical power failure, a Diesel electric generator set will provide electric power to strategic pumps and equipment. The reactor can be initially cooled under these circumstances by the reserve water in the working reservoir and then by 1000 gpm of process water recirculated by one of the seal-tank pumps, of which one is electric motor driven and one is gasoline engine driven.

The loops in the 24-in. water lines in the reactor structure help keep the active lattice under water at all times. This assures natural convection water cooling during an emergency shutdown and thereby prevents melting of the fuel assemblies. Should local water boiling occur, the localized steam pressure could conceivably force the water out of the active section and cause melting. However, sufficient pressurized water sources are available to practically ensure some water flow through the active lattice. A 2-in. line between the top of the reactor tank and the water loop is provided to prevent siphoning of the water from the active lattice if the process-water pipe line should be severed. In the event of any major emergency due to electrical outage, steam failure, etc., the reactor power will be nulled, and shutdown cooling will be the only requirement.

**7.2.3 Instrumentation and Flow Balance.** In order to maintain a continually balanced water flow through the process-water system, a scheme has been established in which water is pumped from the sump tank to the working reservoir. The valve that determines the amount of water lifted is governed by the level of water in the working reservoir. The water flow rate from the working reservoir to the reactor at a given control-valve setting and a constant head is uniform. The water flows from the reactor to the seal tank and then through the flash evaporators into the sump tank. Any overflow from the seal tank also goes to the sump tank. Any shortage of water in the seal tank is automatically compensated for by a decreased flow to the flash evaporators because the flash evaporators are fed water at a rate that is determined by the level of water in the seal tank and the pressure in the flash evaporators. In this manner the water flow through the whole system is kept in equilibrium. Normally the sump tank contains 60,000 gal of water. However, to accommodate undue surges and accumulations of water during emergencies such as electrical outages, the capacity of the sump tank is 100,000 gal.

Instrumentation on the flash evaporators provides the maximum possible vacuum. The flow of water to the flash evaporators is adjusted by means of valves in the inlet lines. The temperature of the water leaving the flash evaporators determines, by means of a thermal control, the quantity of cooling water passing through the condensing tubes.

The following instrumentation is related to the working reservoir: A temperature-indicating controller limits the lowest water temperature in the working reservoir by allowing steam to flow into a heating coil at the base of the reservoir. The floating roof of the working reservoir is arranged so that at water levels exceeding normal a mechanical switch shuts off the main process-water lift pumps. Within the normal operating range, the level in the working reservoir is maintained by means of an instrument which governs the opening of the main valve controlling the water flow up into the working reservoir. At lower water levels in the working reservoir the reactor is abruptly shut down and the main water valve is closed in 30 sec. The water flowing to the reactor passes through a pit in which there are two block gate valves which are pneumatically operated. The purpose of these block valves is to isolate the pit so that the portion of pipe between the block valves can be drained and inspected. Between the block valves, going downstream, there are, respectively, a flow recorder, a motor-operated control valve, and a strainer. Adjacent to the pit are taps for instruments which make continuous

recordings of the water resistivity and *p*H. Upstream of the first block valve is a 3-in. tap for water used in cooling the experimental plugs; the full head due to the height of water in the working reservoir is always available at this tap. The only valves that control the flow in this line are at the experimental plugs. An 8-in. by-pass line before the main motorized valve allows a flow of about 1000 gpm of water to the reactor when the main valve is shut. This line joins another 8-in. line from the demineralized water supply and an 8-in. line from the seal-tank pumps. Each of these three lines has an on-off remotely controlled pneumatic valve. Immediately after shutdown, process water goes through the 8-in. line directly to the reactor. Shortly thereafter demineralized water is used to purge the reactor-seal tank system, and, finally, seal-tank pumps recirculate the fresh water in this system. This 8-in. line enters the reactor-cooling water line at the point where the two 24-in. lines branch from the main 36-in. water inlet line. It also passes through the valve pit with the main water line and has two block gate valves. Going downstream there are, respectively, a flow meter, an 8-in. pneumatic flow-control valve, and a strainer.

The temperature and pressure of the water entering and leaving the reactor are recorded. The signal of the temperature difference between entering and leaving reactor water and that of the water flow are multiplied in a power recorder. The power of the reactor is also obtained from instruments measuring the gamma intensity of the water leaving the reactor and the neutron flux. Instruments are provided to record the temperature rise of the water in going through each fuel assembly, to record the radiation intensity of the water emerging from each fuel assembly, and to indicate the amount of flow through each assembly.

A flow recorder is provided on the water lines from each experimental plug, and, unless the flow through the plugs is satisfactory, the reactor cannot be started. A water meter is provided in the line supplying demineralized water to the rabbits and experimental plugs so that an inventory may be kept on the amount of water introduced into the process-water system in this manner.

The water leaving the reactor enters the seal tank in the Process Water Building. In the seal-tank entrance there is a radiation recorder for detecting fission breaks. The seal tank is provided with a demineralized-water inlet line and meter for supplying and metering make-up water. It also is provided with a temperature recorder and a level-indicating alarm. All pumps

in the process-water system are provided with pressure gauges. The flow to the flash evaporators from the seal tank is controlled by a valve and is indicated by an instrument in the outlet line from the flash evaporators. The flash evaporators are operated at full vacuum potential. Instruments are provided to indicate the vacuum on each evaporator and to control the amount of cooling water going to each evaporator condenser. The pressure of the steam going to the vacuum system is recorded and controlled. Temperatures in various points of the inter and after condenser vacuum-cooling system are indicated. The vapors discharged from the vacuum system are monitored by a radiation instrument which detects radiations and gives an alarm if large amounts of activity are discharged from the Process Water Building into the atmosphere. The sump tank is provided with a level-indicating-control-alarm and a temperature recorder. A flow recorder is provided for the purge from the sump tank.

**7.2.4 Water-cooled Experimental Plugs and Rabbits.** In the reactor there are seven water-cooled experimental plugs which require a total of about 125 gpm of cooling water. Water to cool these plugs is tapped from the main process-water line upstream of the valve pit. This assures that pressure sufficient to force the water through the plugs is always available, even during shutdown periods. The water leaving the experimental plugs flows into the process-water exit line from the reactor. In case it is not advisable to return the plug-cooling water to the process-water system, the water can be purged directly to the process-water sump. Demineralized water, which is one of the service facilities provided at beam holes, may also be used for plug cooling. During electrical outages these experimental plugs must have cooling water, and since demineralized-water pressure will fail at this time, a by-pass valve is provided from the process-water system to the demineralized-water system so as to maintain cooling of the plugs.

Entering the bottom of the reactor are four hydraulic rabbits used for experimental purposes. About 5 gpm of demineralized water is provided for each of these rabbits.

**7.2.5 Process Water Building.** The 20,000-gpm flow of MTR process water passes through the Process Water Building before entering and after leaving the reactor. This building contains the equipment for controlling the water flow, the pumps for lifting the water to the working reservoir, the shutdown

and emergency cooling-water pumps, and the flash evaporators for cooling and degassing the process water. Figures 7.2.B and 7.2.C show Process Water Building floor plans and building sections, respectively.

The seal tank receiving the water from the reactor section and the sump tank receiving the process water from the flash evaporators are enclosed in this building for shielding and instrument maintenance reasons. The equipment is located on seven main levels so as to obtain building compactness and to achieve proper hydrostatic balance between the seal-tank level, suction lift to the flash evaporators, and barometric discharge to the sump tank.

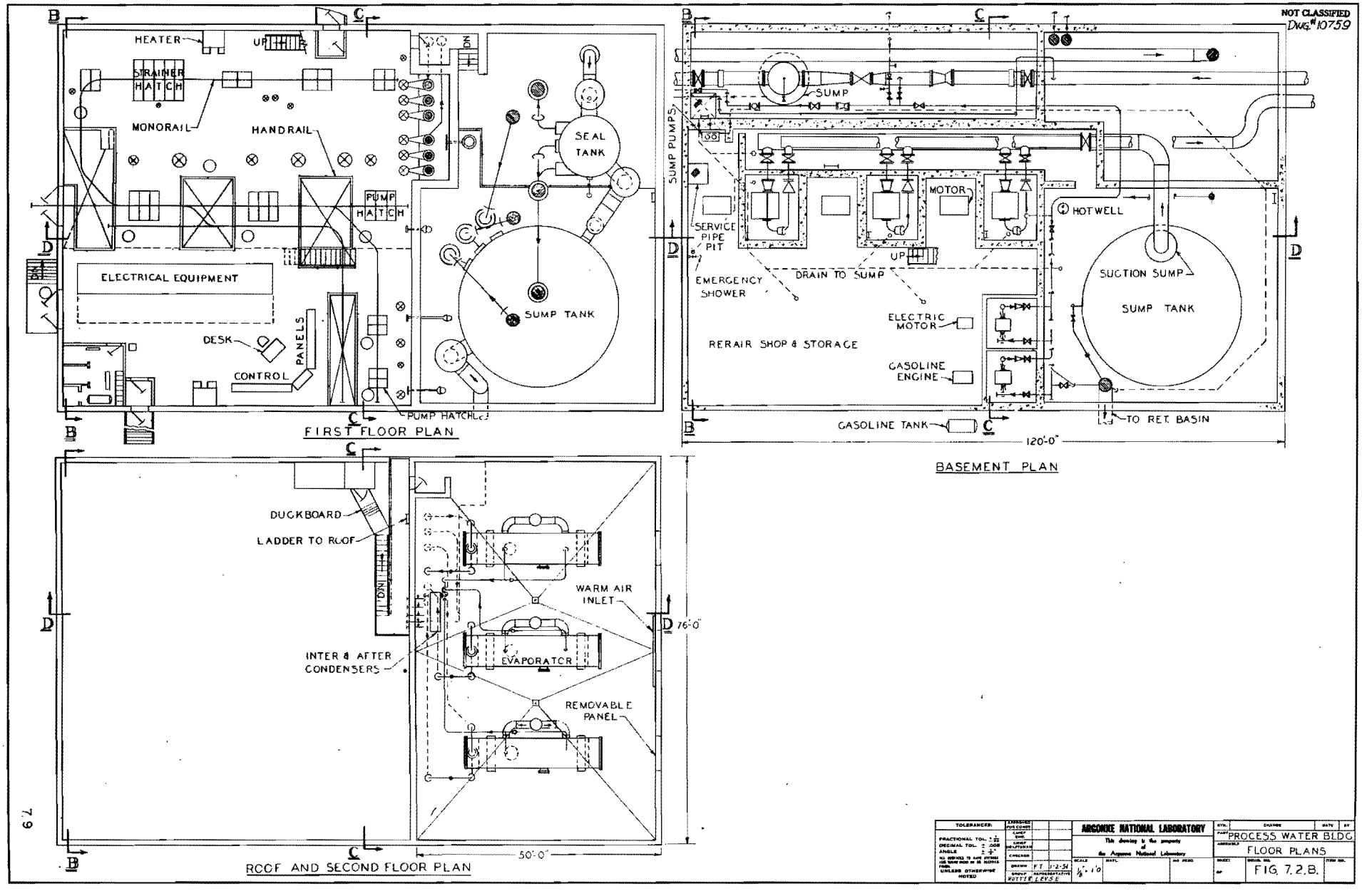
The 30-in. water line from the Reactor Building enters the Process Water Building through the control valve pit and then discharges into the 17,000-gal seal tank. From this tank the process water flows successively through the flash evaporators, sump tank, process-water pumps, and working reservoir, and then to the reactor through a control valve in the Process Water Building.

All piping and pumps are shielded with about 1 ft of concrete, and the valve control handles are extended through the shielding. All instrumentation and controls for maintaining proper process-water temperatures and flows are centralized in the Process Water Building instrument room. Overhead cranes and removable floor slabs provide accessibility to shielded pipes, valves, pumps, etc. Each pump cubicle is shielded from it neighboring cubicle so that repairs can be made during reactor operation.

## 7.3 MAJOR COMPONENTS OF THE SYSTEM

**7.3.1 Sump Tank.** The sump tank has a capacity of 100,000 gal. The primary functions of the sump tank are to receive the process water from the flash evaporators and to supply it to the process-water pumps. The tank normally contains a 3-min holdup of about 60,000 gal. This holdup is necessary for satisfactory operation of the process-water pumps. The additional 40,000 gal capacity of the tank provides for the accumulation of water in the event of failure of the process-water pumps. The sump tank also receives process water via an overflow from the seal tank.

**7.3.2 Process Pumps.** Three 10,000-gpm 225-ft-head horizontal single-stage double-suction centrifugal pumps, each driven by a 700-hp motor, are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HEATER
UP
STRAINER HATCH
MONORAIL
HANDRAIL
PUMP HATCH
SEAL TANK
SUMP TANK
ELECTRICAL EQUIPMENT
DESK
PANELS
CONTROL
PUMP HATCH
FIRST FLOOR PLAN
SUMP
SUMP PUMPS
MOTOR
HOTWELL
SERVICE PIPE PIT
EMERGENCY SHOWER
DRAIN TO SUMP
UP
SUCTION SUMP
SUMP TANK
ELECTRIC MOTOR
REPAIR SHOP & STORAGE
GASOLINE ENGINE
GASOLINE TANK
TO RET. BASIN
120'-0"
BASEMENT PLAN
DUCKBOARD
LADDER TO ROOF
WARM AIR INLET
INTER & AFTER CONDENSERS
EVAPORATOR
REMOVABLE PANEL
76'-0"
50'-0"
ROOF AND SECOND FLOOR PLAN
TOLERANCES:
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PROCESS WATER BLDG.
FLOOR PLANS
FIG. 7.2.B.
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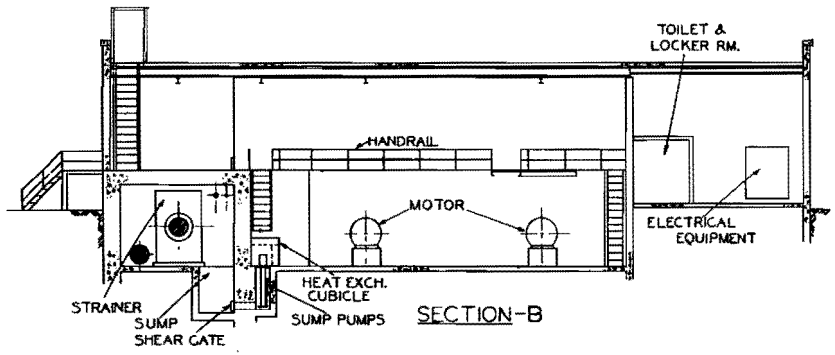
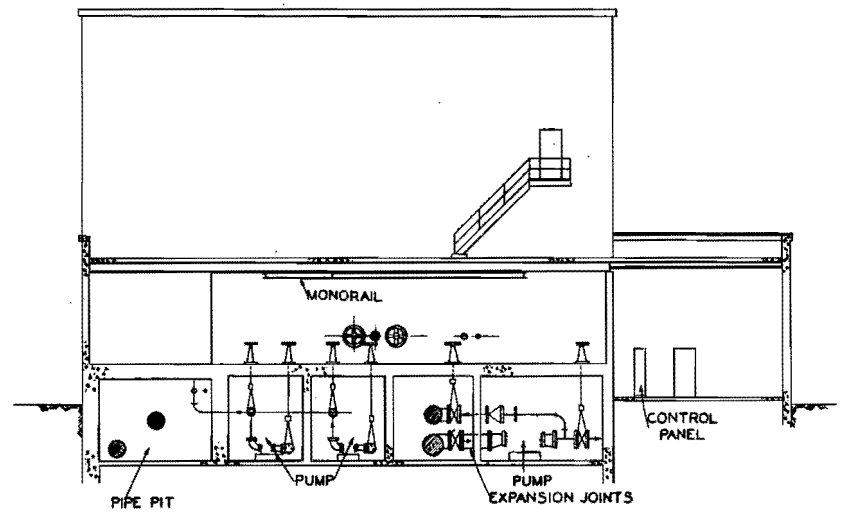



SECTION-B

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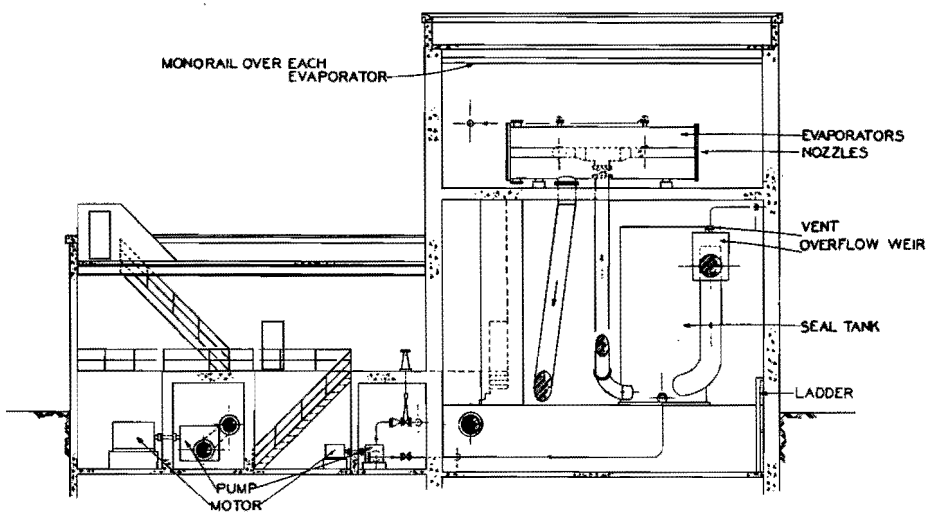


SECTION-D

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| | DRAWN C.C. 12-27-50 | SCALE 1/8"=1'0" | FIGURE 7.2.C |
| | HUTTER LEYSE | | |

7.10

provided for lifting the process water up to the working reservoir. The process pumps are constructed of cast iron since it has been found relatively satisfactory in use with demineralized water, and since the area of the pumps that is exposed to the water is quite small compared to the whole water system. The use of two pumps to supply the 20,000 gpm required, with one stand-by pump, affords the least amount of moving equipment yet offers reserve in case of breakdown. Each pump is in a cubicle surrounded by 1 ft of concrete. This provides personnel radiation protection and permits isolation and repair. To prevent leakage of radioactive water from around the shafts, demineralized water under pressure can be added at the shaft seal. However, allowing the active water to leak into drainage to the retention basin is not serious.

**7.3.3 Working Reservoir.** The purpose of the working reservoir is to supply water at a constant head to the reactor active lattice. In the event of electrical or mechanical failure of the water-pumping system, the reserve of an overhead water supply is most advantageous. If local boiling would not occur in the active lattice and/or if a continual full water flow were not required, the pumps could deliver water directly to the reactor as is done at Hanford.

The height of the working reservoir is based on the allowable internal pressure for which the aluminum reactor tank is designed. Thus the working reservoir is 170 ft above a horizontal centerline through the active lattice. This elevation permits an increase of the process-water flow rate to the highest value attainable with the present reactor tank and fuel elements.

The size of the working reservoir is based on a full process-water flow of 20,000 gpm for 7.5 min. The 150,000-gal capacity of the reservoir can provide this flow. There are several reasons that such a large reserve capacity is desired. It is conceivable that an emergency condition may not be recognized or acted upon for several minutes. Also, during an electrical outage the flow does not cease for half a minute and it takes another half-minute to close the main water valve. In either of these situations an appreciable quantity of reserve water may be lost. In addition, the reactor must be cooled after it is shut down, and by allowing a 1000-gpm flow to the reactor for this purpose, about half the reserve can be lost in an hour.

Protection of personnel against radiation during reactor operation is provided by a fence around the base of the working reservoir. Since approach

to the reservoir is limited in this manner, and because of the height of the reservoir, shielding of the tank is not necessary. Immediately after reactor shutdown, radiation from the tank and pipes has been calculated[4,5] to be above tolerance (up to 0.8 r per 8 hr); however, they can be approached for short periods for maintenance of instruments, etc.

**7.3.4 Valve Pit.** The valve pit, at which all water flowing to the reactor is controlled, was appropriately designed as a part of the Process Water Building. In the pit there are two block gate valves which permit isolation of the strainers, a flow meter, and a motorized control valve, all of which are in the water line from the working reservoir to the reactor. This provides for maintenance of these items. Ahead of the upstream block valve there is an 8-in.-diameter pipe take-off which ties into an 8-in. demineralized-water line and an 8-in. seal-pump circulatory-water line. Each of these three feed lines has an open-closed motor-driven valve. When the 20,000-gpm flow is cut off by closing the main water-control valve, as after reactor shutdown, a 1000-gpm flow automatically by-passes the main water-control valve and flows to the reactor. The 8-in. line to the reactor also has two block gate valves, a strainer, flow meter, and motorized control valve in the valve pit. In case of emergency both the main water-control valve and the control valve in the 8-in. line to the reactor can be operated manually.

**7.3.5 Seal Tank and Pumps.** The purpose of the 17,000-gal seal tank is to maintain a definite water level in the reactor tank by means of the outlet weir during shutdown periods, and also to serve as a water supply source for the flash evaporator intakes. During normal shutdown periods cooling is provided by fresh demineralized water, circulated at 1000 gpm through the seal tank and reactor tank. During electrical outages a gasoline engine-driven 1000-gpm seal-tank pump can be utilized.

**7.3.6 Flash Evaporators.** Flash evaporators are used for cooling the MTR process water because they have been found to be the most efficient and economical of all types of equipment examined. These stainless steel flash evaporators, each 25 ft long and 8 ft in diameter, containing about 5000 ¾-in. cooling tubes, handle the 20,000-gpm flow and the 30,000-kw heat load of the MTR. The 110°F process water from the reactor is sprayed into the flash evaporators where the absolute pressure is 1.9 in. Hg, which is the saturation pressure of 100°F steam. Part of the process water evaporates, thereby cooling the bulk of the water, which falls into the sump tank below. The

evaporated water is condensed outside the water-cooled tubes and then rejoins the main process-water flow. Thus the process water is not only cooled 10°F, but it is degassed by the low pressure in the flash evaporators. During start-up the air is removed from the water at a rate of 175 lb/hr, and the hydrogen and oxygen formed in the active lattice during reactor operation are removed at a rate of 40 lb/hr. The water vapor pressure load under these conditions is 425 lb/hr or about two-thirds of the total vapor pressure load. At 1.9 in. Hg about 3.2 lb of 125 psig steam per pound of gas removed is required.

It is desirable to remove any gases in the water before returning it to the reactor since otherwise excessive gas evolution will occur in the active lattice where the water is heated and where hydrogen and oxygen are formed by water decomposition due to irradiation.

Analysis of gas solubility data for the MTR process water and existing operating piles show the following:

1. If nitrogen from the air is not removed from reactor-cooling water, it will be oxidized to nitrates under irradiation and increase the reactor-water acidity and, therefore, the corrosive aggressiveness on reactor structural materials in contact with water.

2. The process-water cycle should be closed with floating heads on storage tanks to prevent air absorption in water. If this is not done, the vacuum degassing system will have to be 35% larger.

3. With no re-absorption of air, the hydrogen and oxygen evolved from the water as a result of decomposition will almost saturate water at 212°F; hence the water must be de-aerated before returning to the reactor.

Instead of degassing the whole process-water stream, degassing of only a by-pass stream may be adequate. However, owing to the present state of knowledge, a conservative design demands degassing of all the process water. Also, it appears that if the process water could be completely demineralized by the recently developed mixed-bed resin deionization methods, the amount of induced activity and the amount of hydrogen and oxygen gas produced in the active lattice would be considerably reduced.

Other methods considered for cooling the process water are given below, each with the principal reason for rejection.

1. Outdoor spray cooling: This involved dispersion of radioactivity to the atmosphere and the pick-up of dirt particles therefrom.
2. Refrigeration: This was too expensive.
3. Heat exchangers: These were satisfactory for heat removal, but expensive auxiliary degassing equipment would be necessary for removing gases from the water.

7.3.7 Piping. The 30-in.-diameter pipe was selected primarily on the basis of a maximum allowable pressure drop of 25 psi from the working reservoir to the reactor tank. In the Reactor Building the main water line diameter is 36 in. to allow for future expansion without any major excavations or structural changes. The two 24-in. diameter water lines entering and leaving the reactor tank are primarily to distribute the water flow evenly. Since the greatest security against pipe rupture is desired in the line from the working reservoir to the reactor tank, all valves, the flow meter, and the strainer are made of stainless steel; in other locations cast iron is used. In the reactor structure the 24-in. process-water pipe is hung from the structural steel by suspension hangers and is free of any concrete contacts. However, it is surrounded by a spiral welded-steel pipe, and there is Zerolite fiber packing in the annulus between the two pipes. The exit part of the pipe is supported on the bottom stainless steel section of the reactor tank. Stainless steel--clad pipe, 20% or at least 1/16 in. thick, is used because it is more economical than solid stainless steel. The process-water pipe outside the reactor building lies in the ground, whereas in the Reactor Building it is suspended from spring-loaded hangers in the concrete tunnel.

The only possible cause of water hammer in the process-water system would be an instantaneous closing of the 24-in. main process-water-control valve. If this valve closed in less than 0.2 sec, the maximum possible increase in pressure would be 385 psi; if the closing time were 16 sec, the maximum pressure would be 4 psi. However, since the valve is designed to close in 30 sec, the possibility of water hammer is eliminated. Failure of the pumps lifting water to the working reservoir will not cause water hammer because the reverse momentum of the water is quite low.

Pipe with ¼-in.-thick walls is adequate to prevent collapse under vacuum or bursting due to the internal working pressures. Temperature stresses are not appreciable.

## 7.4 BASES FOR DESIGN

7.4.1 Flow Quantity and Heat Transfer. An over-all picture of the heat transfer and flow characteristics of the MTR is given in this section. The details of heat removal and water flow in the active lattice are discussed thoroughly in Section 2.7.1.

The quantity of water required for cooling the MTR is determined by the amount of heat liberated in both the active lattice and the beryllium reflector. Because of the high neutron flux the amount of heat liberated per unit volume is very large. Clean surfaces and high water velocities are therefore necessary in order to obtain high heat transfer fluxes. Also, high water velocities through the active lattice are required to carry away any bubbles of steam or air that may be formed. The water velocity provided through the active lattice is 30 ft/sec.

It became apparent during early investigations with the MTR mock-up that as much as a third of the total water flow was passing through the beryllium reflector. Previous to obtaining these data it had been believed that 15,000 gpm of process water would be sufficient for cooling the reactor. However, because of the large amount of water passing through the beryllium, the process-water system is designed for a total flow of 20,000 gpm. This flow is based on 30,000-kw operation of the reactor. However, it is expected that the same flow will be sufficient for operation at higher power levels.

The temperature of the process water increases about 20°F in passing through a fuel assembly; but the temperature rise of the 20,000 gpm total water flow is only about 11°F. This is because of the relatively large amount of water flowing through the beryllium.

Experiments conducted with a single fuel assembly showed that a 40-psi pressure drop is sufficient to produce a 30-ft/sec water velocity through an assembly. During the design of the MTR water system, it became apparent that, if higher water velocities were desired, a higher head of water would be needed in the working reservoir. Since the aluminum tank around the active section is designed for a pressure of only 74 psig, and since this is the pressure produced when the water is at its highest level (170 ft above the center of the active lattice) in the working reservoir, the greatest theoretical water velocity obtainable through the fuel assemblies is 42 ft/sec. This possible

40% increase in flow velocity could increase the heat transfer film coefficient only 31%. These two percentages are based on the assumption that the pressure drop and film coefficient are respectively proportional to the 1.8 and 0.8 powers of the water velocity. Thus the design of the reactor tank limits the heat transfer film coefficient attainable.

The amount of water flowing through the active lattice also depends on the fact that the pressure existing in the active section corresponds to a particular boiling point, and the temperature of the water film adjacent to the fuel plates must not closely approach this boiling temperature. Localized hot spots in the active lattice determine how high the temperature of the water film will be. Operation of the reactor will determine the highest possible working temperatures of the process water in the active lattice.

**7.4.2 Degasification.** In the MTR it is necessary to maintain a high heat transfer from the fuel elements to the passing water. In order to do this, bubble formation due to dissolved gases in the water must be prevented in the active section. If the amounts of dissolved gases in the process water are kept below their saturation values at the active-section operating pressures and temperatures, no gas evolution will occur. The following discussion describes briefly the degasification of the MTR process water and gives pertinent data connected therewith.

The water in the process-water system has essentially no contact with air since the tanks of the system are provided with floating heads. The dissolved gases in the process water during reactor operation are thus composed of the $H_2$ and $O_2$ produced by water decomposition in the active lattice, the air in the air-saturated demineralized make-up water, and water vapor. However, when the reactor is started, either initially or after a complete purge of the process-water system, the system contains air-saturated fresh demineralized water and functions primarily as a system in contact with air. Although the demineralized water in contact with air in the demineralized water storage tank will not come to saturation equilibrium, in order to have a conservative design it is assumed that it does. Thus the gas load anticipated during reactor start-up has the following composition: 4 to 8% $H_2$ and $O_2$ that are produced in the active section; 35% $N_2$ and $O_2$ from the air-saturated water; 57 to 61% water vapor. During continued reactor operation the contribution due to the $N_2$ and $O_2$ from air-saturated water is greatly reduced since only 50 gpm of demineralized water is added as make-up.

The rates of formation of $H_2$ and $O_2$ in the active section are related to the quantity of radiated energy that the water absorbs; an approximation is that one $H_2$ molecule is formed for every 100 ev absorbed, but this may vary severalfold depending on the purity of the water. The energy absorbed in the process water of the MTR at 30,000-kw operation is 28,500 kw, or $1.8 \times 10^{25}$ ev/sec.[24] The rates of formation of $H_2$ and $O_2$ are then, respectively, 6.4 and 3.2 liters/sec at NTP.

The following two tabulations summarize the operating conditions in the active lattice, and the solubility of gases at these conditions. The gas solubilities are calculated for the surface temperature of most of the transfer areas of the fuel rods.

**Active Lattice Operating Conditions**

| | |
|---|---|
| Temperature of entering water | 95°F |
| Temperature of leaving water | 115°F |
| Temperature of hot metal surface points | 240°F |
| Temperature of most metal surface points | 212°F |
| Absolute pressure | 1.5 atm. absolute |
| Process-water flow | 20,000 gpm |
| Reactor power | 30,000 kw |
| $H_2O_2$ concentration | 0.005 molar |

**Gas Solubilities at Active Lattice Operating Conditions: 212°F, 1.5 atm.**

| | |
|---|---|
| Nitrogen | 5.0 cc/liter |
| Oxygen | 3.0 cc/liter |
| Hydrogen | 5.5 cc/liter |
| | 13.5 cc/liter |

The gas solubilites of the water under the conditions that exist in the demineralizer tank and upon entering the process-water system are as follows:

**Gas Solubilities of Demineralized Water: 95°F, 1 atm.**

| | |
|---|---|
| Nitrogen | 9.4 cc/liter |
| Oxygen | 4.8 cc/liter |
| Carbon dioxide | 0.2 cc/liter |
| | 14.4 cc/liter |

The amounts of the gases remaining in the process water when it leaves the flash evaporator after having been degassed are given in the following tabulation. These values are calculated and indicate the approximate amounts expected during operation.

**Dissolved Gases in Degasified Water: 95°F, 1.8 in. Hg**

| | |
|---|---|
| Nitrogen | 0.10 cc/liter |
| Oxygen | 0.05 cc/liter |
| Hydrogen | 0.10 cc/liter |
| | 0.25 cc/liter |

In the following table are summarized the quantities of the various gases and the total quantity of gases that the vacuum steam jet must remove. It is necessary to remove the water vapor because it is carried along with the other gases.

**Vacuum Steam Jet Load**

| | |
|---|---|
| Gases formed due to water decomposition | |
| Hydrogen | 5 lb/hr |
| Oxygen | 35 lb/hr |
| Gases from air-saturated water | |
| Nitrogen | 110 lb/hr |
| Oxygen | 65 lb/hr |
| Water vapor | 425 lb/hr |
| | 640 lb/hr |

The above table is based on start-up conditions. During continued operation the gas load from air-saturated water will be negligible.

**7.4.3 Water Quality and Materials of Construction.** In selecting the quality of water used for cooling the Materials Testing Reactor, the guiding principle was to minimize the amount of total dissolved solids in the water so as to maintain high heat transfer coefficients at the fuel assemblies. Any deposition of material from the water onto the hot surfaces of the fuel assemblies could reduce the heat transfer coefficients to undesirable values.

Any constituents in the water that would act as catalysts in the corrosion of aluminum, beryllium, or stainless steel parts of the process-water system would also be undesirable. For this reason it is necessary to minimize the amounts of lead, nickel, copper, and chlorides in the water. An additional advantage of a low mineral content in the water is that the amount of radioactivity induced in the water is appreciably reduced, thus reducing shielding requirements of the system.

In the original heat transfer experiments it became apparent that the water system should be constructed of stainless steel in order to minimize corrosion and maintain a low mineral content in the water. Columbium-stabilized type 347 stainless steel was selected so as to facilitate welding. At a later point in the heat-transfer experimentation program, it was found feasible to construct the MTR process-water system of cast iron. However, because of the limited experimental program and the resulting uncertainties in the use of cast iron piping, the process-water piping is of stainless steel while cast iron pump casings are used so as to minimize cost.

In comparing the MTR process-water system with water systems of other reactors, it is found that the Materials Testing Reactor is the first recycling natural-water—cooled reactor. The reactors at Hanford utilize coagulated and filtered Columbia River water on a once-through basis, and thus the amount of induced activity in this water is small. The heavy water pile at Argonne and the higher-flux heavy water pile at Chalk River, Canada, have their heavy water recirculated in a tank surrounding the fuel assemblies. The volume of natural water in the MTR system is many times the volume ofwater in either the Chalk River or Argonne heavy water systems. It is interesting to note that both heavy water reactors were operated for several years without any water treatment, and in the course of this time, the amount of iron and aluminum hydroxides increased. The *p*H values of the water decreased to as low as 4, and the amount of induced radioactivity in the water increased to values that necessitated removal of the dissolved radioactive solubles. At Argonne ammonium hydroxide made from heavy hydrogen was added to the water to raise the *p*H and thus reduce its corrosive effects on the aluminum parts. However, this developed into a vicious cycle in that owing to ionizing radiations the ammonium hydroxide appeared to decompose to nitric acid, once more reducing the *p*H of the water in the system. The addition of more ammonium hydroxide was required with a consequent build-up of total dissolved solids. The

problem was solved at Chalk River by redistilling the $D_2O$ and then demineralizing the recirculating heavy water. This removed the accumulating dissolved minerals and helped increase the pH of the water to nearly 7, and at the same time eliminated any catalytic ions that might have caused corrosion in the heavy water system. Provisions have been made in the MTR system to add fresh demineralized water and to purge the same quantity of water from the system so as to maintain a high-quality circulating process water. It is expected that, when the MTR is in operation, the purge can be put through a demineralization system and used as make-up. With the recent development of mixed-bed deionizers, the water in the system can be brought to a resistivity of practically pure water, which is apparent from the following tabulation:

| NaCl (ppm) | RESISTIVITY (ohms) |
|---|---|
| 1.00 | 185,000 |
| 0.10 | 1,850,000 |
| 0.01 | 18,500,000 |
| 0.00 (pure water) | 20,500,000 |

Ever since the Hanford reactors have been operated, sodium dichromate and sodium silicate have been added to the single-pass water system to minimize corrosion, and at one time it was believed that additions of similar reagents should be made to the MTR water system. A few years ago the sodium silicate addition to the Hanford water was removed, and no increased corrosion or scaling rate was found to occur. Although the additions of sodium dichromate have never been stopped, it is believed that they could be without increasing corrosion or scaling.

The heat transfer experiments conducted at Oak Ridge differed from the MTR system in that the water used at Oak Ridge was cycled in an open system whereby oxygen and nitrogen of the air were readily dissolved. It is expected that any gases in the MTR process water will be practically completely removed by the flash evaporation process, and therefore the corrosion rates may be even less than those determined by the experiments conducted at ORNL.

**7.4.4 Induced Activity.** Elements and sources which contribute to the water activity can be listed under the following categories:

1. Isotopes of oxygen in pure water itself.
2. Solutes in the water such as mineral impurities or dissolved gases.
3. Radioactive and nonradioactive elements corroded from structures in the reactor into the water.
4. Radioactive elements knocked into the water by recoil upon the instant of neutron capture.

Radioactive isotopes formed from oxygen produce an enormous amount of activity but lose their controlling influence in about 1 min because of their short half-lives. These isotopes are the primary source of induced radioactivity in the MTR process water. The dissolved impurities in the water form the main secondary source of water activity. During the first minute the radiation products of oxygen dominate radiation; $Al^{28}$ (144 sec, 1.8 Mev) and $Mg^{27}$ (692 sec, 1.02 Mev) are the principal activities for the next 5 to 10 min. Thereafter, $Na^{24}$ (14.8 hr, 1.4 and 2.8 Mev) is the dominating radioactive isotope for the 15-min MTR water cycle.

The cooling water system of the MTR reactor is a closed recirculating system with a small percentage of purge and make-up water in order to control the intensity of long-lived induced activity and the accumulation of corrosion products.

When a recirculation system is employed, regardless of the method used for purge and make-up, the number of radioactive atoms ($N$) at equilibrium when production equals decay is given by[25]

$$N = \frac{nv\sigma N_v (1 - e^{-\lambda t_i})}{\lambda (1 - Fa)}$$

where

$nv$ = neutron flux, neutrons/cm$^2$-sec

$\sigma$ = absorption cross-section, cm$^2$

$N_v$ = concentration of target element, atoms/cc

$t_i$ = time of irradiation (time to pass through active section), sec

$$F = \frac{\text{process flow rate}}{\text{process plus purge flow rate}}$$

$$a = e^{-\lambda t_c}$$

$t_c$ = cycle time, sec

$\lambda$ = transformation constant, $sec^{-1}$

The recoils from the aluminum in the active section can be calculated by[8]

$$P_r = \frac{(nv)\sigma N_v(R)AD}{4G(1 - a)} \quad \text{dis/sec-cc}$$

where

$R$ = recoil range, cm

$A$ = area of exposed aluminum, $cm^2$

$D$ = density of aluminum, g/cc

$G$ = cycle flow rate, g/sec

The constants used in the activity calculation for the MTR reactor are:

| | |
|---|---|
| Flow rate | 20,000 gpm |
| Purge rate | 50 gpm |
| System holdup | 350,000 gal (now 320,000 gal) |
| Cycle time | 17.4 min |
| Irradiation time | 0.08 sec |
| Slow flux | $2 \times 10^{14}$ neutrons/$cm^2$-sec |
| Fast flux | $1 \times 10^{14}$ neutrons/$cm^2$-sec |
| Exposed aluminum | $10^6$ sq cm |

The desired composition of the MTR process water, which is the composition that is used in the activity calculation, is:

| | | | |
|---|---|---|---|
| Fe | 0.05 ppm | Cl | 0.25 ppm |
| Al | 0.05 ppm | Ca | 0.5 ppm |
| $CO_2$ | 2.0 ppm | Mg | 0.5 ppm |
| Na | 2.0 ppm | A | 0.6 ppm |
| $SiO_2$ | 1.0 ppm | | |

The results of the activity calculations are summarized in Table 7.4.A. The activities of the constituents in the process water and the total activity for water containing 2 and 8 ppm of Na are given as functions of decay time.

**7.4.5 Shielding.** The shielding calculations for the MTR process water system were based on the induced activities given in the previous section. In the calculations involving the process water piping it was assumed that the source was an infinitely long cylindrical source with self-absorption. The results of the calculations are given in Fig. 7.4.A, which shows the thickness of concrete shielding as a function of time. Separate curves are given for water containing 2 ppm of Na and 8 ppm of Na.

The following discussion gives a brief summary of the shielding of the process-water system. The intensity of radiation from the process water leaving the MTR active lattice requires 5 ft of concrete shielding within the Reactor Building. Within the reactor this shielding is provided by the biological shield; outside the reactor the pipe tunnel underneath the basement floor has sufficient concrete cover to meet the requirements. From outside the Reactor Building to the Process Water Building, 5 ft of earth covering bring the inlet and outlet process water pipes below the local frost line; this depth is sufficient to protect personnel from radiations above tolerance. In the Process Water Building the 12- to 18-in. concrete walls are adequate to reduce radiations from the pipes, valves, etc., to less than 50 mr/hr. No special shielding is required on the working reservoir owing to its height above the ground. A fence about the working reservoir base provides adequate protection by limiting the approach of personnel so that maximum daily tolerances are not exceeded.

**7.4.6 Cooling After Shutdown.** After an operating reactor is shut down, the total heat liberated due to the absorption of gamma and beta radiation

## TABLE 7.4.A

### Activity of HTR Process Water

| REACTION | CONCEN-TRATION (ppm) | HALF-LIFE | CROSS-SECTION | RECOIL TYPE | RANGE (cm) | ACTIVITY (dis/cc-sec) 0 sec DECAY | 50 sec DECAY | 100 sec DECAY | 500 sec DECAY | 1000 sec DECAY |
|---|---|---|---|---|---|---|---|---|---|---|
| $O^{16}(n,p)N^{16}$ | | 7.4s | 0.04 mb | | | $4.3 \times 10^5$ | 3,843 | 35 | | |
| $O^{18}(n,\gamma)O^{19}$ | | 29.4s | 0.44 mb | | | 5,620 | 1,730 | 530 | | |
| $Al^{27}(n,\gamma)Al^{28}$ | | 2.3m | 0.21 b | Slow | $10^{-6}$ | 3,040 | 2,355 | 1,840 | 332 | 32 |
| $Al^{27}(n,p)Mg^{28}$ | | 10.2m | 2.8 mb | Fast | $10^{-4}$ | 1,320 | 1,320 | 1,320 | 850 | 488 |
| $Al^{27}(n,\gamma)Al^{28}$ | | 2.3m | 0.4 mb | Fast | $10^{-4}$ | 576 | 452 | 352 | 64 | 6 |
| $Na^{23}(n,\gamma)Na^{24}$ | 2 | 14.8h | 0.5 b | | | 406 | 406 | 406 | 406 | 406 |
| $Al^{27}(n,\alpha)Na^{24}$ | | 14.8h | 0.6 b | Fast | $10^{-4}$ | 163 | 163 | 163 | 163 | 163 |
| $Mn^{55}(n,\gamma)Mn^{56}$ | | 2.6h | 13 b | Slow | $10^{-3}$ | 145 | 145 | 145 | 145 | 145 |
| Total (2 ppm Na) | | | | | | $4.8 \times 10^5$ | 10,414 | 4,791 | 1,960 | 1,241 |
| (8 ppm Na) | | | | | | $4.8 \times 10^5$ | 13,592 | 6,688 | 3,800 | 2,500 |

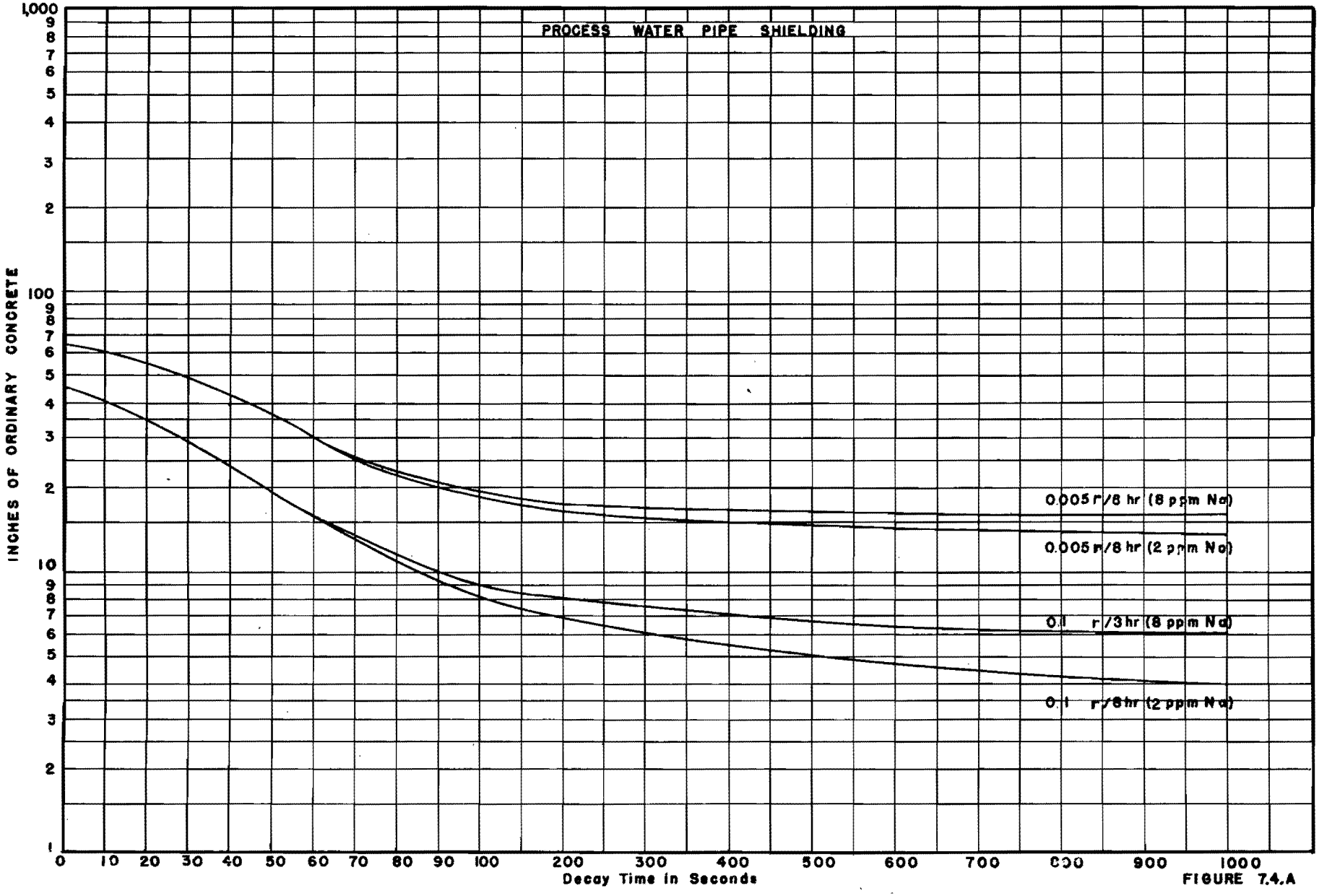


FIGURE 7.4.A

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from the fission products is given by the empirical equation[23]

$$P = 187BFA[t^{-0.2} - (t + T)^{-0.2}] \qquad \text{Mev/sec}$$

where

$B$ = reactor operating level before shutdown, kw

$F$ = conversion factor, fissions/sec-kw

$A$ = Reduction factor; a reasonable value is 0.08, which is an extrapolation to $t = 0$ from experimental measurements of energy after shutdown;[12] also, since at shutdown 14.7 Mev/fission of beta and gamma heat is released compared to 187 Mev/fission during operation, the ratio of the two heat values is 0.08 (i.e., 14.7/187).

$t$ = time after shutdown, sec

$T$ = time of pile operation, sec

From the above it is found that at the instant of shutdown the reactor heat falls from 30,000 to 2400 kw, of which 400 kw is in the beryllium reflector. The reactor active lattice heat development at shutdown falls from full power of 28,800 to 2000 kw or about a 1/15 reduction. Hence, although the water flow through the active lattice at shutdown can be reduced about 1/15, the water flow through the beryllium reflector can be reduced only 1/3 since its heat generation falls only from 1200 to 400 kw. Depending on the amount of excess water flow through the beryllium at full power, the total flow can be reduced from 1/3 to 1/15 at shutdown. At any time $t$ after shutdown, the flow required will vary as $t^{-0.2}$. Calculations involving the varying film heat transfer coefficient and varying flow show that with a 220°F film temperature and 90°F entering water, 270 gpm must pass through the active lattice at the instant of shutdown. This gives a 50°F water temperature rise. The ratio of flow in the active lattice to that in the beryllium is estimated as 3:1. The following table gives the minimum process water flow required at various times after shutdown (Table 7.4.B).

TABLE 7.4.B

**Minimum Process-water Flow After Shutdown**

| TIME AFTER SHUTDOWN (hr) | MINIMUM FLOW (gpm) |
|---|---|
| 0 | 360 |
| 1 | 267 |
| 5 | 190 |
| 10 | 168 |
| 50 | 126 |

The shutdown cooling-water flow can be initially supplied by the reserve water in the working reservoir and, when the change-over can conveniently be made, by a 1000-gpm pump circulating water through a circuit including only the reactor and seal tank as major components. There are two 1000-gpm pumps supplied; one is electric motor driven while the other is gasoline engine driven.

It is of interest to show that the water in the closed seal-tank and reactor-tank circulating system will not reach a prohibitive temperature too quickly as a result of accumulation of shutdown heat. This heat is given by the equation

$$q = \int_0^t 30{,}000 \times 0.08 \left[t^{-0.2} - (t + T)^{-0.2}\right] dt \quad \text{kw-sec}$$

where 30,000 kw is the operating power level of the MTR.

The water temperatures given in the following table are based on this equation, a pile operating period of 15 days, and a 24,000-gal content of the circulating system.

TABLE 7.4.C

**Process-water Temperature After Shutdown**

| TIME AFTER SHUTDOWN (hr) | $q$ (Btu x $10^{-6}$) | WATER TEMPERATURE (°F) |
|---|---|---|
| 10 | 7.5 | 110 |
| 20 | 12.0 | 130 |
| 30 | 15.5 | 150 |
| 40 | 18.5 | 164 |
| 50 | 21.5 | 174 |

Water from the demineralized-water storage tank can be added to the reactor-seal tank system at any time in order to cool this cycling water.

## 7.5 REFERENCE REPORTS


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(20) Lane, J. A., *Cooling Requirements at Reduced Power Level*, ORNL CF-50-9-92, September 19, 1950.

(21) Farmer, W. S., *Activities in Process Water*, ORNL CF-47-4-115, April 7, 1947.

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(23) Way, K., *Pile Technology Lecture No. 16*, M-3690.

(24) Etherington, H., *Water Cooled Pile for Naval Application*, ORNL-133, Septemper 1, 1948.

(25) Sivetz, M., "Radioactivity Induced in Water," *Atomics*, Vol. 6, No. 5, p. 4 (Sept. - Oct., 1950).

## 7.6 REFERENCE DRAWINGS

(Blaw-Knox Construction Company Drawings)

| DWG. NO. | TITLE |
|---|---|
| 3150-5-4 | Process Water Pump House - Layout - Elevations |
| 3150-5-5 | Process Water Cooling Building - Plans |
| 3150-5-6 | Process Water Cooling Building - Plans and Sections |
| 3150-5-7 | Process Water Cooling Building - Layout and Sections |
| 3150-19-1 | Reservoir and Water Pump House - Layout |
| 3150-92-6 | Process Water System |
| 3150-304-1 | Working Reservoir |
| 3150-702-2 | Process Water Piping |
| 3150-703-11 | Piping - Internal Plan and Elevation - Process Water at Reactor |
| 3150-703-12 | Piping - External Plan and Elevation - Process Water at Reactor |
| 3150-802-65 | Valve Pits - Process Piping Plan - Sections and Details |
| 3150-803-39 | Process Water Pipe Tunnel - Support Details and Piping |

## Chapter 8*

# REACTOR COOLING-AIR AND CONTAMINATED-AIR SYSTEMS

The reactor cooling-air system and the contaminated-air system are grouped together because the air from both is exhausted to the atmosphere through the main stack. However, they serve entirely different purposes. The reactor cooling-air system provides a flow of air through the reactor for cooling the graphite reflector; it is a secondary reactor-cooling system. The contaminated-air system removes radioactive gases from the hoods and caves of the laboratories.

## 8.1 REACTOR COOLING-AIR SYSTEM

**8.1.1 Introduction.** Immediately surrounding the reactor tank is the graphite reflector. This reflector is composed of a removable bed of 1-in.-diameter graphite balls adjacent to the tank and a section of graphite blocks between the balls and the thermal shield. Heating due to the absorption of gamma radiation makes it necessary to cool the graphite reflector. Because of relatively low heat generation, air cooling is satisfactory. The cooling and design of the graphite reflector are discussed in other sections. This section will deal with the flow path and processing of the air going through the reactor cooling-air system.

A schematic diagram of the reactor cooling-air system is shown by Fig. 8.1.A. The description of the system will in general follow the air flow path, starting with the inlet air filters at the reactor faces and ending with the stack.

**8.1.2 Air Flow Data.** The reactor is designed for normal operation at a power level of 30,000 kw. The design air flow of 2000 lb/min and pressure drop of 55 in. of water through the reactor cooling-air system are based on maximum temperatures of 570°F in the graphite pebble bed and 500°F in the permanent graphite at 45,000-kw operation of the reactor. The 2000-lb/min

* ANL contribution.

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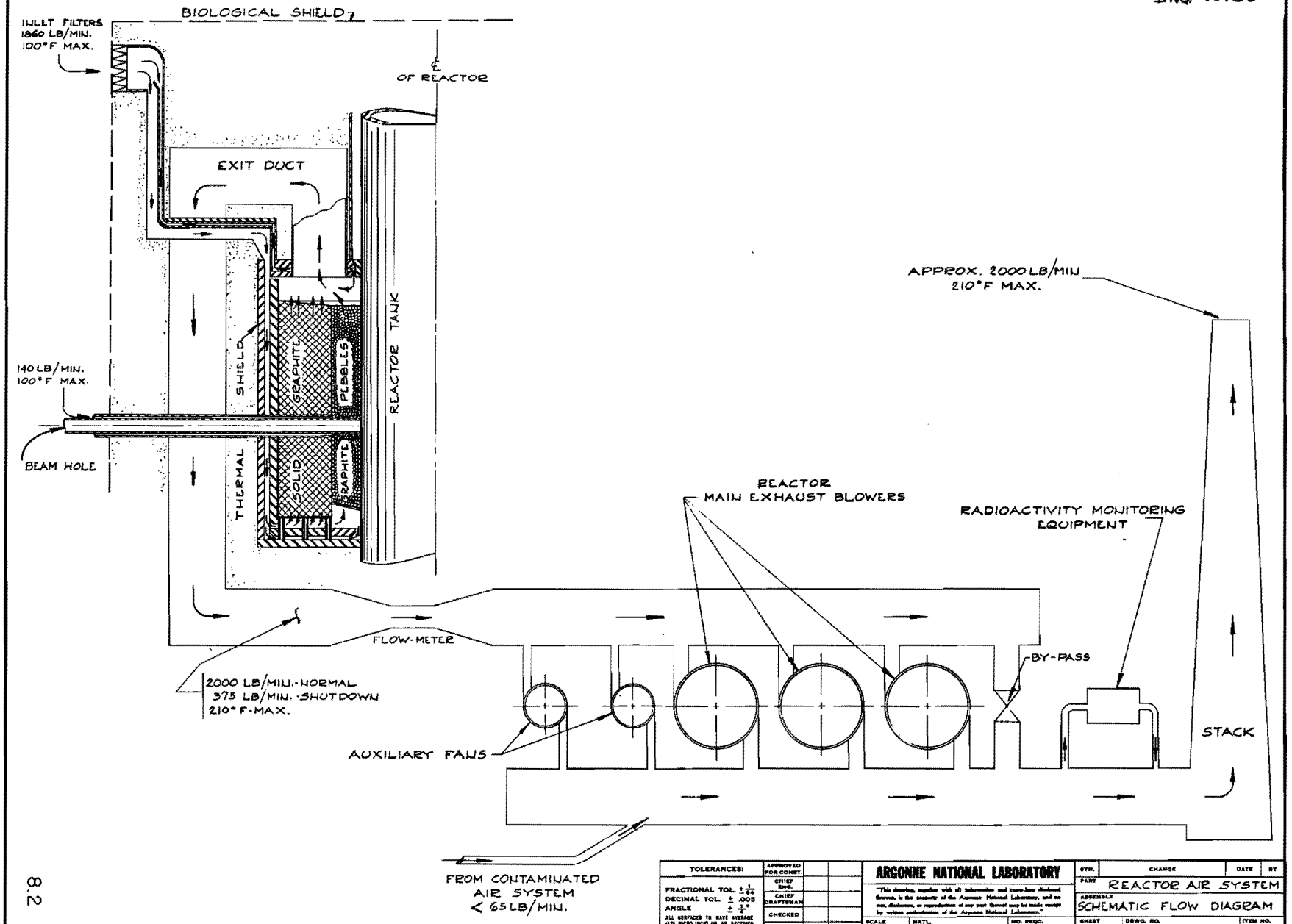



| TOLERANCES: | APPROVED FOR CONST. | | ARGONNE NATIONAL LABORATORY | SYM. | CHANGE | DATE | BY |
|---|---|---|---|---|---|---|---|
| FRACTIONAL TOL. ± 1/64 | CHIEF ENG. | | "This drawing, together with all information and know-how disclosed thereon, is the property of the Argonne National Laboratory, and no use, disclosure, or reproduction of any part thereof may be made except by written authorization of the Argonne National Laboratory." | PART | REACTOR AIR SYSTEM | | |
| DECIMAL TOL. ± .005 | CHIEF DRAFTSMAN | | | ASSEMBLY | SCHEMATIC FLOW DIAGRAM | | |
| ANGLE ± 1/2° | CHECKED | | | SCALE | MATL. | NO. REQD. | SHEET / OF | DRWG. NO. | ITEM NO. |
| ALL SURFACES TO HAVE AVERAGE (125 RMS) ROUGHNESS OR AS DESIGNED FINISH. UNLESS OTHERWISE NOTED | DRAWN | V.H. 11-2-50 | | NONE | — | — | | FIG. 8-1-A | |
| | GROUP HUTLER | REPRESENTATIVE LEYSE | | | | | | | |

design flow consists of the following quantities:

| | |
|---|---|
| Air flow through graphite | 1610 lb/min |
| Air to cool top thermal shield | 250 lb/min |
| Allowed for leakage inward through experimental holes, etc. | 140 lb/min |
| Total | 2000 lb/min |

Subsequent calculations gave the air requirements as 1526 lb/min to cool the graphite and experimental facilities, and 174 lb/min to cool the top thermal shield, for a total air flow of 1700 lb/min. Also, critical experiments conducted at ORNL indicated that the heat generation in the reflector would be somewhat less than that used for the above calculations and the air requirements would therefore be less. However, the design figures were maintained at 2000 lb/min flow and 55 in. of water pressure drop since it may be desirable and possible to operate the reactor at a power level as high as 60,000 kw at some time in the future.

The approximate pressure drops through the reactor air system for a flow of 2000 lb/min are summarized in Table 8.1.A.

**TABLE 8.1.A**

**Approximate Pressure Drops in Reactor Air System**

| PART OF SYSTEM | PRESSURE DROP (in. $H_2O$) |
|---|---|
| Inlet filters and ducts | 1.6 |
| Thermal shield | 0.4 |
| Graphite reflector | 29 |
| Exit ducts and stack | 5.4 |
| Total | 36.4 |

The total pressure drop of 36.4 in. of water given above is considerably lower than the 55 in. of water design value. Two factors enter to account for the difference. First, the reactor cooling-air system is designed without any filters on the exit air stream. If it should become necessary to add

filters in the future, additional head will be required. Based on the filter system used on the X-10 pile at ORNL, the additional head allowed for exit filters is about 8 in. of water. Second, to assure that the air system will be adequate for operation at a power level of 60,000 kw, the design head has been further increased to 55 in. of water.

The above discussion has been concerned with the air flow through the reactor air system during operation of the reactor. It is also necessary to maintain a flow of air through the reactor after it has been shut down. This flow serves to remove heat from the graphite reflector generated by the absorption of gammas from fission product decay and from induced activity, and also to prevent the escape of radioactive dust into the Reactor Building during removal of experimental hole plugs. To provide adequate cooling of the reactor after shutdown, a flow of about 135 lb/min is required. The flow necessary to ensure that radioactive dust does not escape into the Reactor Building is based on an air velocity of 150 ft/min directed inward along the annulus between a plug and liner. The air flowing into this annulus flows through holes at the inward ends of the liners and joins the reactor-cooling air flowing upward through the graphite pebbles. A satisfactory design of the experimental hole liners and plugs is obtained when the flow through the reactor is about 400 lb/min; therefore, this is the design flow through the reactor during shutdown. This is also the flow desired through the reactor in the event of an emergency shutdown of the reactor. The pressure drop through the graphite for a flow of 400 lb/min is about 0.5 in. of water, and the design pressure drop through the entire system is about 2 in. of water.

Once the reactor has been placed in operation there will always be active particles within the air passages. To prevent these active particles from entering the Reactor Building, a basic requirement of the reactor air system is that there always be a flow of air in the proper direction. Upon failure of the main exhaust blowers and the auxiliary fans, this flow will be induced by whatever stack draft is available.

After the reactor has been in operation long enough so that the reactor, reactor air system, and stack are hot, there is a stack draft of about 0.5 in. of water available. This is based on air temperatures of 200°F in the stack and 90°F atmospheric. It is estimated that this draft would produce a flow of 2000 to 3000 cfm through the reactor in the event of failure of all blowers

and fans. This flow is sufficient to meet shutdown cooling requirements. However, because of the possibility of a complete power failure within a short time after reactor start-up, in which case there would be essentially no stack draft, and because of the problem of radioactive dust escaping into the Reactor Building, the reactor cooling-air system does not rely on stack draft to produce the necessary shutdown air flow. This flow is provided by either of the auxiliary fans. Only as a last resort, i.e., in the event of complete failure of the blowers and fans, is the available stack draft utilized.

ADDITIONAL DATA

| | |
|---|---|
| Air temperatures | |
| Inlet to reactor | 75 - 100°F |
| Outlet from reactor | 185 - 210°F |
| Site elevation | 4930 ft |
| Barometric pressure | 12.25 psi |

**8.1.3 Inlet Air Filters.** Air used for cooling a reactor will emerge radioactive. The activity is due partially to gaseous atoms naturally occurring in air, such as argon, and partially to dust in the air. Not much can be done to reduce the activity from the first, but the latter can be reduced considerably by suitable filtering.

In the first considerations of the radioactivity problem of the MTR cooling-air system, a filter design similar to that used for the pile at ORNL was included in the exhaust side of the system. Further consideration of the problem along with economy measures prompted omission of the filters in the exhaust side of the reactor cooling-air system.

The only filters included in the reactor air system are the glass wool filters at the inlets to the system. A brief description of the processing the air encounters before entering the reactor is now given.

Normal atmospheric air enters the Reactor Building ventilation system through coarse-fiber-glass or wire-mesh prefilters. The air then goes through an electrostatic filter of 90% discoloration efficiency before passing into the Reactor Building. The Reactor Building is pressurized to about ½ in. of water so that leakage in through windows, doors, etc. will be kept to a

minimum. The air entering the reactor is taken directly from the Reactor Building and is filtered at the inlets by glass wool filters. There is an additional small amount of air (about 2%) which enters the air system via experimental holes without passing through the glass wool filters.

The elimination of the exhaust side filters, from the technical standpoint, is justified principally by a comparison of the MTR cooling-air system with that of the NRX pile at Chalk River, Ontario. The NRX pile has been operated satisfactorily since about September, 1946, without exhaust air filters. The air systems of the two reactors are similar in the following respects:

1. The major portion of air entering the reactor has been filtered to some degree.
2. The air does not make direct contact with the fuel elements, and thus fission products from ruptured fuel elements will not enter the air stream.
3. The air flows in direct contact with the graphite reflector.
4. The air flows in direct contact with the thermal shield; however, the NRX shield is cast iron while the MTR shield is low-carbon steel.
5. The air is susceptible to pollution from accidents occurring in the experimental facilities.
6. The exhaust air is not filtered.
7. The levels at which the effluent air is ejected into the atmosphere are 200 ft for the NRX pile and 250 ft for the MTR.

Rather extensive investigations have been conducted at the NRX with the finding that it is not necessary to add exhaust filters to the NRX air system. Based on this finding, the above general similarities, a more detailed comparison of the NRX pile and the MTR, and economics, the exhaust filters were omitted from the MTR air system. Should it be necessary to add exhaust filters at some later date, there is sufficient space for them adjacent to the Blower and Fan House.

The inlet filters are mounted in the inlets in the reactor faces. The filters are Airmat Type PL-24 units made by American Air Filter Company. Six units are mounted in a 2- by 12-ft opening in each pile face, giving a total of twenty-four 2- by 2-ft units. The face velocity of the air entering the

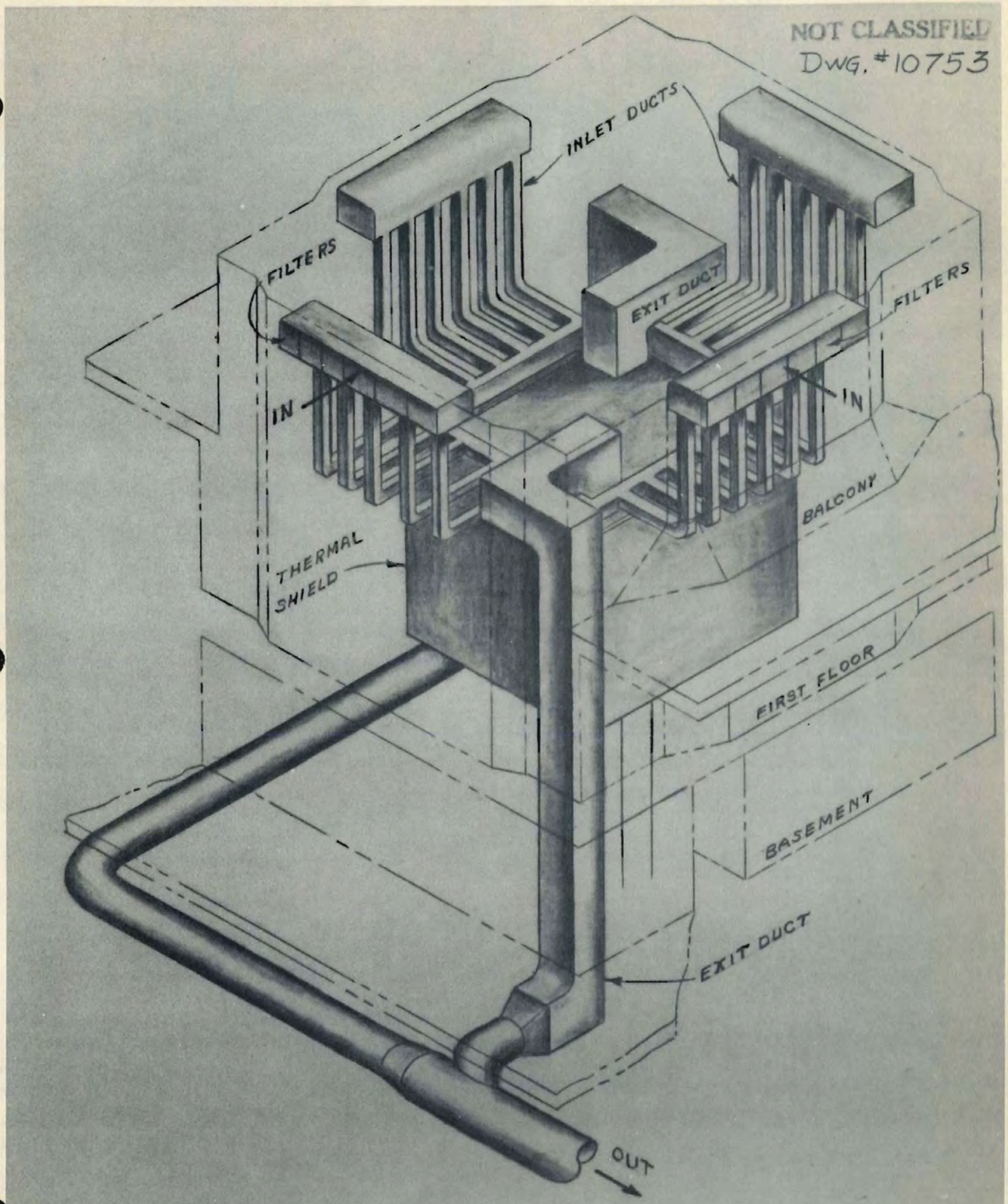
filters is about 350 ft/min. The pressure drop through the clean filters is approximately 0.1 in. of water.

**8.1.4 System Within Reactor Structure.** The system of duct work within the reactor is shown by Fig. 8.1.B, and the schematic flow of air within the reactor is shown by Fig. 8.1.A. In the design of the system within the reactor it was necessary to meet the following requirements:

1. The air must enter the pile structure at a level higher than 8 ft above the centerline of the active lattice, and the exit ducts must loop upward to a level at least that high in order for it to be possible, in so far as the air system is concerned, to flood the interior of the thermal shield with water.
2. All ducts must be arranged within the shield so that radiation outward, both in straight-line paths and in "shine" through the ducts, will not be sufficient to bring radiation at any point outside the shield above permissible levels.
3. The air must enter the thermal shield around the outer edge of its top and pass downward, cooling the inner and outer walls and possibly the top and bottom of the thermal shield before passing into a plenum chamber under the graphite. The air then must flow upward through the graphite to a plenum chamber above the graphite and out through exit ducts.
4. Exit air ducts must be shielded from the outside by the equivalent of 1 ft of ordinary concrete.
5. Arrangement of the air ducts within the reactor structure must not interfere with any experimental facilities.

The design shown in Figs. 8.1.A and 8.1.B essentially meets all the above requirements.

The inlet system consists of four sets of filters, plenum chambers, ducts and manifolds, one set in each wall of the reactor structure. The air enters each set through louvers located near the top of each pile face. Back of each set of louvers are six 2-ft-square filters and a 2- by 2- by 12-ft plenum chamber. Each plenum chamber is joined to the top edge of the thermal shield by five or six 8-in.-square ducts and a manifold which is continuous around the top of the thermal shield. A separate channel with a damper is provided in each of the 8-in. ducts whereby a small amount of air is by-passed for cooling the top thermal shield. There is a total of 22 8-in. ducts. Each duct has one 90° elbow and connects to the plenum chamber and the thermal shield manifold

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MTR AIR SYSTEM
INSIDE REACTOR STRUCTURE

FIGURE 8.1.B

at 90° to the principal direction of flow. Thus there are effectively three 90° turns in the inlet air system. The walls of the ducts are lined with 1-in. steel plate to reduce gamma-ray leakage through the ducts.

The path followed by the air within the reactor proper is shown in Fig. 8.1.A. The main air stream flows down between the walls of the thermal shield, and some then passes directly into a plenum chamber beneath the graphite while a portion flows between the top and bottom plates of the bottom thermal shield before passing into the plenum chamber. The air from the chamber passes up through the graphite pebbles and cooling holes in the permanent graphite and is joined by air entering from the experimental facilities. The air enters a plenum chamber above the graphite where it is joined by the air by-passed to cool the top thermal shield. The air then enters the exit air ducts.

The exit air duct system within the reactor consists of two 30-in.-square ducts connected to the thermal shield at diagonally opposite corners of the top plate and turned downward through the biological shield to carry the air out below the basement floor. There are three 90° bends in each of these ducts, as shown in Fig. 8.1.B.

**8.1.5 Exit Ducts.** The 30-in.-square ducts leading down through the reactor connect to a single 48-in.-diameter steel duct beneath the basement floor. This duct connects to a plenum chamber outside the Reactor Building. An underground duct of 48-in.-diameter precast concrete pipe leads from this plenum chamber to a plenum chamber beneath the Blower and Fan House. The concrete duct is a minimum of 2 ft below grade so that the earth cover provides adequate shielding against radiation.

**8.1.6 Blowers, Fans, and Auxiliaries.** The blowers and fans of the reactor air system are housed in the Blower and Fan House, which is described in a later section; the reactor-air-system equipment contained in it is described here.

The blowers and fans for the reactor air system take their air from an inlet plenum chamber beneath the building and exhaust it to another, also beneath the building.

*Blowers.* There are three half-capacity blowers in parallel, one of which serves as a spare, for cooling during normal operation of the reactor. Each

blower is capable of delivering 1000 lb/min of 210°F air against a head of 55 in. of water, and is driven by a 250-hp motor.

The blowers are of the positive displacement type, this type being selected because of its adaptability to wide ranges of flow rates and pressure heads. This is important since, while it is desired to have a flow rate of 2000 lb/min, the pressure drop through the system initially will be considerably below the design value of 55 in. of water. Also, since the cooling requirements may be below design calculations, it is desirable for the units to operate economically at reduced flows and heads. The positive displacement type of blower offers the most economical operation at the reduced requirements.

The blowers are electric-motor driven and receive power from the normal commercial electrical supply only. Extra pulleys are provided with the blowers to provide for the slower speeds at reduced requirements. These pulleys will adequately cover the range from 700 to 1000 lb/min.

*Auxiliary Fans.* Besides the blowers there are two 400-lb/min fans for emergency and normal shutdown operation, each fan being in parallel with the three main blowers. One of the fans is driven by an electric motor and the other by a gasoline engine. The motor-driven fan receives its power from the normal commercial electrical supply and from the site emergency power supply and will be used during normal shutdown periods. The gasoline-driven fan will be used for emergency shutdowns when both electrical supplies have failed.

*Valves.* There are butterfly valves at both inlets and outlets of all blowers and fans. The valves on the inlet sides are manually operated while the valves on the exhaust sides are air operated. The valves on the exhaust sides can seal airtight to permit removal of a blower or fan without outward leakage of radioactive air. The valves on the inlet sides do not have to be absolutely airtight since any leakage here will be directed into the reactor air system.

Since the blowers are of the positive displacement type, it is necessary to have both the inlet and outlet valves open when the blowers are operating. To ensure this, limit switches are provided on both inlet and outlet valves. These switches require both valves to be open before the blowers can be started.

Also, because of the use of positive displacement blowers, it was necessary to provide a by-pass around all the blowers and fans so that the stack draft will be available to provide a flow of air through the reactor air system. The by-pass is provided by a valve between the plenum chamber ahead of the fans and blowers and the exit duct leading to the stack. This valve will be operated manually.

**8.1.7 Instrumentation and Control.** In the description of the instrumentation and control the flow path of the air will again be followed, starting with the inlets to the system.

There are four differential pressure indicators to determine the pressure drop across the glass wool inlet filters. They are mounted locally on the pile faces. Each indicator determines the pressure drop across the filters on one side of the pile.

In the exit duct near the reactor is a pressure indicator to determine the pressure in the air system at that point. The indicator is remotely mounted in the Reactor Control Room and is equipped with an alarm to notify the operator when the pressure has risen above a certain predetermined value.

Mounted in the concrete pipe line leading to the Blower and Fan House is a venturi tube for determining the flow rate through the system. The flow rate is indicated on a panel in the Blower and Fan House. Since the Blower and Fan House is an unattended station which will have periodic inspections only, the information for operation of the reactor air system is transmitted to the Process Water Building and the Reactor Control Room. The Process Water Building serves as the center for operational information on the reactor air system. The flow rate is therefore recorded at the control panel in the Process Water Building, and there are low flow rate alarms at both this panel and the panel in the Reactor Control Room. Besides the above, the venturi information is transmitted to the reactor control system to automatically shut the reactor down on failure of the air system.

In the exit air duct near the Blower and Fan House is a pressure indicator to measure the pressure in the system at this point. The information is indicated on a panel in the Blower and Fan House.

The valves on the inlet sides to the blowers and fans are manually operated. All of them will normally be open, being closed only if necessary to repair or remove a blower or fan.

The main exhaust blowers are started manually within the Blower and Fan House. To promote safe operation of the reactor air system, both of the auxiliary fans start automatically on failure on the main blowers, and one may then be shut down manually. However, in the event of failure of the one left running, the other will automatically start again. The bearings on the main blowers and their motors are provided with thermocouples to indicate bearing temperatures and give an alarm at the panel in the Blower and Fan House. The high bearing temperature is also indicated on a master alarm at the Process Water Building panel.

A pressure indicator is provided in the duct on the exhaust side of the blowers and fans. The pressure is indicated at the panel in the Blower and Fan House and is for operational information.

There are radiation-measuring instruments in the main duct leading to the stack. These instruments are located toward the stack from the point where the line from the contaminated-air system enters this duct. Thus the instruments measure the total activities of the gases going up the stack. In general, the contributions from the contaminated-air system will be small or nil, and the measurements are essentially those of the reactor air system. The total activity of the gases to the stack is determined by an ion chamber, while that part due to active dust particles is determined by a particulate-activity meter. The information from both is transmitted to recorders in the radiation-instrument area of the Blower and Fan House. Alarms to indicate excessive activity going up the stack are provided at the Blower and Fan House, Process Water Building, and the Reactor Control Room panels.

**8.1.8 Shielding.** The shielding requirements of the reactor air system are based on an air flow of 30,000 cfm through the reactor at 60,000-kw operation. These assumptions give shielding thicknesses which should be adequate for all reactor operating conditions. To correspond to the 60,000-kw operating level, an average flux of $4 \times 10^{12}$ neutrons/cm$^2$-sec throughout the volume within the thermal shield was assumed. The volume of air subject to activation by this flux is approximately 850 cu ft.

An examination of the elements normally occurring in air shows that it is necessary to consider only the active atoms $N^{16}$, $O^{19}$, and $A^{41}$, which result from neutron capture by $N^{15}$, $O^{18}$, and $A^{40}$, respectively.

In Table 8.1.B are summarized the unit source intensities due to the active atoms, along with other pertinent information.

**TABLE 8.1.B**

**Activity of Reactor-cooling Air**

| ORIGINAL ATOM | ABUNDANCE IN REACTOR AIR AT 150°F (atoms x $10^{-23}$/ft$^3$) | ACTIVATION CROSS SECTION (cm$^2$) | ACTIVATION RATE (atoms/sec-ft$^3$) | ACTIVE ATOM | UNIT SOURCE STRENGTH (Mev/sec-ft$^3$) | $\gamma$ ENERGY (Mev) |
|---|---|---|---|---|---|---|
| $N^{15}$ | 0.032 | $0.02 \times 10^{-27}$ | $2.56 \times 10^5$ | $N^{16}$ | $3.12 \times 10^5$ | 6.5 |
| $O^{18}$ | 0.0045 | $0.2 \times 10^{-27}$ | $3.5 \times 10^5$ | $O^{19}$ | $7.8 \times 10^5$ | 1.6 |
| $A^{40}$ | 0.049 | $0.62 \times 10^{-24}$ | $7.2 \times 10^9$ | $A^{41}$ | $4 \times 10^6$ | 1.37 |

Using the source intensities given in Table 8.1.B the shielding requirements were determined with the use of standard shielding formulas. These are summarized in Table 8.1.C. The shielding thicknesses which are mentioned in other sections generally exceed the values given in Table 8.1.C.

**TABLE 8.1.C**

**Shielding Requirements for the Reactor Air System**

| ITEM | TOLERANCE (r/8 hr) | SHIELDING THICKNESS | |
|---|---|---|---|
| | | CONCRETE (in.) | STEEL (in.) |
| Ducts in reactor building | | | |
| 2.5-ft square | 0.005 | 11 | |
| 4-ft diameter | 0.005 | 13.5 | |
| Ducts outside | | | |
| 4-ft diameter | 0.05 | 6.5 | |
| Main blower cells | | | |
| Outside walls (north and south) | 0.005 | 4 | 1 |
| Outside walls (east and west) | 0.005 | 8 | |
| Intercell walls | 0.05 | * | |
| Floor above inlet plenum chamber | 0.05 | 3 | |
| Floor above outlet duct to stack | 0.05 | 3 | |
| Roof above cells | 0.1 | * | |
| Auxiliary fan cells | | | |
| Floor above inlet plenum chamber | 0.05 | 2 | |
| Floor above exit duct | 0.05 | 2 | |
| All other walls | | * | * |

*Structural requirements only must be met.

## 8.2 CONTAMINATED-AIR SYSTEM

**8.2.1 Introduction.** The contaminated-air system removes radioactive gases and acid vapors from the hoods and caves of the laboratories. This is supplemental to the normal hood-ventilation system which removes air that is relatively free from acid vapors and radioactivity. The off-gas openings of the contaminated-air system provide a relatively high-velocity stream to remove the active gases and acid vapors. Figure 8.2.A shows a schematic flow diagram of the contaminated-air system. In general the description of the system will follow the flow diagram.

**8.2.2 Air Flow Data.** The system is based on the removal of a minimum of 50 cfm of gases from each hood, with a maximum of 10 hoods in use at any time. The system is sized for 500 cfm up to and including the scrubber. This is adequate to provide for a considerable increase over the present number of 15 hoods, since it is expected that normally only about 15% of the total off-gas openings would be in use at any one time. The capacity of the system after the scrubber is 1000 cfm; this is to provide for the future disposal of additional small volumes of radioactive gases. The pressure drop through the system is approximately 25 in. of water.

**8.2.3 System from Hoods to Scrubber.** The off-gas openings in the hood are 2 in. in diameter and are provided with quick-opening caps. The lines leaving the hoods are of 2 in. diameter. They connect to a 6-in. header leading to the scrubber. All the ducts ahead of the scrubber are of acid-resistant material. No shielding is provided for the ducts since the activity contained within them will generally be small unless an appreciable build-up occurs owing to deposits in the ducts. If the latter occurs, remedial steps will be necessary.

**8.2.4 Scrubber.** The purpose of the scrubber is to remove the acid vapors from the gases. It is not important that radioactivity be removed since the gases are ejected to the atmosphere through the main stack.

The scrubber has a packed column of Raschig rings. The removal of the acid vapors from the gases flowing up through the column is accomplished by the flow of a caustic solution down through the rings. The contaminated-air inlet line to the scrubber extends down to below the minimum operating level in the reservoir to prevent a back flow of vapors or radioactive gases to the hoods in the event of failure of the exhaust fans in the system.

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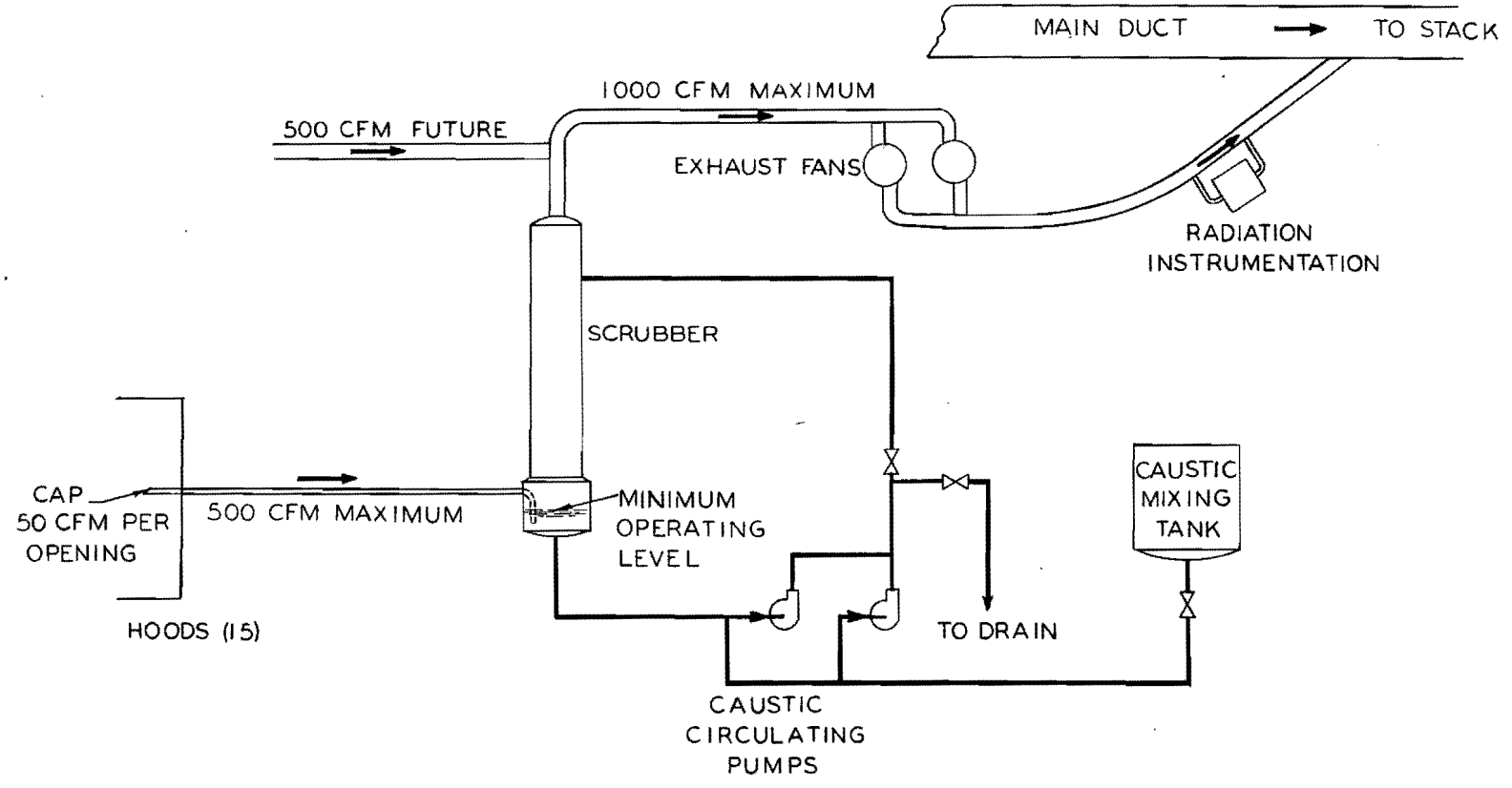


| TOLERANCES: | APPROVED FOR CONST. | | ARGONNE NATIONAL LABORATORY | SYM. | CHANGE | DATE | BY |
|---|---|---|---|---|---|---|---|
| FRACTIONAL TOL. ± 1/64 | CHIEF ENG. | | "This drawing, together with all information and know-how disclosed thereon, is the property of the Argonne National Laboratory, and no use, disclosure, or reproduction of any part thereof may be made except by written authorization of the Argonne National Laboratory." | TITLE CONTAMINATED AIR SYSTEM | | | |
| DECIMAL TOL. ± .005 | CHIEF DRAFTSMAN | | | ASSEMBLY SCHEMATIC FLOW DIAGRAM | | | |
| ANGLE ± 1/4° | CHECKED | | | | | | |
| ALL SURFACES TO HAVE AVERAGE (125 MICRO INCHES OR AS RECEIVED FINISH. UNLESS OTHERWISE NOTED | DRAWN FT 11-3-50 | | SCALE / MAT'L. / NO. REQD. | SHEET OF | DRWG. NO. FIG. 8.2.A | | ITEM NO. |
| | GROUP HUTTER | REPRESENTATIVE LEYSE | | | | | |

The caustic-circulating system for the scrubber has two pumps, one of which serves as a spare. The caustic solution for the scrubber is supplied from a caustic mixing tank where the solution is prepared as required to replace the depleted caustic solution in the scrubber. To provide for periodic sampling of the caustic solution a tap is provided at the bottom of the reservoir.

Since the amount of radioactivity which the scrubber may contain at any one time is very indefinite, the scrubber is not provided with shielding. Again, if necessary, this can be readily provided at a later date.

**8.2.5 Exit Duct from Scrubber.** The exit duct from the scrubber is an 8-in.-diameter carbon-steel pipe with suitable protective covering. The duct leads underground from the Reactor Building wing basement to the Blower and Fan House, where it connects to the exhaust fans. The earth cover provides adequate shielding against radioactivity.

**8.2.6 Exhaust Fans and Auxiliaries.** *Exhaust Fans.* There are two full-capacity fans in parallel to provide the flow of gases through the contaminated-air system. One of these fans serves as a spare. Each fan is capable of delivering about 1000 cfm of air against a head of approximately 25 in. of water. The fans are of a standard centrifugal type and are motor driven. Motors of both fans are connected to both the normal commercial electrical supply and the site emergency power supply.

*Valves.* Butterfly valves are provided at both the inlet and outlet to each fan. The inlet valves are manually operated while the exhaust valves are air operated. The exhaust valves can be sealed airtight to permit repair or removal of the fan without leakage of radioactive air into the Blower and Fan House. It is not necessary to have the inlet valves seal airtight since any leakage is into the system.

**8.2.7 Instrumentation and Control.** The order of the following description will follow the flow diagram shown in Fig. 8.2.A, starting at the off-gas openings in the hoods.

There is a locally mounted pressure indicator in the off-gas line near each hood. This will inform the experimenter that there is or is not sufficient static head for him to use the off-gas opening.

At the scrubber, instrumentation is included to provide the following operational information:

1. Static pressure head at inlet of air line to the scrubber.
2. Pressure drop in the air passage through the scrubber.
3. Level of the caustic solution in the reservoir.
4. Temperature of the solution in the reservoir, which is desired during the neutralization of the depleted caustic with sulfuric acid.

Pressure indicators are provided at the caustic-circulating pumps. These indicators are locally mounted.

A pressure indicator is provided near the inlets to the exhaust fans. The indicator is mounted on the panel in the Blower and Fan House.

Radiation-measuring instruments are located in the duct leading from the exhaust fans. There is an ion chamber to determine the total activity of the gases from the scrubber and a particulate-activity meter to determine the radiation due to active particles in the gases. The information from both instruments is recorded. The recorders are located in the area of the Blower and Fan House that is designated solely for radiation instrumentation.

**8.2.8 Shielding.** It was difficult to estimate what the activity of the gases going through the contaminated-air system would be under actual operating conditions. The maximum radiation intensities of experiments conducted in the laboratory hoods and caves will be 1 and 10 curies, respectively. To arrive at some shielding requirements it was assumed that the activity is emitted at a rate of 0.01 curie per minute and that the gamma energy is 2 Mev.

The shielding requirements arrived at using standard shielding formulas and the above assumptions are summarized in Table 8.2.A.

Based on the shielding requirements listed in Table 8.2.A it was decided to provide no shielding of the contaminated-air system except the cells for the exhaust fans. The 8-in. pipe from the scrubber to the Blower and Fan House is shielded by an adequate earth cover, and the calculated radiation from the lines preceding the scrubber is small enough to make shielding of them initially unnecessary.

TABLE 8.2.A

**Shielding Requirements for the Contaminated-air System**

| ITEM | TOLERANCE (r/8 hr) | SHIELDING | |
|---|---|---|---|
| | | MATERIAL | THICKNESS (in.) |
| 2-in. pipe | 0.3 (at pipe) | Steel | 1/16 |
| 6-in. pipe | 0.3 (at pipe) | Steel | 1/4 |
| 8-in. pipe outside | 0.05 | Earth | 3 |
| Fan cells | | | |
| Outside walls | 0.005 | Concrete | 2.5 |
| | 0.005 | Steel | 1 |
| Motor room wall, east-north | 0.005 | Concrete | 2.5 |
| | 0.005 | Concrete | 6 |
| Intercell wall | 0.05 | Concrete | * |
| Roof | 0.1 | Concrete | * |

*Structural requirements only must be met.

## 8.3 BLOWER AND FAN HOUSE

**8.3.1 Introduction.** The Blower and Fan House contains the blowers and fans for the reactor-cooling- and contaminated-air systems along with their associated equipment. The equipment, function, and some phases of operation have already been discussed in Sections 8.1 and 8.2. The present section describes the general layout of the building equipment.

The general layout of the building is shown in Fig. 8.3.A. The building is a single-story structure and is located several hundred feet east-southeast of the Reactor Building.

**8.3.2 Air Ducts and Plenum Chamber.** The underground duct of the reactor cooling-air system enters the building at the north end of the west wall. This duct connects to a plenum chamber beneath the reactor air system blower

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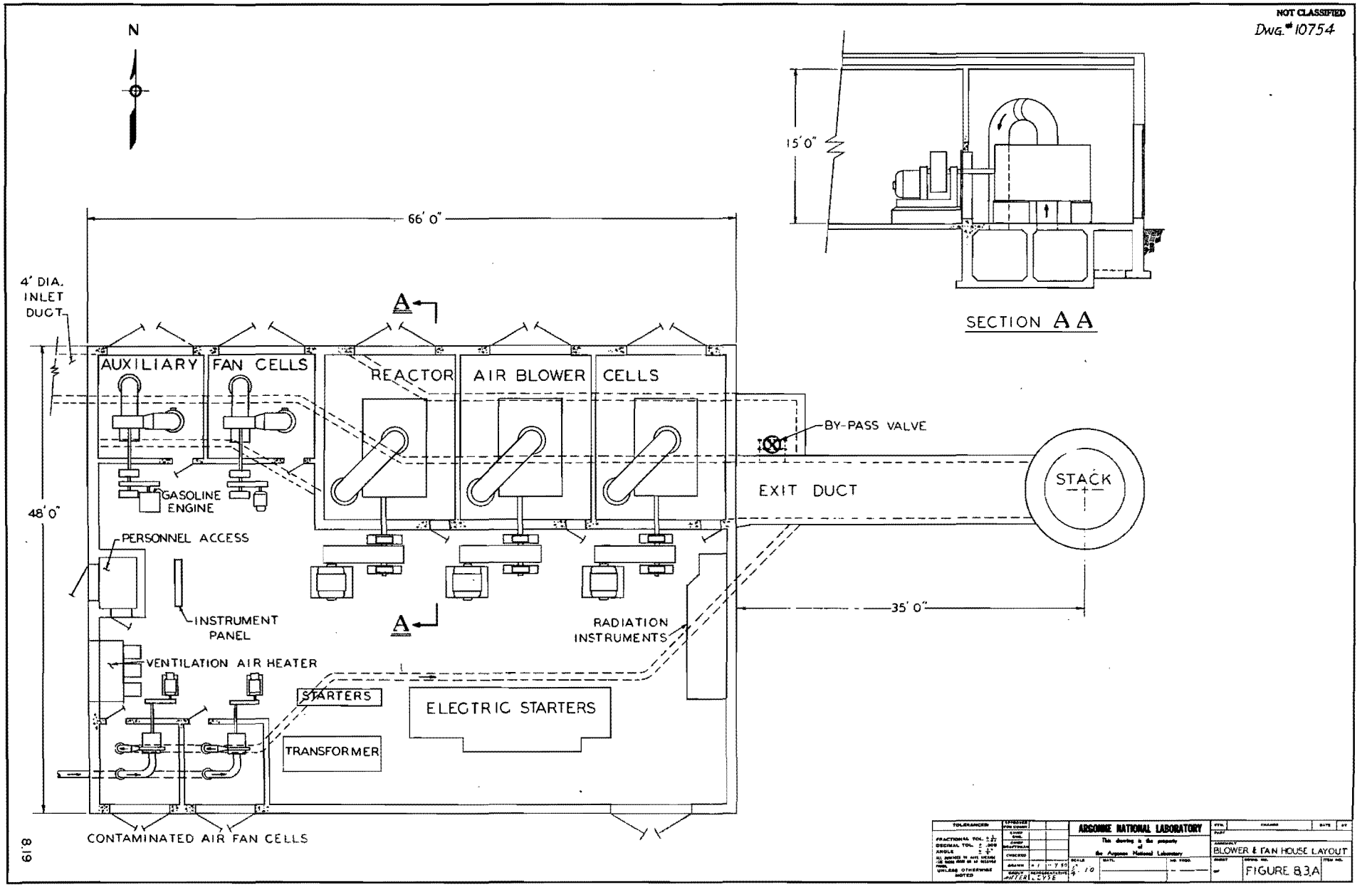


and fan cells. The plenum chamber extends slightly beyond the full length of and building. Starting at the west wall of the building, the exhaust plenum chamber also lies beneath the blower and fan cells and roughly parallels the inlet plenum chamber. The main exit duct to the stack is essentially an extension of the exhaust plenum chamber. The top of the duct is approximately at grade level. The plenum chambers and duct just described are constructed of poured concrete. Manways to the plenum chambers are provided. The by-pass valve linking the inlet plenum chamber to the exit duct is located at the east end of the inlet plenum chamber.

The duct of the contaminated-air system enters the building below grade at the south end of the east wall. This duct leads to the exhaust fans, and an exhaust duct leads from the fans, beneath the building floor, to connect with the main exit duct near the east wall of the building.

**8.3.3 Blower and Fan Cells.** The blower and fan cells for the reactor cooling-air system are located along the north wall of the building while those for the contaminated-air system are at the southwest corner. The inlets and exhausts to all blowers and fans are through the floors of the cells. The walls of all cells are of poured concrete construction. Poured concrete is used instead of masonry because it provides a more uniform and dense shield. The walls of all cells which form a part of the building external wall and the external walls of the reactor main blower cells are 1 ft thick. The remaining cell walls are 6 in. thick. These thicknesses are more than adequate for shielding requirements.

Each cell has a door which opens to the interior of the building. These doors are for convenience during periodic inspections and minor servicing. In an outside wall of each cell is a large double door. These doors provide access to the cells for installing, replacing, or making major repairs of the equipment. All doors are of steel and provide adequate shielding.

There is a plugged drain opening in each cell. The drains are provided for use during washdowns of any contaminated equipment within the cells. These drains, along with the one in the main exit duct, drain to a sump beneath the building floor. The waste is pumped from the sump to the retention basin.

**8.3.4 Miscellaneous.** Besides the blowers, fans, and associated equipment, the Blower and Fan House contains the recording instruments for the

radiation measurements of the activity of both reactor and contaminated-air systems. The recorders are situated in an area at the east wall of the building where they are convenient to the exit ducts from both air systems.

## 8.4 STACK

**8.4.1 Introduction.** The air from the reactor and contaminated-air systems contains more radioactivity than can be safely discharged to the atmosphere at ground level in an inhabited area. Thus it is necessary to eject this active air through a stack of considerable height. This results in the diffusion of the activity over a large area, thereby greatly reducing the hazard presented by the active gases.

**8.4.2 Activity of Stack Gases.** The activity in the stack gases due to air from the contaminated-air system will generally be negligible compared to that due to the air from the reactor. Thus the design of the stack is based principally on the radioactivity of the reactor-cooling air.

The activity of the reactor-cooling air is due to radioactive particles and gases. For obvious reasons, the activity due to particles does not lend itself to calculation. Therefore the activity of the stack gases is considered to be that due solely to the 110-min half-life $A^{41}$ (argon) formed in the reactor.

The rate of activation of the argon in the reactor-cooling air is given by

$$\text{Activation rate} = \Phi N \sigma_a V$$

where

$\Phi$ = average flux over volume $V$, neutrons/cm$^2$-sec

$N$ = number of $A^{40}$ atoms per ft$^3$

$\sigma_a$ = activation cross-section of $A^{40}$ = $0.62 \times 10^{-24}$ cm$^2$

$V$ = volume of region, ft$^3$

The activation rates for the various regions are summarized in Table 8.4.A. The term "air in graphite" refers to the air spaces naturally occurring in graphite.

**TABLE 8.4.A**

**Production of $A^{41}$ in the MTR**

| REGION | VOLUME ($ft^3$) | FLUX, $\Phi$ (neutrons/$cm^2$-sec) | ACTIVATION RATE (atoms/sec) |
|---|---|---|---|
| Graphite ball zone | | | |
| Voids | 130 | $6.75 \times 10^{12}$ | $3.1 \times 10^{12}$ |
| Air in graphite | 52 | $6.75 \times 10^{12}$ | $1.24 \times 10^{12}$ |
| Permanent graphite zone | | | |
| Cooling holes | 8.6 | $2.1 \times 10^{12}$ | $0.06 \times 10^{12}$ |
| Beam hole annuli | 2 | $1.4 \times 10^{12}$ | $0.01 \times 10^{12}$ |
| Air in graphite | 275 | $1.4 \times 10^{12}$ | $1.36 \times 10^{12}$ |
| Plenum chambers | | | |
| Top | 201 | $1.75 \times 10^{11}$ | $0.12 \times 10^{12}$ |
| Bottom | 175 | $3.5 \times 10^{11}$ | $0.22 \times 10^{12}$ |
| Total | | | $6.11 \times 10^{12}$ |

The $6.11 \times 10^{12}$ atoms/sec activation gives a total activity of about 1500 curies/day for 30,000-kw operation with a thermal flux of $2.5 \times 10^{14}$ neutrons/$cm^2$-sec at the center of the reactor. This amounts to a concentration of about $2.5 \times 10^{-3}$ microcurie/cc at the stack exit. This is 2500 times the tolerance value of $10^{-6}$ microcurie/cc for $A^{41}$ in the atmosphere.

**8.4.3 Stack Height.** The stack height depends on the amount of dilution necessary to reduce the concentration of the radioactivity to below tolerance before it reaches ground level. This involves consideration of the meteorological conditions encountered at the MTR site. The stack is located near the southeast corner of the MTR site. Since the winds at the site are predominantly southwest or northeast, the exhaust gases from the stack generally pass over the relatively uninhabited southeast corner of the site. Thus, from the standpoint of wind direction the stack is well situated.

Calculations of stack height necessary were made using O. G. Sutton's diffusion equation and also various other geometrical interpretations. Using the results from these calculations and experimental information from published literature, the stack height for the MTR was established at 250 ft. This

height is believed adequate to ensure that exposure to the radioactive stack gases will not exceed tolerance anywhere on the site. Also, it is believed sufficient to prevent a dangerous increase of background activity due to the deposit of active particles on or off the site.

In order to give some indication of the activity at ground level due to the stack gases, the results of several calculations using Sutton's equation are included. Sutton's equation yields a relation which says that the distance from the stack at which the maximum activity at ground level occurs is independent of wind velocity. For a 250-ft-high stack this distance is about 3600 ft; this is well outside the MTR site which lies within a distance of about 1600 ft from the stack. The maximum concentrations at ground level for the 250-ft stack are calculated to be $1.56 \times 10^{-12}$, $0.31 \times 10^{-12}$, and $0.156 \times 10^{-12}$ curie/cc for wind velocities of 1, 5, and 10 mph, respectively. Also based on Sutton's equation, the wind velocity necessary to reduce the argon activity to tolerance ($10^{-12}$ curie/cc) at ground level on the site is about 0.15 mph. A meteorology report[21] on the Reactor Testing Station Area states that winds of 0 to 3 mph occur 15 to 20% of the time. Thus, based on Sutton's equations, the activity at ground level should very seldom, if ever, exceed tolerance.

**8.4.4 Stack Diameter.** The exit diameter of the stack is 5 ft. This is based on the exhaust volume of the gases and the desired exit velocity.

The volume of air exhausted from the stack at present is about 41,000 cfm, of which a minor portion comes from the contaminated-air system. The diameter of the stack is based on a maximum flow of 80,000 cfm so as to provide for future additional requirements. It was desired to have the exit velocity as high as practical so as to minimize "downwash," a term referring to gases drawn downward in the vortices formed in the lee of a stack. With regard to stack noise and friction losses, a maximum exit velocity of 4,000 fpm can easily be tolerated. The exit velocity for the present flow is then about 2000 fpm. This velocity is sufficient to prevent "downwash" from conveying the highly active stack gases to the turbulent air around the buildings, a phenomenon which would give intolerable concentrations of radioactivity near ground level. The stack diameter of 5 ft provides the 2000-fpm exit velocity.

**8.4.5 General Construction.** The stack is constructed of reinforced concrete. There is a ladder extending the entire height of the stack. Lightning arresters are provided at the top of the stack. Aircraft obstruction warning lights are not provided.

## 8.5 REFERENCE REPORTS


(1) Gall, W. R., *MTR Report of Internal Air System*, ORNL CF-49-11-208, November 21, 1949.

(2) Gall, W. R., *MTR Internal Air System Design Report - Addendum*, ORNL CF-50-2-61, February 13, 1950.

(3) Leyse, C. F., *Argon Activity from the Materials Testing Reactor*, ANL-EKF-19, November 17, 1949.

(4) Lane, J. A., *Argon Activity in Reactor Cooling Air*, ORNL CF-49-11-180, November 17, 1949.

(5) Lane, J. A., *Air Cooling System*, ORNL CF-49-11-219, November 23, 1949.

(6) Leyse, C. F., *Maximum Temperature Rises in Graphite During Reduced Air Flow*, ANL-EKF-28, January 6, 1950.

(7) Leyse, C. F., *Cooling Requirements of Graphite After Shutdown*, ANL-EKF-29, January 6, 1950.

(8) Leyse, C. F., *MTR Shutdown Air Requirements*, ANL-EKF-30, January 10, 1950.

(9) Leyse, C. F., *Preliminary Investigations of Stack Height*, ANL-EKF-31, January 17, 1950.

(10) Leyse, C. F., *Shielding of the External Reactor Air System*, ANL-EKF-32, January 26, 1950.

(11) Leyse, C. F., *Shielding of the Contaminated Air System*, ANL-EKF-35, February 7, 1950.

(12) Lane, J. A., *MTR Shutdown Cooling Requirements*, ORNL CF-49-12-84, December 16, 1949.

(13) Leyse, C. F., *Reactor Air System Pressure Drops*, ANL-EKF-36, February 8, 1950.

(14) Leyse, C. F., *Pressure Drop Through the Contaminated Air System*, ANL-EKF-37, February 9, 1950.

(15) Leyse, C. F., *Design Report No. 32, MTR External Air System*, ANL-EKF-40, February 14, 1950.

(16) Leyse, C. F., *Stack Design for the MTR*, ANL-EKF-45, March 30, 1950.

(17) Leyse, C. F., *Omission of MTR Cooling Air System Discharge Filters*, ANL-EKF-47, March 31, 1950.

(18) Briggs, R. B., *Cooling of the Graphite Reflector for the MTR*, ORNL CF-50-3-102, March 20, 1950.

(19) Leyse, C. F., *Determination of Stack Dimensions*, ANL-EKF-49, April 6, 1950.

(20) Leyse, C. F., *Addendum No. 1, Design Report No. 32, MTR External Air System*, ANL-EKF-52, April 25, 1950.

(21) U. S. Weather Bureau, Meteorology of the Reactor Test Site, Arco, Idaho, September 16, 1949.

## 8.6 REFERENCE DRAWINGS

(Blaw-Knox Construction Company Drawings)

| DWG. NO. | TITLE |
|---|---|
| 3150-92-10 | Flow Diagram - Reactor Air System |
| 3150-92-13 | Flow Diagram - Air Exhaust System |
| 3150-10-1 | Fan House and Stack - Layout |

# Chapter 9 *

# SITE AND BUILDINGS

## 9.1 SITE PLAN

An artist's sketch of the MTR site is shown in Fig. 9.1.A. The Materials Testing Reactor site plan is shown in Fig. 9.1.B. The layout shown by these figures is based on consideration of the following major factors:

1. The building or facility function and its relation to the reactor and other facilities.

2. Accessibility to and from the site and facilities from the viewpoint not only of function but also of security.

3. Contamination and pollution as they affect other facilities.
   a. Wind direction.
   b. Deep water direction.
   c. Subterranean drainage.
   d. Surface drainage.

4. Power, railroad, and highway connections.

5. Topography of the MTR plot.
   a. Flat terrain for the quarter-mile beam.
   b. Natural drainage for the retention basin.
   c. Overburden.

6. Security.
   a. Fire fighting.
   b. Patrolling.

7. Future expansion of initial facilities.

8. Future buildings and facilities.

9. Most feasible and shortest piping, air ducts, wiring, fencing, roads, and walks.

The general layout of the site is centered around the Reactor Building in an area where the terrain is reasonably level. The level terrain is necessary

*ANL contribution.

DWG. 10768 [redacted]

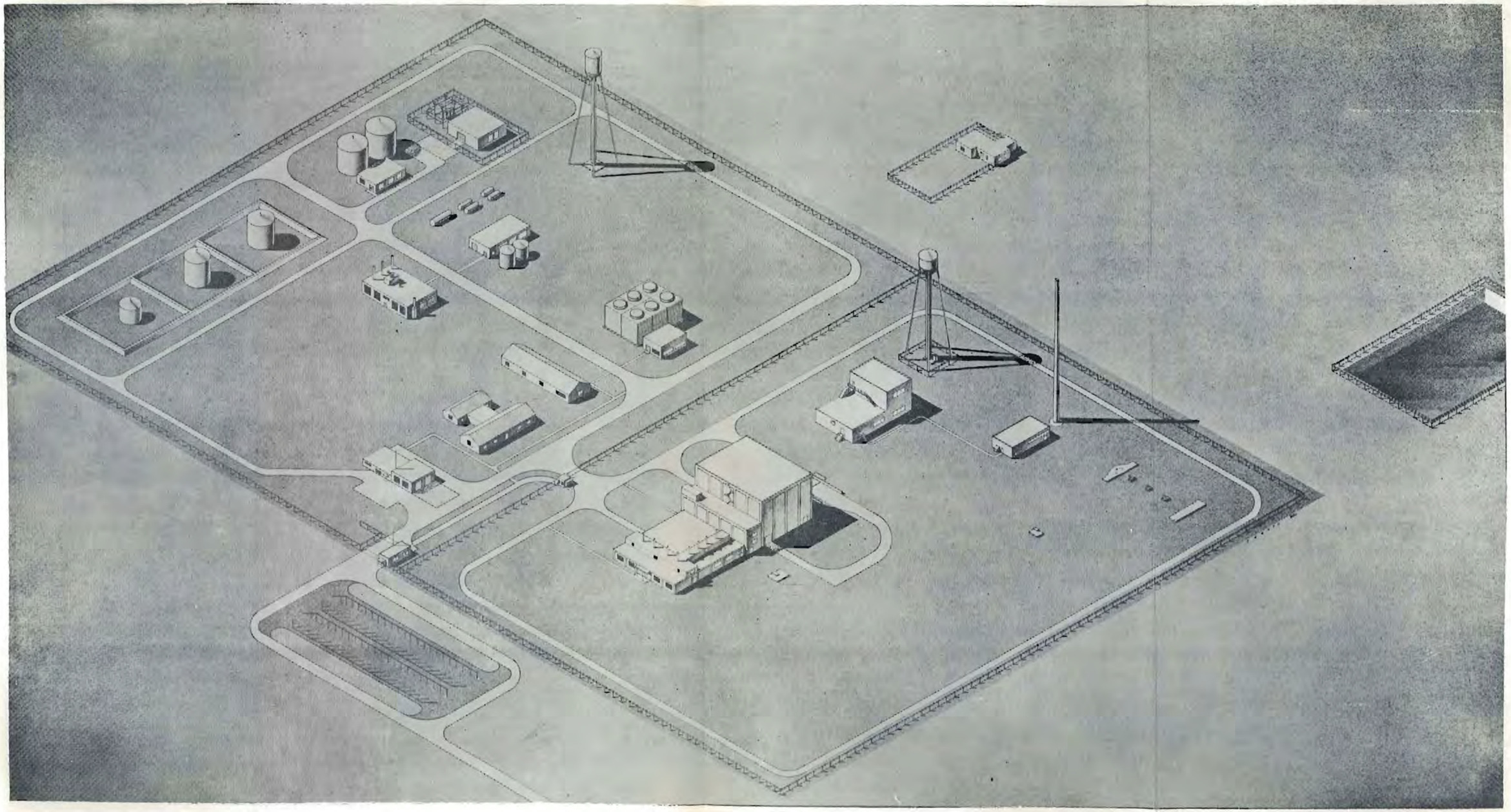
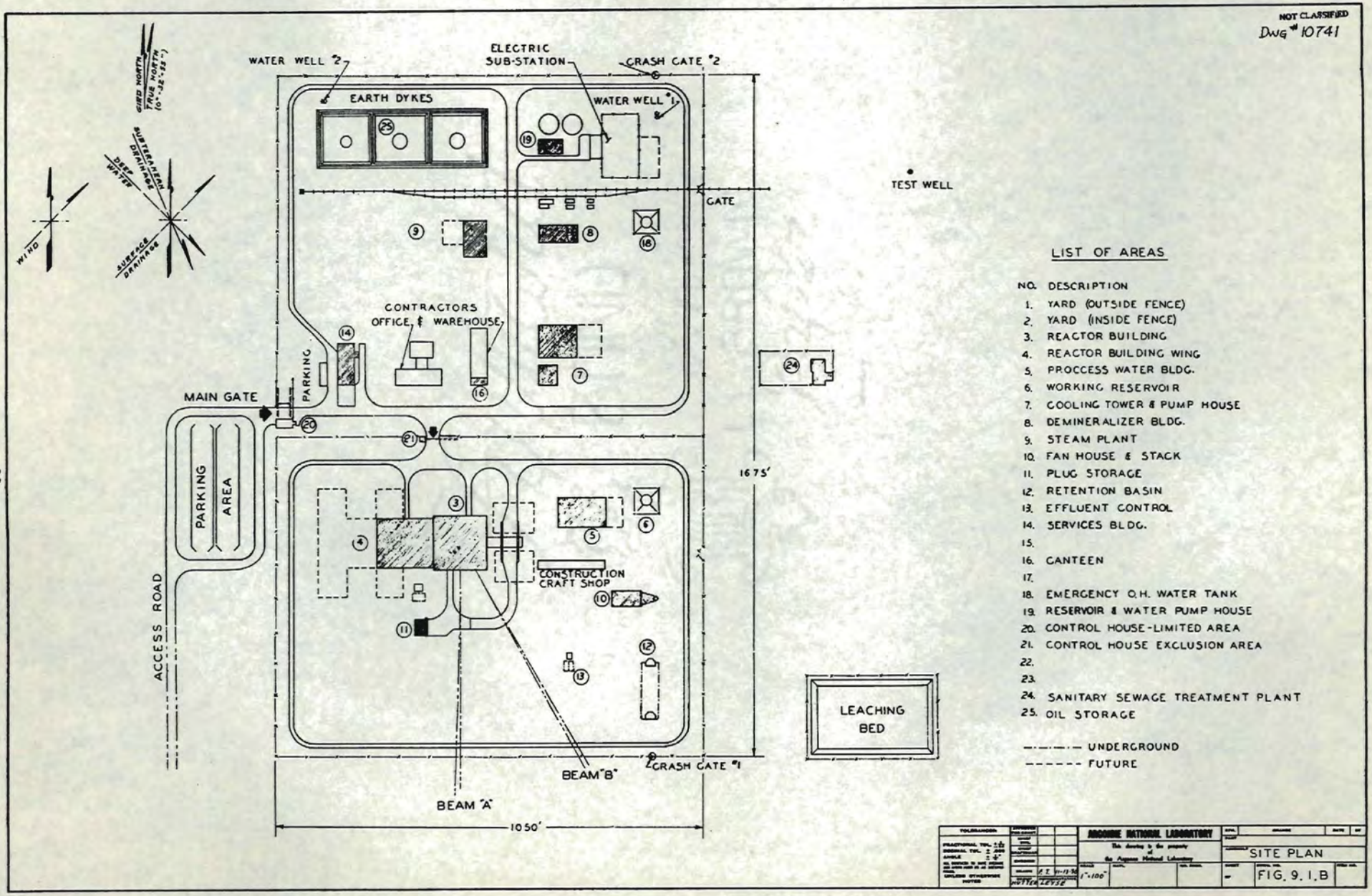


FIGURE 9.1A MTR SITE-ARTIST'S SKETCH

NOT CLASSIFIED

Dwg # 10741



since beam experiments may be conducted at distances up to one-quarter of a mile from the reactor. The location has a minimum of 40 ft of overburden above the underlying rock strata, thus eliminating costly and time-consuming rock blasting for the extensive foundations of this building.

The site area is divided into two general areas. A "limited" or "clean" area containing most of the nonradioactive operations and service functions is separated by a fence from an "exclusion" or "hot" area containing the reactor and its closely related radioactive auxiliaries. The exclusion area is approximately the south half of the plot. Since the subterranean and deep water flows are to the southeast, this location decreases the possibility that escaping radioactive water may flow toward the wells located in the north portion of the limited area. The most prominent wind direction at the site is from the southwest, and the direction of secondary frequency is from the northeast. Therefore the stack for the exhaust of reactor-cooling air is near the southeast corner of the site.

The main entrance to the site is at the west side of the limited area. This location permits personnel to travel to and from work without going through the area into which the exhaust reactor-cooling air is generally swept. With the main entrance on the limited area side, the number of persons entering the exclusion area can be kept at a minimum. The laboratory facilities are housed in the Reactor Building wing adjacent to the west side of the Reactor Building. This presents the closest approach of these two buildings to the site main entrance gate. Personnel for both buildings enter through a single entrance located on the west side of the laboratory wing, thereby keeping the Reactor Building free of persons not having specific duties there.

The Process Water Building and working reservoir are located east and slightly north of the Reactor Building. This location effects a reasonably close approach both to the reactor and the water supplies located in the limited area. Effluent control, the retention basin, and the effluent pond are in the southeast corner of the site because of the direction of the undersurface water flows.

The arrangement of facilities in the limited area has the Services Building near the main site entrance and the canteen located centrally. The various water-storage, pump, and treatment buildings lie north of the Process

Water Building and are arranged approximately in the sequence of flow starting from the wells near the north fence and going south to the various points of usage.

In accordance with AEC security regulations all facilities are located a minimum of 150 ft inside the perimeter fence. The perimeter fence is made of 8-ft-high chain-link fencing topped with three strands of barbed wire angling outward. The limited and exclusion areas are separated by similar fencing. A truck trap is provided at the main entrance to allow thorough guard inspection of vehicles entering or leaving the site. Limited- and exclusion-area control houses are provided at the entrances to each area. These serve as guard stations for badge exchange and personnel screening. A patrol road encircles each subarea and is located within approximately 25 ft of the site perimeter fence. A stationary lighting system provides continuous lighting during hours of darkness. Pendant luminaries with refractory lenses designed so as to direct most of the light in the direction needed are used. A portion of this lighting circuit is on the emergency power system.

Largely for purposes of fire protection all buildings have a minimum of 50 ft separation from adjacent buildings.

A parking lot for vehicles not used on the site is provided outside the west perimeter fence near the main entrance.

Sufficient space remains for future additions to the Reactor Building wing , Process Water Building, Cooling Tower. Water Treatment Building, and Boiler House. Also, space is provided for future facilities such as an administration building, reactor service building, cafeteria, and hot laboratory.

The plot plan originally conceived for the Idaho area, was considerably larger than the one shown in Fig. 9.1.A. The north-south dimension was slightly longer and the east-west dimension was approximately 900 ft longer. When economy forced a cut-back of the facilities planned for the reactor, the site was decreased in size. The sewage-treatment plant remains in its original location, which is now outside the perimeter fence. As this plant is a potential source of odors, it is isolated as much as is practical.

## 9.2 REACTOR BUILDING

An artist's sketch of the Reactor Building and wing is shown in Fig. 9.2.A. This figure shows the architectural treatment of the two adjoining buildings.

DWG. 10769 [redacted]

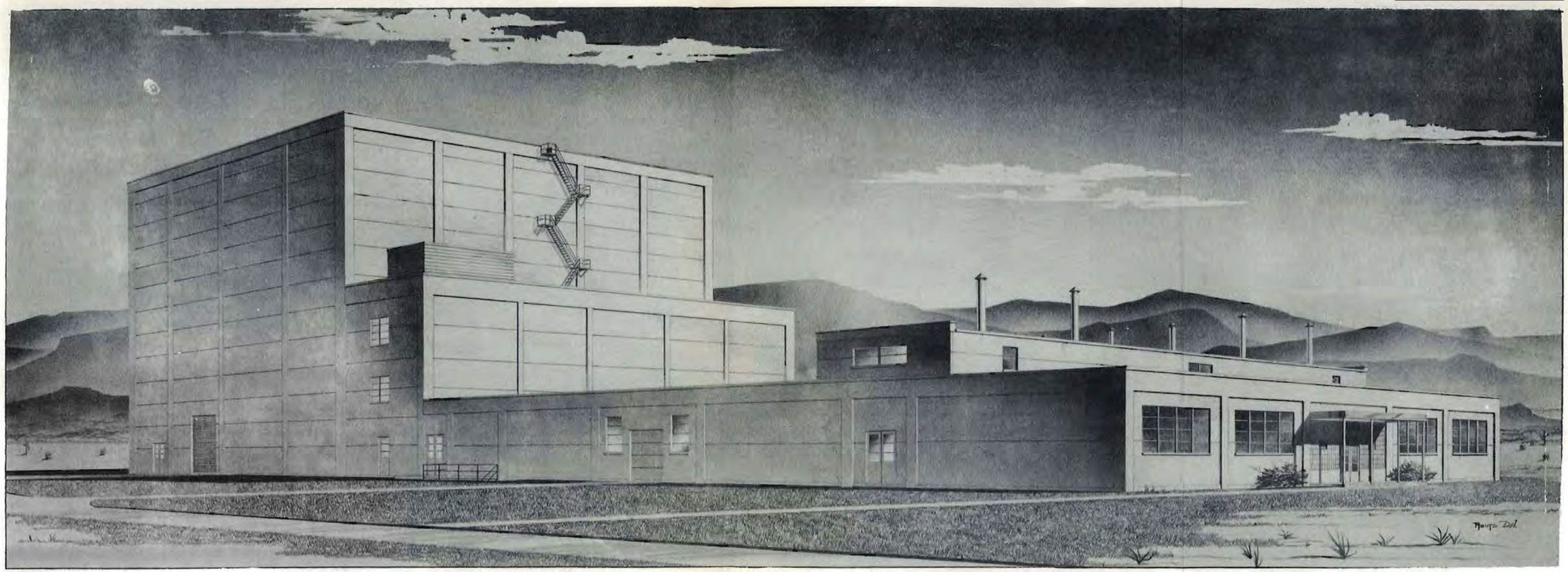


FIGURE 9.2A REACTOR BUILDING AND WING ARTIST'S SKETCH

The wing is in the foreground, and on its west side is the main entrance to the two buildings.

**9.2.1 General Building Details.** The height of the main portion of the building is based on the requirements of the main building crane. The main hoist of the crane has a capacity of 30 tons. The 30-ton hook is 30 ft above the reactor top when at its highest position. The upper limit of travel is governed entirely by the height necessary to permit hoisting the top plug over the railing around the reactor top. The crane capacity is such that it can handle the 22-ton coffins for dummy plugs and also heavier coffins for probable experimental plugs. A 5-ton auxiliary hoist is available on the main crane, since this load capacity is sufficient for the major portion of crane work. Since the auxiliary hoist has a higher lifting speed, jobs under 5 tons can be dispatched in less time than if the relatively slow 30-ton hoist were the only one available. Each hoist on the main crane has sufficient cable to provide a 75-ft lift, which length is sufficient for operations involving the placing of objects into or taking them out of the canal located below basement floor level. The crane span of 100 ft gives coverage between the control room balconies and the east outside wall. The bridge can travel along the full north-south building dimension. This wide working latitude of the crane permits servicing of the entire operating area of the first floor.

The ventilation requirements of the building are consistent with standard practice. Six air changes per hour prevail in all areas except the reactor room, the instrument room, and control room. The six air changes exist in such areas as the sub-pile room, the vault, and the canal tunnel. As the reactor room is 75 ft high, only two air changes per hour are provided in this area.

The entire Reactor Building is pressurized to a minimum of 1/8 in. water. This promotes air leakage out of rather than into the building and thereby tends to prevent outside dust particles from contaminating the filtered building air. This is important since the reactor draws its cooling air supply from within the building.

The instrument and the control rooms are air-conditioned and humidity controlled. The actual quantity of air used in these two rooms is governed by the heat load involved. Seventy-five percent of the air is recirculated.

These two rooms have a slightly higher static pressure than the remainder of the building.

**9.2.2 First Floor.** The building, the layout of which is illustrated in Figs. 9.2.B, C, and D, is planned around the first floor area. Figure 9.2.B shows the first floor plan. Survey work among the scientific personnel interested in conducting experiments on a high-flux reactor such as the MTR revealed that some experiments required apparatus reaching out 25 ft from a reactor face. History of experimental work indicates that this distance has grown rather steadily in the past few years. Assuming that the high flux available in the MTR would speed up this trend of elongated apparatus, an additional 15 ft is available. An 8-ft traffic aisle remains when experiments require all of the 40 ft allowed. To provide this space for experimental apparatus the first floor is approximately 130 ft square.

The first floor area is kept as free as possible of permanent equipment so that few areas are restricted from experimental apparatus. The instrument and the control rooms are on the second and third floor levels, primarily to avoid first floor congestion.

The features of the first floor and the reasons for their existence are as follows:

*Stairways.* Stairways in the northeast and southeast corners are for traffic between the first floor and basement; stairways in the northwest and southwest corners are for traffic between all floor levels. This number of stairways is necessary not only for convenience, but also for safety considerations. The stairways are encased to provide fire wall protection. A one-way escape door is located on the first floor level of each stairwell. The southwest stairwell contains the main entrance door into the Reactor Building. A similar setup is provided in the northwest stairwell; however, this door should be normally locked so that only one screening point is necessary for personnel entering the Reactor Building.

*Truck doors.* Two truck doors are provided, one in the south wall west of the north-south building centerline and one in the north wall east of the same centerline. These two are necessary for ease of vehicle movement, particularly in the event that the quarter-mile beam experiment is installed. The doors are made sufficiently large to allow a highway-size tractor-trailer unit

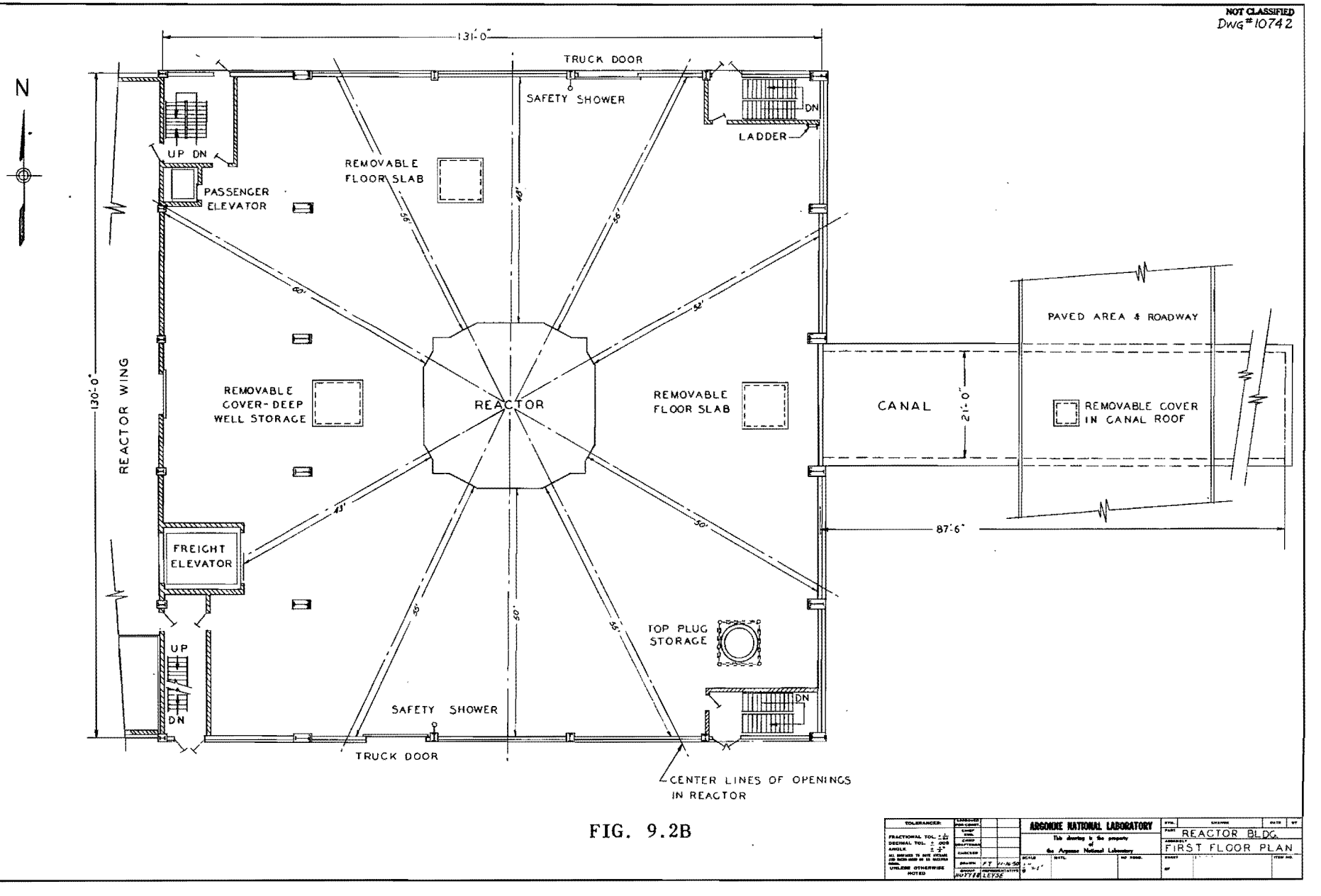
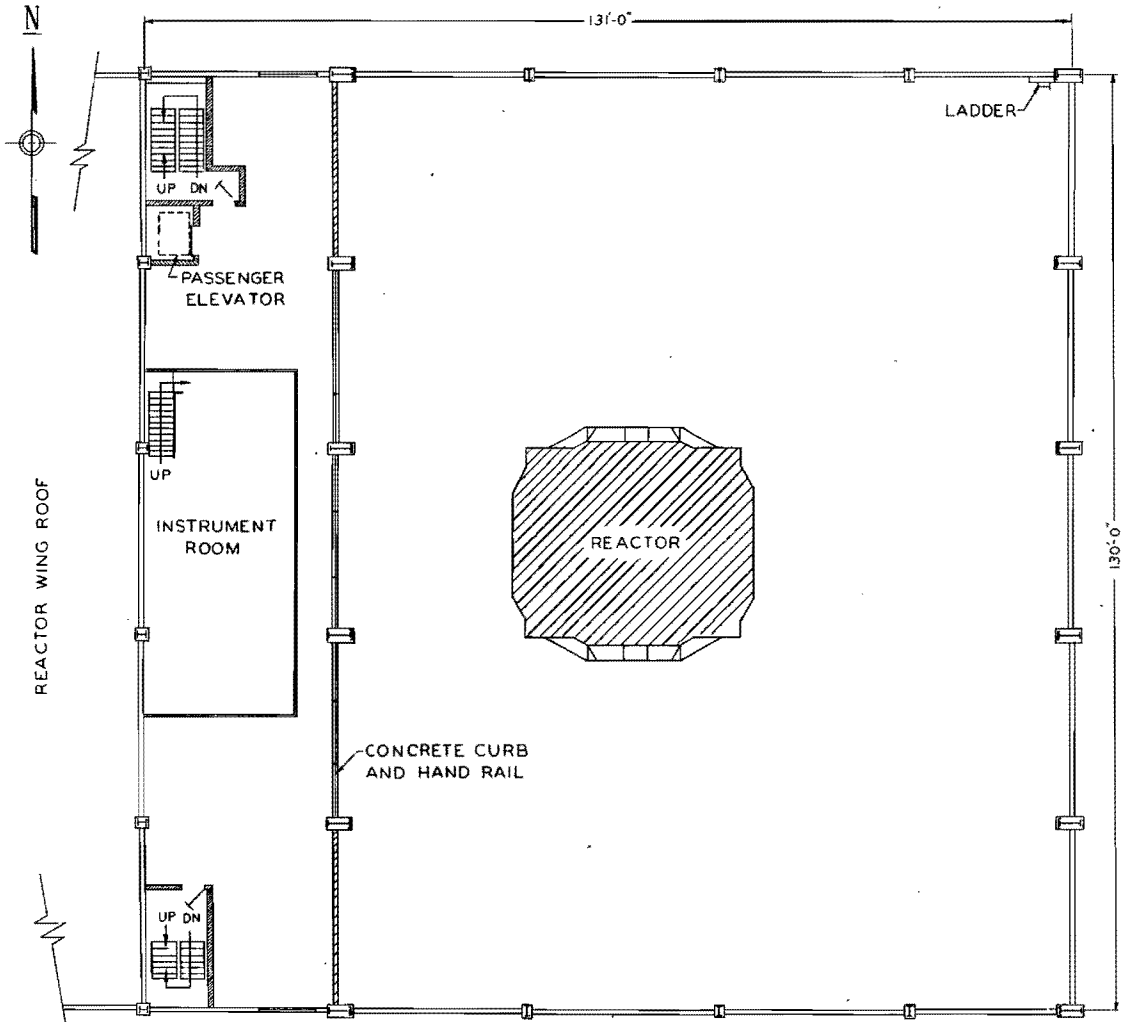

NOT CLASSIFIED
DWG # 10742
N
131'-0"
TRUCK DOOR
SAFETY SHOWER
DN
LADDER
UP DN
PASSENGER ELEVATOR
REMOVABLE FLOOR SLAB
130'-0"
REACTOR WING
REMOVABLE COVER- DEEP WELL STORAGE
REACTOR
REMOVABLE FLOOR SLAB
PAVED AREA & ROADWAY
CANAL
21'-0"
REMOVABLE COVER IN CANAL ROOF
87'-6"
FREIGHT ELEVATOR
UP
DN
TOP PLUG STORAGE
SAFETY SHOWER
DN
TRUCK DOOR
CENTER LINES OF OPENINGS IN REACTOR
ARGONNE NATIONAL LABORATORY
This drawing is the property of the Argonne National Laboratory
REACTOR BLDG.
FIRST FLOOR PLAN

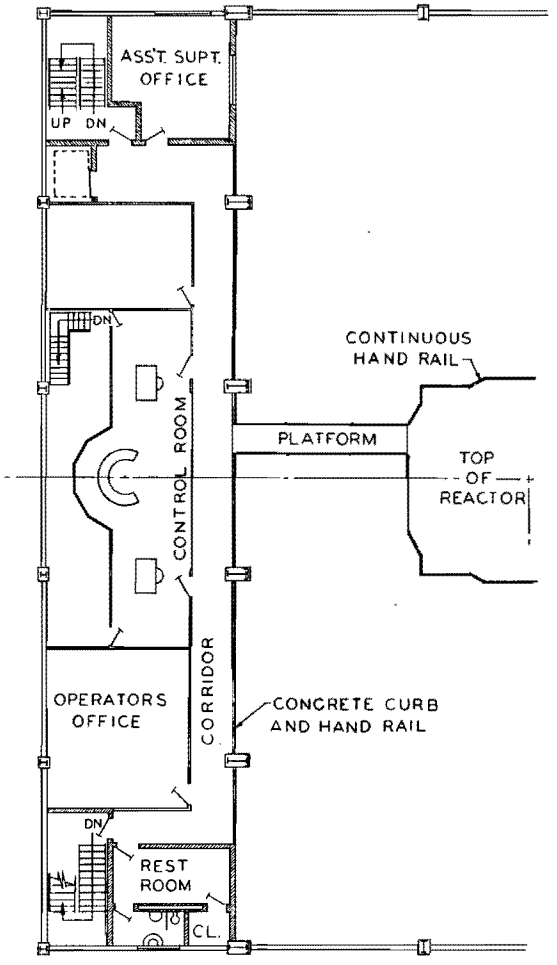
FIG. 9.2B

NOT CLASSIFIED

Dwg # 10743



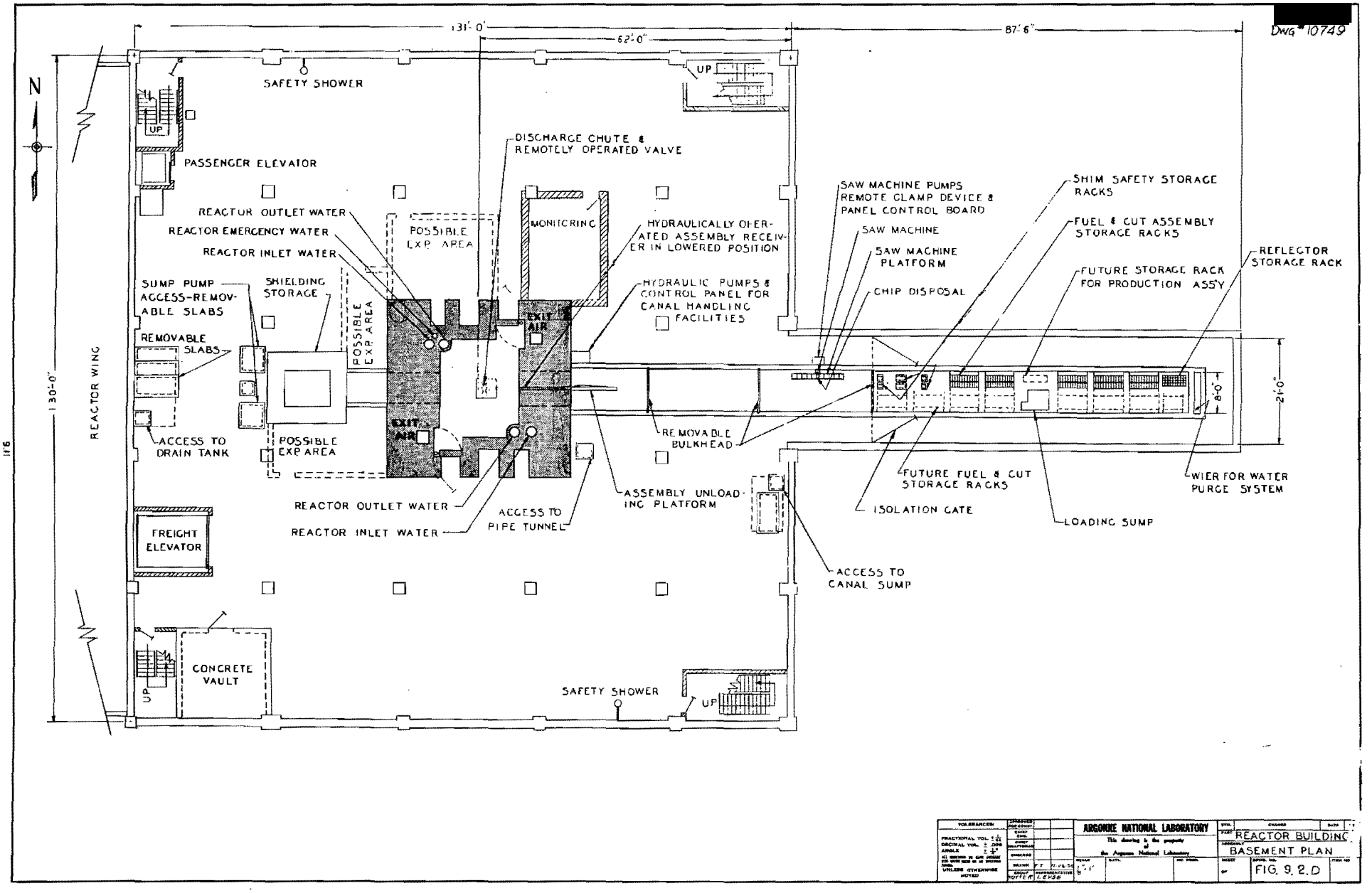

SECOND FLOOR PLAN




PARTIAL THIRD FLOOR PLAN

| | | | |
|---|---|---|---|
| TOLERANCES | ARGONNE NATIONAL LABORATORY | PART: REACTOR BUILDING | |
| FRACTIONAL TOL. ± 1/32 | This drawing is the property of the Argonne National Laboratory | ASSEMBLY: SECOND & THIRD FLOORS | |
| DECIMAL TOL. ± .005 | | | |
| ANGLE ± 1/2° | | | |
| ALL DIMENSIONS TO BE ACCURE. FOR MORE DIGS IN AS REQUIRED | | | |
| UNLESS OTHERWISE NOTED | | SHEET | DRWG. NO. FIG. 9.2.C |

9.10


DWG # 10749
131'-0"
62'-0"
87'-6"
N
SAFETY SHOWER
UP
PASSENGER ELEVATOR
DISCHARGE CHUTE & REMOTELY OPERATED VALVE
REACTOR OUTLET WATER
REACTOR EMERGENCY WATER
REACTOR INLET WATER
POSSIBLE EXP. AREA
MONITORING
HYDRAULICALLY OPERATED ASSEMBLY RECEIVER IN LOWERED POSITION
SAW MACHINE PUMPS REMOTE CLAMP DEVICE & PANEL CONTROL BOARD
SAW MACHINE
SAW MACHINE PLATFORM
CHIP DISPOSAL
SHIM SAFETY STORAGE RACKS
FUEL & CUT ASSEMBLY STORAGE RACKS
FUTURE STORAGE RACK FOR PRODUCTION ASS'Y
REFLECTOR STORAGE RACK
SUMP PUMP ACCESS-REMOVABLE SLABS
SHIELDING STORAGE
REMOVABLE SLABS
HYDRAULIC PUMPS & CONTROL PANEL FOR CANAL HANDLING FACILITIES
EXIT AIR
REACTOR WING
130'-0"
ACCESS TO DRAIN TANK
POSSIBLE EXP AREA
REMOVABLE BULKHEAD
8'-0"
21'-0"
FUTURE FUEL & CUT STORAGE RACKS
WIER FOR WATER PURGE SYSTEM
ASSEMBLY UNLOADING PLATFORM
ISOLATION GATE
LOADING SUMP
REACTOR OUTLET WATER
REACTOR INLET WATER
ACCESS TO PIPE TUNNEL
FREIGHT ELEVATOR
ACCESS TO CANAL SUMP
CONCRETE VAULT
SAFETY SHOWER
UP
ARGONNE NATIONAL LABORATORY
This drawing is the property of the Argonne National Laboratory
REACTOR BUILDING
BASEMENT PLAN
FIG. 9.2.D
TOLERANCES UNLESS OTHERWISE NOTED
DRAWN
DRAFT. REPRESENTATIVE
SCALE 1/8" = 1'-0"

to be brought into the building. To cut down the amount of dirt carried into the building, cold drains are available directly inside the truck doors to allow for hose cleaning of vehicle wheels.

*Elevators.* For convenience of operators and visitors, a passenger elevator operates between the basement and control room levels with outlets on all floors. A 10-ton hydraulically operated freight elevator operates between the first floor and basement, and has an intermediate stop at the Reactor Building wing basement level. This freight elevator provides for the normal movement of supplies and equipment between floors. The elevator has a 10- by 14-ft platform; it is capable of handling a 7000-lb-capacity electric fork-lift truck with a fully loaded skid on the forks.

*Removable floor slabs.* The first floor has two 8- by 8-ft removable floor slabs, one located directly over the canal, and the other over a clear area of the basement. The first permits heavy coffins to be placed into or taken from the canal, while the second allows heavily shielded apparatus to be placed onto or taken from the basement floor.

*Top plug storage.* To accommodate storage of the top plug during the reactor loading and unloading operations, a 6-ft-diameter hole is located near the southeast corner of the building. The hole is placed so that adequate working space is available on all sides of the plug. The weight of the top plug is carried on the first floor slab by the plug cover flange. In this position the magnet end of the plug is at a convenient working height above the basement floor.

*First floor thickness.* As protection for personnel working in the basement when "hot" experiments are being conducted on the first floor and to protect personnel on the first floor when the reverse situation is true, the floor is 3 ft thick in a zone of approximately 25 ft around the reactor. The thickness is an arbitrary one which gives good shielding and yet is within practical construction limits. The 25-ft distance is the most likely zone for "hot" work.

*Floor loading.* The first floor loading is the result of studies of probable loads resulting from lead shielding walls and plug coffins. The floor designed for these concentrated loads allows 900 lb per square foot of uniform load. Concentrated loads in excess of 4000 lb per square foot can be

sustained.

*First floor services.* The most frequently used services are steam, compressed air, filtered water, and 110-volt single-phase and 220-volt three-phase current. Outlets for these services are provided at numerous spots on the reactor faces and at two locations on each of the north, east, and south walls of the building. Sufficient space is available at these outlet locations for future additions.

*Facility holes.* Since it is impractical to install all services which are desired by the experimenter, the so-called "facility holes" are placed in the first floor at likely points of use. By means of these holes, special services can be added by running leads from the source of supply along the basement ceiling and through the hole nearest the user. This will prevent cluttering the first floor and the reactor faces with future piping.

**9.2.3 Second and Third Floors.** The layout of the second and third floors is shown in Fig. 9.2.C, as mentioned in the discussion of the first floor; the instrument and control rooms are on the second and third floor levels. These two rooms operate much as a unit with the instrument room having the amplifiers and relays and the control room having the indicating and recording instruments for reactor operation. The control room is at the higher level so that conduits and raceways do not come from an overhead position.

The third floor level is at the same elevation as the reactor top. This gives a level 4-ft walkway between these two areas. This walkway provides the main access path to the mechanisms located on the reactor top.

The second and third floor levels extend completely across the building even though the two primary rooms require somewhat less than half the total area available. A rest room, toilet, chart storage room, stairwells, elevator shaft, reactor operator's office, and probable office space for reactor operating or technical personnel occupy the remainder of the space.

Standard metal partitioning is used for the wall construction around the instrument and control rooms, adding flexibility to the design. These rooms can be made larger or smaller or additional rooms can be added without upsetting reactor operations. The wall construction used is completely salvable.

The instrument room is restricted to all but instrument maintenance and repair personnel. For the convenience of these technicians, a special stairway between the instrument and control rooms is provided.

Other features of these floor levels are the following:

*Balconies.* There are corridors on the reactor side of this section of the building. A railing along the reactor side of these corridors allows them to be used as observation balconies.

*Toilet and rest room.* This facility is located on the third floor for the convenience of the reactor operators.

*Reactor operator's office.* This office is for the reactor shift supervisors and the operators themselves. It is placed adjacent to the control room for their convenience.

*Assistant Superintendent's office.* The Assistant Superintendent is defined on the tentative organization chart as being the man directly in charge of reactor operations. For this reason his office is on the third floor at the end nearest the passenger elevator. The office has a window overlooking the reactor and the first floor.

*Additional floor area.* The remaining area on these floors is not subdivided. The operating contractor can subdivide to suit his organizational setup.

**9.2.4 Basement.** The Reactor Building basement has the same outside dimensions as the first floor. The floor elevation corresponds to that of the sub-pile room. The general layout is shown in Fig. 9.2.D.

The sub-pile room is directly below the reactor and is a shielded room allowing access to several experimental facilities and reactor components. The experimental facilities include two VG holes which originate at the top of the reactor and terminate in the sub-pile room, the two standard hydraulic rabbit tubes which extend up alongside the active lattice, possible hydraulic rabbit tubes to be inserted into the shim or control rod locations, and possible VT holes extending from the top of the reactor through the shim rod locations. In addition to serving these facilities, the sub-pile room permits access to the canal and to several reactor components for maintenance, repair,

or replacement. These include items such as the reactor bottom plug, neutron curtain, and unloading machine. Heavy lead shielding doors are available on each of the two sub-pile room entrances to provide protection for personnel outside the room during reactor unloading operations. Service connections for compressed air, demineralized water, warm drain, electricity, telephone, and hot exhaust are located in this room for the benefit of those conducting experiments.

The remaining facilities of the basement and their functions are as follows:

*Canal.* This is designed for the handling and storage of radioactive materials in general and spent fuel elements in particular. The canal and its facilities are discussed fully in Chap. 6.

*Storage vault.* This vault is designed in accordance with security regulations. Walls and ceiling are of 1-ft-thick reinforced concrete. The entrance door is a Merchandise Vault Door, Class D. The vault is primarily a storage place for nonactive fissionable material.

*Monitoring room.* This shielded room contains the instruments and other equipment used in the continual monitoring of reactor cooling-water temperature and flow and to detect fission breaks in the fuel elements. In addition, instrumentation for the log $N$ measurements is located in this room.

*Sumps.* Three sumps are located below the basement floor; each one is set at an elevation that provides gravity flow toward it. The canal sump is provided as the pumping point for the canal cooling-water overflow and for the water resulting from draining the canal or sections of the canal. The process-water sump collects water which might possibly, but not probably, pick up some radioactive contamination. The sump is both a retention and a monitoring station. The effluent sump contains a collecting tank where water that has a higher degree of radioactive contamination can be collected and held if necessary.

*Piping and conduits.* Process piping and duct work are installed in tunnels under the basement floor. This is necessary not only to properly shield them but also to keep them from occupying basement space. The bulk of the electrical conduits, excluding reactor control and instrument circuits, are

also placed under the floor since there appears to be little need for maintenance or repairs on such lines. Service piping and reactor conduits are suspended from ceiling hangers, for it is anticipated that changes and repairs on these lines will be necessary from time to time. Groupings of service outlets are set along the outside basement wall in a fashion similar to those on the first floor.

*Experimental area.* Much of the remaining open space in the basement is available for various "hot" experiments. Experiments involving the circulation of liquids into an active zone of the reactor are to have their auxiliary apparatus located in the basement. Piping from beam-hole experiments to their auxiliary apparatus is taken through the facility holes. This keeps the major portion of "dirty" work in the basement area.

## 9.3 REACTOR BUILDING WING

**9.3.1 General Building Details.** The architectural treatment is consistent with that used for the Reactor Building. While these two buildings appear to be one, they are separate facilities. They have no structural dependence on each other in order to avoid transmission of vibrations from one building to the other.

The wing building is a single-story structure whose internal layout is determined primarily by the laboratories and their auxiliary facilities. This laboratory building has a flexibility such that offices can be readily converted to laboratories or the reverse. A modular plan is the means of achieving this flexibility.

The windows are provided to utilize daylight and to eliminate blank wall space. Owing to the dust problem and the severity of the winters, these windows are sealed-in, double-glazed sash.

The ventilation system represents considerable planning in accordance with general principles set down for radioactive work in laboratories. In general, six air changes per hour are supplied. The air flow pattern is such that the flow within the building is from "subnormal" to "normal" to "above normal" areas. This procedure is based on proved project experience and requires that 100% fresh air be brought into the single "subnormal" area, the

counting room. The air exhausted from this room is used for any remaining areas. "Normal" facilities, such as offices and shops, exhaust their air into the corridors for reuse in the laboratories. The "above normal" facilities, the laboratories, completely exhaust their air supply to the atmosphere. To make certain that no laboratory air enters the corridor, the fresh air supply to the room is set below the exhaust fan capacity. With this provision the excess is taken into the room from the corridors through louvers in the doors. These elaborations on normal building ventilation are made in order to avoid spread of contamination in the event of spills involving radioactive materials.

Basement ventilation is set at one air change per hour and is based on the entire basement volume. The total air supply is provided through one register. As the need arises for subdivision of the basement area into rooms, ventilation ducts can be connected to this outlet.

**9.3.2 First Floor.** The plan of the first floor of the Reactor Building wing is such that the facilities having a permanent operating function as their initial function could have fixed partitions. Those not falling into this classification follow a modular plan with partitions of a type readily removed or set in place.

The facilities in the former class are the machine shop, staff shop, instrument shop, glass shop, stores, shipping and receiving rooms, counting room, and men's and women's toilets and change rooms. In the latter classification are the offices and laboratories. Although the laboratories are in no sense temporary, some are smaller than desired because of the necessity of providing temporary office space in this building. The first floor layout is illustrated in Fig. 9.3.A.

*Machine shop.* The prime purpose of this shop is to provide maintenance and repair facilities for reactor components. It is located as shown in the figure chiefly in order to be convenient to the Reactor Building. The floor is capable of taking a uniform load of 250 lb/sq ft, which permits fabrication of or repairs to light-to-moderately heavy coffins. Layouts of several possibilities of machine tools are available which prove the feasibility of placing a variety of light- and medium-capacity machines. Specifying the size, number, and brand of this equipment is left to the operating contractor. As opinions vary widely on the relative merits of machine tools, it appears that the shop superintendent should choose those in which he has most confidence.

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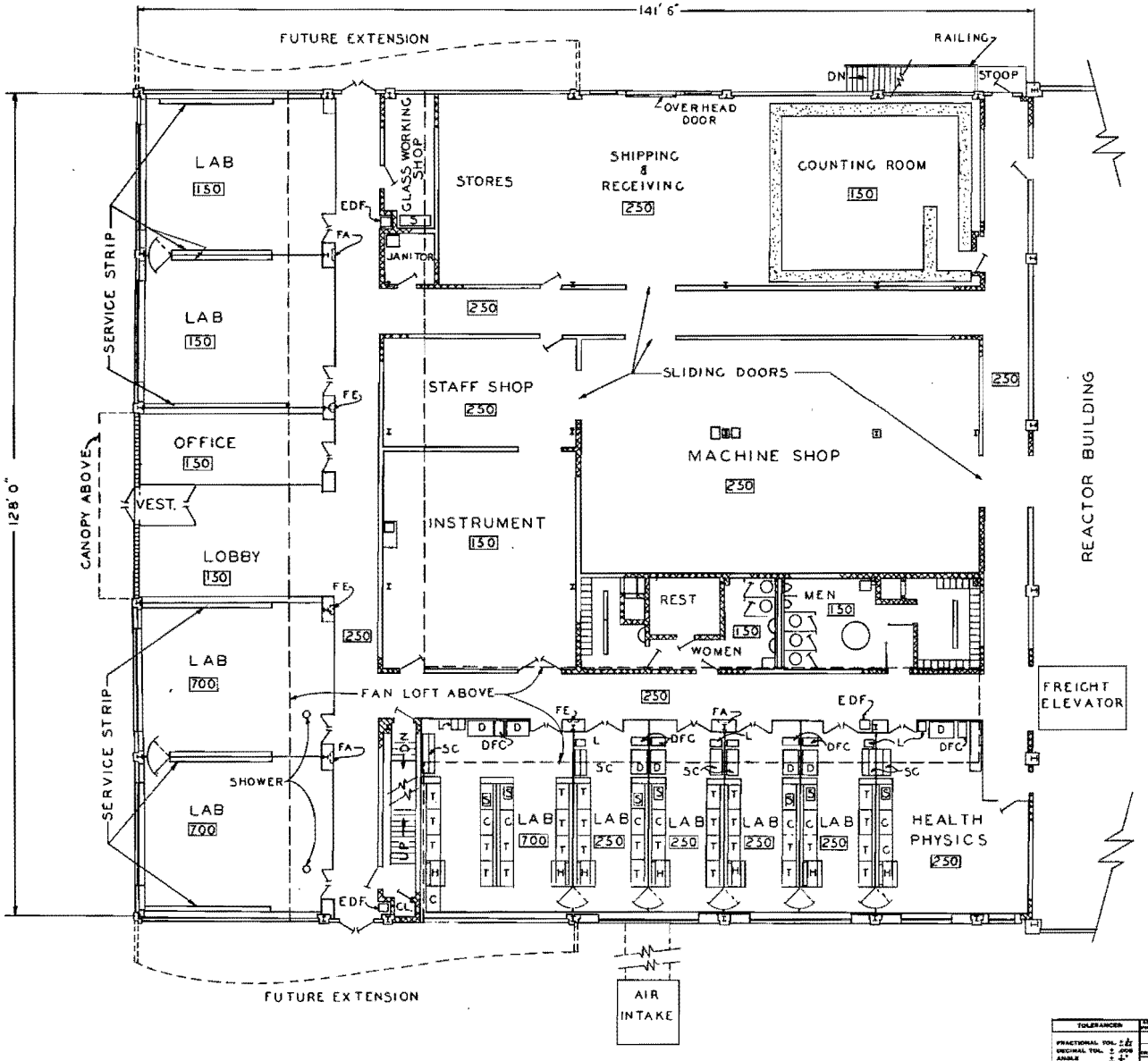


SYMBOLS FOR FURNISHINGS AND EQUIPMENT

| | |
|---|---|
| UNDERBENCH CABINET | C |
| DESK | D |
| DIAL LOCK FILE | DFC |
| HOOD | H |
| LOCKER | L |
| SINK | S |
| SUPPLY CABINET | SC |
| TABLE | T |
| FIRE EXTINGUISHER | FE |
| FIRST AID CABINET | FA |
| ELEC DRINKING FOUNTIAN FIRE HOSE ABOVE | EDF |

NOTE:

▭ INDICATES LIVE LOAD – LBS. PER SQ. FT.

| TOLERANCES | | ARGONNE NATIONAL LABORATORY | | |
|---|---|---|---|---|
| FRACTIONAL TOL. ± 1/8 | | This drawing is the property of the Argonne National Laboratory | | REACTOR BLDG. WING |
| DECIMAL TOL. ± .005 | | | | FIRST FLOOR |
| ANGLE ± 1/2° | | | | |
| ALL DIMENSIONS TO BE CHECKED ON SHOP WORK BY AN INSPECTOR | | | | |
| UNLESS OTHERWISE NOTED | | | | FIG. 9.3.A |
| DRAWN | F.T. 11-24-50 | | | |
| DRAFT | BUTTER LEVSE | | | |

*Staff shop.* This shop is adjacent to the machine shop, convenient to the supply point, and centrally located to the possible users. It is provided to serve the experimenter who wishes to do some of his own machine work. It also contains the light power tools necessary for instrument shop work.

*Instrument shop.* This shop is primarily for maintenance and repair of the electronic instruments used for reactor operations. The shop size is such that it also permits some maintenance of electronic instruments used for experiments. Construction of new instruments is beyond the intent or facilities of this shop. As with the other shops, equipment and its layout are placed within the operating contractor's jurisdiction.

*Glass shop.* This small shop is intended to be a service function for the experimenter. The room itself is relatively small but should be adequate for the initial testing program.

*Counting room.* This room is next to the corridor separating the reactor and the wing buildings. This represents a somewhat central location to facilities in both buildings. Its location is along an outside wall, rather than in the center of the building, in order to aid in eliminating the transmittal of machine vibrations into the room. Although present philosophy states that the counting instruments rather than the room proper should be shielded, 2-ft-thick concrete walls and ceiling are provided. The reasons for adhering to the older practice are (1) the unknown factors connected with the stray radiations from a high-intensity reactor located approximately 60 ft away and (2) the increase of cosmic radiations due to the relatively high altitude of the site. Beta and gamma counting are done here, but alpha counting is done in the laboratories. The foundations for this room are separate from the remainder of the building, which is an additional precaution in the elimination of transmitted machine vibrations. Services into this room are low-pressure air, hot and cold water, and 110-volt regulated current. The room is air-conditioned and the humidity is controlled to 40% R.H. ± 10%. The air supplied is 100% fresh air. Air exhausted from this room is reused as a portion of the ventilating supply in the remainder of the building.

*Stores, shipping, and receiving rooms.* Although these are labeled as somewhat separate areas, they represent a single integrated unit. The only barriers necessary are a counter or possibly a wire mesh partition. The functions of these facilities are defined adequately by the title.

*Laboratories.* All remaining facilities are designed for use as laboratories. To achieve a flexibility for enlarging or subdividing laboratories, a modular plan is used. The module selected is 12 ft wide and 28 ft deep. The width is one which has had some degree of general acceptance in laboratory design. The depth provides 20 ft for bench space for test work, and 8 ft for desk, storage cabinets, and locker. The first 8 ft in from the corridor wall is for this office-storage function. If any laboratory experiment should involve considerable amounts of alpha work, this 8-ft space can be readily partitioned off from the remainder of the laboratory. This would form somewhat of an air lock to prevent spread of air-borne contamination. The remaining 20 ft of laboratory is used for work bench service. A 4-ft sub-module is used for determining the sizes of equipment such as benches and cabinets and for setting the locations of service outlets such as gas, water, air, and electricity.

Metal partitioning is used because of the following characteristics:

1. Ease of installation or removal without dust problem.
2. Factory baked-on finish, easily cleaned or decontaminated.
3. Nonporous surface giving minimum absorption of radioactivity.
4. Absence of large cracks for accumulation of dusts; the fine cracks between panels are sealed with a putty or a plastic tape.

The ceilings are perforated metal pans with insulation placed in the pans. These are effective, both as sound deadeners and heat insulators. Lighting is by means of fluorescent strip lights set flush with the ceiling.

The floor covering is vinyl sheet flooring. This is used because of its resistance to most reagents and many of the common organic solvents. Also, this material requires less maintenance than other available floor coverings.

The services or utilities available in the laboratories are steam, low-pressure air, gas, hot and cold soft water, 110-volt single-phase and 220-volt three-phase power, demineralized water, and telephone. All but the last two have connection outlets at 8-ft intervals along a bench. Demineralized water and telephones are provided on the basis of one outlet per laboratory, regardless of whether the laboratory is a 12- or 24-ft-wide room. All services are brought into the room from the supply headers located in the basement. Floor slots for these pipes are centered on the 12-ft module separation lines.

Ventilating-air supply is through anemostats. Exhaust is to the atmosphere through fans, of which one is provided for each module. These exhaust fans have a capacity higher than the supply inlets to the room. By this means, air is constantly entering the room from the corridors through louvers in the doors. The exhaust fans have a capacity to provide a minimum of 100 cfm of air through the working openings of two 4-ft hoods. When the hoods are not in use, the same quantity of air is drawn into a dampered by-pass in the hood duct work.

An opening leading to the contaminated-air system is provided in each hood. These off-gas openings are for use with dry-box work and experiments which may give off radioactive or acidic vapors. Each off-gas opening provides an exhaust flow of about 50 cfm.

Four laboratory modules intended for normal laboratory work permit floor loadings of 150 lb/sq ft. The four modules that permit 250 lb/sq ft floor loadings are forwork involving moderate activity requiring light lead shields. The six modules that permit floor loadings of 700 lb/sq ft can take one cave per double module. Such a cave could have a 4- by 6-ft working area and be shielded by 30- to 36-in. concrete walls 6 to 7 ft high. A cave of this type could handle up to 10 curies of activity.

Owing to lack of office space on the site, laboratory furnishings for most of the laboratories along the west wall are not provided. This space will be used for offices until such time as an administration building is constructed.

**9.3.3 Basement.** The building has a full basement which gives adequate space for ventilating and air-conditioning equipment for areas 3 and 4, electrical switch gear for these same areas, the battery room for emergency use by the reactor operating instruments, and the scrubbing system for the contaminated air from laboratory hoods. This is illustrated in Fig. 9.3.B. The massive foundation work for the counting room forms a fireproof record room. Of the total basement area, all space is fairly well occupied except the first bay starting at the west wall and extending the full north-south dimension of the building. This can be used for placing future equipment or for subdividing into storage rooms such as a parts storage room for the instrument shop.

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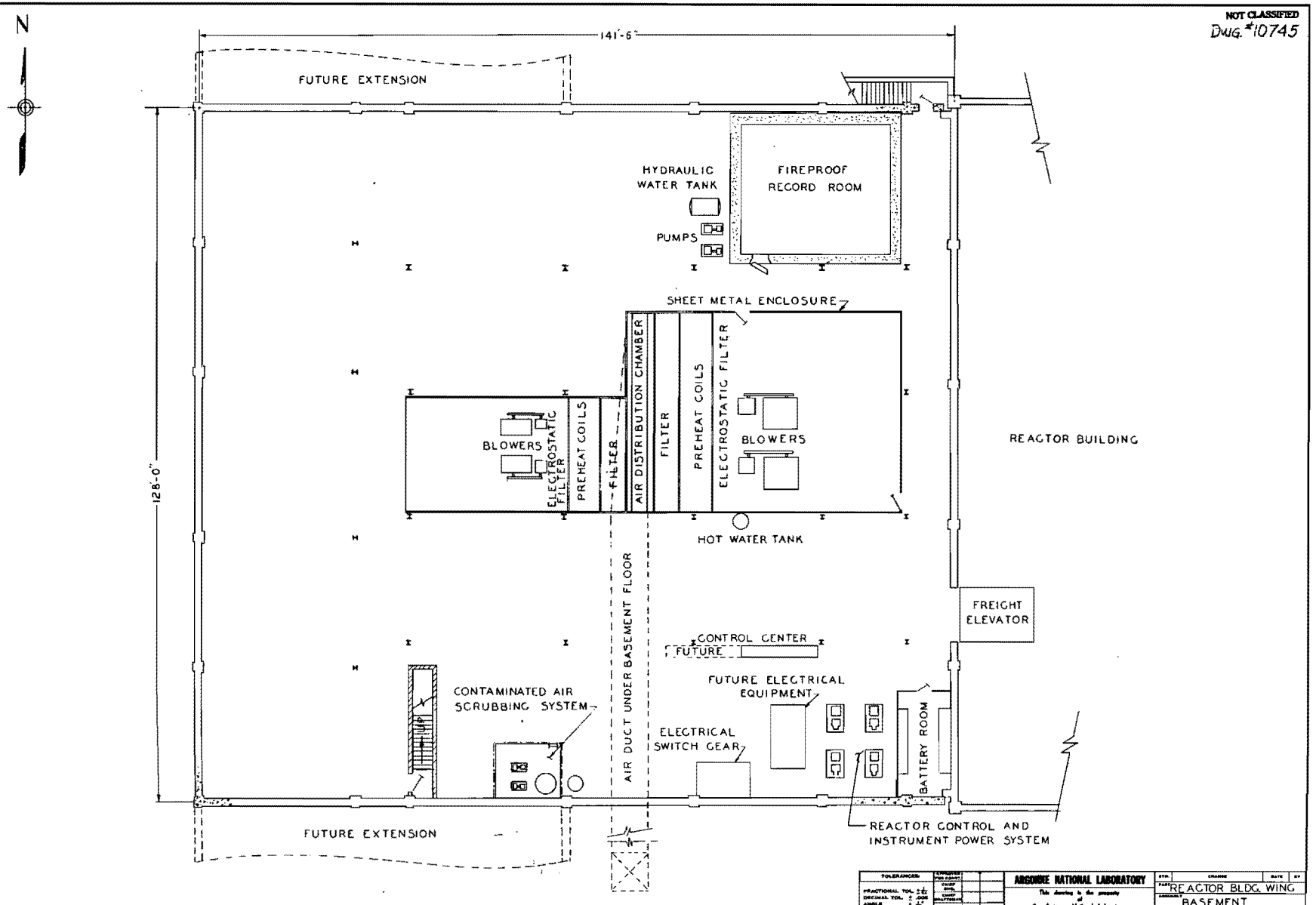


| | | | |
|---|---|---|---|
| TOLERANCES: FRACTIONAL TOL. ± 1/64 DECIMAL TOL. ± .005 ANGLE ± 1/2° ALL DIMENSIONS IN INCHES UNLESS OTHERWISE NOTED | ARGONNE NATIONAL LABORATORY This drawing is the property of the Argonne National Laboratory | PART: REACTOR BLDG. WING | |
| | | ASSEMBLY: BASEMENT | |
| DRAWN F.T. 11-13-50 | SCALE 1/8" = 1' | SHEET OF | FIG. 9.3.B |

**9.3.4 Future Additions.** This building is so designed that 70-ft-wide wings can be added both north and south of the present structure. The points of future extension are shown in Figs. 9.3.A and B. The original building layout included extensions of 72 ft both north and south to give a T-shaped building. Budgetary limitations brought about a cut-back of twenty-four 12- by 28-ft laboratory modules, and thus eliminated facilities such as a library, drafting room, and technical staff offices. These are necessary functions for a long-period operation of the reactor, and for this reason will probably be added shortly after reactor operation begins.

## 9.4 SERVICE BUILDINGS AND AREAS

**9.4.1 Service Building, Area 14.** This single-story 42- by 102-ft building combines four separate service functions into one structure:

1. First aid. This section has a two-bed infirmary, waiting room, first aid room which can be used for physical examinations, toilet rooms, and assorted closets for storage of medical and janitorial supplies.

2. Guards' section. This portion is subdivided into two rooms. One room has locker, shower, and toilet facilities for a total of 50 guards with a maximum of 20 per shift. The second room is the guards' ready room.

3. Communications. This section is also a two-room portion of the building. One room houses the dial telephone exchange equipment; the other has the radio apparatus required by Security.

4. Fire house. This section has garage space for one pumper truck, the firemen's office, and sleeping quarters for firemen on 24-hr shift basis.

**9.4.2 Limited-area Control House, Area 20.** This 22- by 42-ft single-story building is located at the main entrance gate and is the screening point for all personnel and vehicles entering or leaving the site. Guard stations are provided along the central portion of both the north and south walls. These two stations can take two lines of persons checking in or out. In addition, the north guard station has a door leading into the truck trap located directly outside the building, which constitutes the screening point

for vehicles. Both the east and west ends of the building are small lobby sections which can accommodate the back-up of employees during heavy periods of personnel movement. All badge exchange is handled in this building.

**9.4.3 Exclusion-area Control House, Area 21.** This is a single-story 12- by 12-ft building which acts as a second screening point for persons entering the exclusion area. The building is subdivided north and south into two portions, one the guard station and the other a corridor section through which personnel pass on their way to or from the exclusion area.

**9.4.4 Canteen, Area 16.** The canteen is a portion of one of the temporary metal buildings erected for the construction firm. Cafeteria style service is provided. The food is prepared at AEC Central Facilities. Seating capacity is relatively small so that three or four seatings may be required to serve all those purchasing their lunch.

**9.4.5 Plug Storage, Area 11.** Plug storage is located south of the Reactor Building on the main truck road leading out of the building. This is an earth-filled structure with a concrete charging face on the east side. Steel liners used for the holes are held level by means of the concrete charging face and a supporting wall located 13 ft into the earth-filled section. Fourteen holes, ranging in size from 19 by 15 in. to 5½ in. inside diameter, are located at a convenient height above the roadway for charging plugs into the holes. A minimum of 8 ft of earth shielding is used on top, sides, and ends of all these 20-ft-long tubes.

**9.4.6 Reactor Service Building.** This building is not a part of the initial construction. However, planning has progressed to a point where the building layout and facilities have been fairly well established. The building has four prime functions: (1) plug storage, (2) machine shop, (3) warehouse, and (4) operations testing. The four facilities require approximately the same square footage of floor area, 6000 sq ft each, so that a 160-ft-square building is used. The overall building height is 30 ft with 22 ft clearance under the roof trusses.

Plug-storage equipment consists of a concrete block 75 ft long by 10 ft high by 21 ft deep. One of the 10- by 75-ft faces is the charging face for placing the plugs into coffins or into storage. A 20-ton 75-ft-span crane operates over this area. The crane can handle the plugs and coffins or the

storage of shielded pots on top of the plug-storage block. The remainder of the room area in front of the storage block is required for vehicle movement, for storage of vehicles and coffins, and for the erection of cave structures for dismantling "hot" plugs.

The machine shop is adjacent to the plug storage room so that the plug-handling crane can be utilized by the shop. The shop provides for machine tool work, lead burning, welding, and sheet metal and pipe work. All these functions are designed specifically for exclusion-area work.

The warehouse section is a two-floor area. The lower floor is designed for storage of heavy materials such as lead bricks or steel shielding plates. The space, in general, is to store items needed for exclusion-area work. Some office space is also available on this floor. The upper floor is designed for lighter storage and/or offices.

The operations testing area is designed for the setup of mock-up or functional test apparatus. A 10- by 50- by 10-ft-deep pit is located in one corner to allow for vertical apparatus height approaching 40 ft. Service outlets are provided in the pit and along the outer walls for the benefit of the experimenter.

There will be a definite need for such a building on the site. Although it has not been constructed as a part of the initial program, it should be actively considered within the first year of operation.

**9.4.7 Other Buildings and Areas.** The remaining buildings and areas as defined by the Site Plan (Fig. 9.1.B) are directly or indirectly associated with other major sections of the handbook and are discussed therein.

## 9.5 REFERENCE REPORTS


(1) Guzik, R. F., and Link, L. E., *Reactor Building,* MTR Design Report No. 26, October 10, 1949.

(2) Guzik, R. F., and Link, L. E., *Addendum to Reactor Building Report,* MTR Design Report No. 26A, November 11, 1949.

(3) Guzik, R. F., and Link, L. E., *Reactor Building Wing,* MTR Design Report No. 27, October 12, 1949.

(4) Guzik, R. F., and Link, L. E., *Addendum to Reactor Building Wing Report,* MTR Design Report No. 27A, November 25, 1949.

(5) Guzik, R. F., and Link, L. E., *Site Plan,* MTR Design Report No. 36, May 3, 1950.

(6) Sullivan, T. E., *Subpile Room,* MTR Design Report No. 46, ANL-ABS-19, April 19, 1950.

## 9.6 REFERENCE DRAWINGS

(Blaw Knox Construction Company Drawings)

| DWG. NO. | TITLE |
|---|---|
| 3150-3-1, Rev. 4 | Reactor Building - Basement Plan |
| 3150-3-2, Rev. 3 | Reactor Building - First Floor Plan |
| 3150-3-3, Rev. 3 | Reactor Building - Second and Third Floor Plans |
| 3150-3-5, Rev. 3 | Reactor Building - Longitudinal Section |
| 3150-3-6, Rev. 3 | Reactor Building - Transverse Section |
| 3150-3-7, Rev. 2 | Reactor Building - Facilities Below Basement |
| 3150-4-1 | Reactor Building Wing - Basement Plan |
| 3150-4-2 | Reactor Building Wing - First Floor |
| 3150-4-2, Rev. 1 | Reactor Building Wing - First Floor |
| 3150-4-3 | Reactor Building Wing - Elevations and Section |
| 3150-11-1 | Reactor Service Building |
| 3150-14-1, Rev. 1 | Safety and Security Building |
| 3150-20-1, Rev. 1 | Control House - Limited Area |
| 3150-21-1, Rev. 1 | Control House - Exclusion Area |
| 3150-91-5, Rev. 1 | Plot Plan Study |
| 3150-91-5, Rev. 9 | Plot Plan |
| 3150-502-3, Rev. 2 | Mechanical Design - Subpile Room |

# Chapter 10*

# SUPPORTING FACILITIES AND AUXILARIES

## 10.1. STEAM PLANT

The steam plant is a 66- by 110-ft building located approximately in the center of the northern half of the MTR site. This building contains the following equipment:

(1) Steam-generating facilities to supply process and heating steam requirements

(2) Diesel-electric generator to supply emergency power requirements

(3) Air compressors, two to supply plant air during normal operation and one to supply instrument air during emergencies

The steam-generating facilities consist of three package-type steam-generator units with fire-tube boilers and their associated equipment. Each unit has a maximum capacity of 16,500 lb/hr at a design pressure of 150 psig. Two of the units are adequate to supply the maximum process and heating steam requirements, while the third unit serves as a stand-by. The units are oil-fired but can readily be converted to be oil-gas-fired at any time. The operation of the units is fully automatic.

In the event of commercial power failure the Diesel-electric generator, which is rated at 750 kva, can supply power to processes requiring electricity for emergency operation. Each of the two plant air compressors can supply 300 scfm of air at 150 psig. Only one of these is in operation normally, the second serving as a spare. The third compressor can supply 60 scfm of air at 125 psig to instruments during commercial power outages.

## 10.2 ELECTRICAL SYSTEMS

**10.2.1 Introduction.** The electrical systems incident to the operation of the MTR have been designed for high reliability in accord with requirements peculiar to operation of the project and local conditions. As the principal means for precluding interruption or failure of these electrical systems, dual

*ANL contribution.

feeder arrangements are used as a basis of design wherever practical. The systems providing power within the MTR site for all uses are discussed with regard to sources of electrical power, distribution systems, load power factor, and electrical grounding.

**10.2.2 Sources of Electrical Power.** Since the requirements of processes and facilities of the MTR project vary as to the permissible duration of interruptions in electrical power, it has been necessary to provide other sources of electrical power in addition to the normal power from the public utilities. These other sources of electrical power have been made locally available within the site for continuous and emergency use.

The normal electrical power is provided to the MTR site substation through the RTS (Reactor Testing Station) Central Facilities Control Station over a high-line operating at 132 kv. This high-line comprises part of the ultimate RTS distribution system which initially provides only a single power feeder to the MTR and other sites. When the present plan for distribution of power at 132 kv within the RTS area is complete, this feeder will be extended through the area and back to the Central Facilities Control Station to provide a loop transmission system. The loop system will, in effect, provide two feeders to each of the sites so that power transmission to any site may be over either one or both sections of the loop associated with the site.

The MTR substation is located within the MTR site on the north building line near the east corner. This location complies with the requirement that there be a minimum distance of 1000 ft between any electrical system operating at voltages as high as 150 kv and the precise electronic equipment located in the Reactor Building. The location is also desirable with regard to future site considerations and minimizes the problems of interference associated with the entrance and exit of the high-line feeder circuit.

The principal high-voltage equipment at the site substation includes two high-line transformers and one high-line breaker. Each of the transformers, having a self-cooled rating of 3750 kva and forced air cooling to permit a maximum capacity of 4687 kva, is equipped with automatic tap changing equipment which operates under load to permit some control over voltage variations caused by load factors and other conditions. The primaries of these transformers are connected to the high-line circuit with the high-line breaker between them so that opening of this breaker may permit power to be received

by either of the transformers over the associated section of the loop (when completed). The secondary of each transformer delivers power through a breaker to one of the 2400-volt bus sections in the substation control house. These bus sections of the 2400-volt dual feeder system are associated with a tie breaker to permit independent operation of each section or unitary operation of the entire system.

Although an operator at the substation control house may have control over all 2400-volt circuits, the incoming bus circuits and the bus tie circuit may also be controlled by the RTS area power dispatcher. Control over a number of motor-driven disconnects has been made available to the area dispatcher so that control of high-voltage equipment and the distribution at the site substation may be accomplished with only a single high-line breaker.

The sources of locally available electrical power for continuous use are power-accumulator systems which provide uninterrupted power to a number of important loads. The basic components of each system are a storage battery and a conversion device. During normal operation the normal power system furnishes power through the conversion device to maintain the battery "floating" at full capacity and supply the average power required by the load. Since the arrangement of the battery and conversion device permits the load to be supplied from either one or both simultaneously, there is no interruption to the load when an interruption occurs in the principal power system.

A number of relatively small power-accumulator systems are included as parts of the equipment to which they provide power. Equipment having these systems includes the automatic dial and conference telephone system, the security and fire-alarm system, radio installations, and engine-starting systems.

Other power-accumulator systems, each supplying power to a number of different loads, are strategically located in the substation control house, the steam plant, and the basement of the Reactor Building wing. The first two of these systems are primarily to provide continuous power for control of electrical circuits and emergency lighting. The third system primarily serves the power system for control and instrumentation of the reactor.

The power system for operation of reactor controls and instrumentation is provided to ensure these loads continuous a-c power free of interruptions exceeding 2 cycles duration with voltage dips during this time not greater

than 30%. In addition to the conversion devices and battery of the power accumulator, the principal operating components of this system include two independent inversion devices. For reasons of precedence in reliability, M-G (motor-generator) sets are used for both conversion and inversion. Each of the M-G inversion sets operates continuously from the power accumulator to provide a-c power to its respective bus with very close regulation of the voltage and frequency. One of the M-G conversion sets is a spare while the other one operates continuously to maintain a balance of power through the reactor power system. The busses of the two independent inversion systems provide the control and instrument power. These busses are associated by circuit breakers so that the control load may be operated directly from normal or emergency power and the instrument load may be carried by either one or both of the inverters.

Locally available power for emergency use only is provided by a Diesel-electric stand-by unit located in the steam plant. This unit has a capacity of 750 kva for the provision of power during emergency to a limited number of processes and facilities of the project which must operate during shutdown of the reactor. The operation of this emergency unit and part of the associated electrical systems is entirely automatic so that power may be provided to most of the emergency loads with the least possible delay when normal power is interrupted. A very fast starting system for the Diesel engine is arranged with a "fail safe" control circuit associated with the 2400-volt bus sections of the substation. The control circuit initiates starting any time that both busses are de-energized. Since the sizing of this emergency unit has been determined primarily on the basis of the automatically connected emergency loads, other loads that may be connected during emergency have been identified for dispatch control. Some of these other loads which require large amounts of power are provided with control circuits to the stand-by control panel. These circuits are controlled by the dispatcher to permit the connection of the loads only at times when the generator has sufficient reserve capacity to maintain voltage regulation.

**10.2.3 Distribution Systems.** The systems for the distribution of all electrical services for the operation of processes and facilities are installed underground throughout the MTR site as indicated by Fig. 10.2.A. With the exception of the perimeter lighting circuits, the electrical conductors for the transmission of power, control, instrument signals, and communications pass from one part of the site to another through underground concrete-encased fiber ducts at a depth below the frost line.

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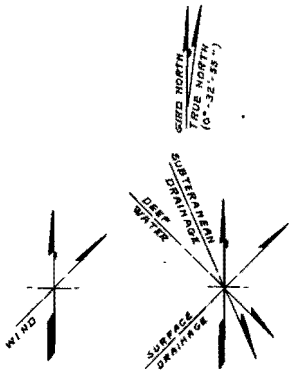
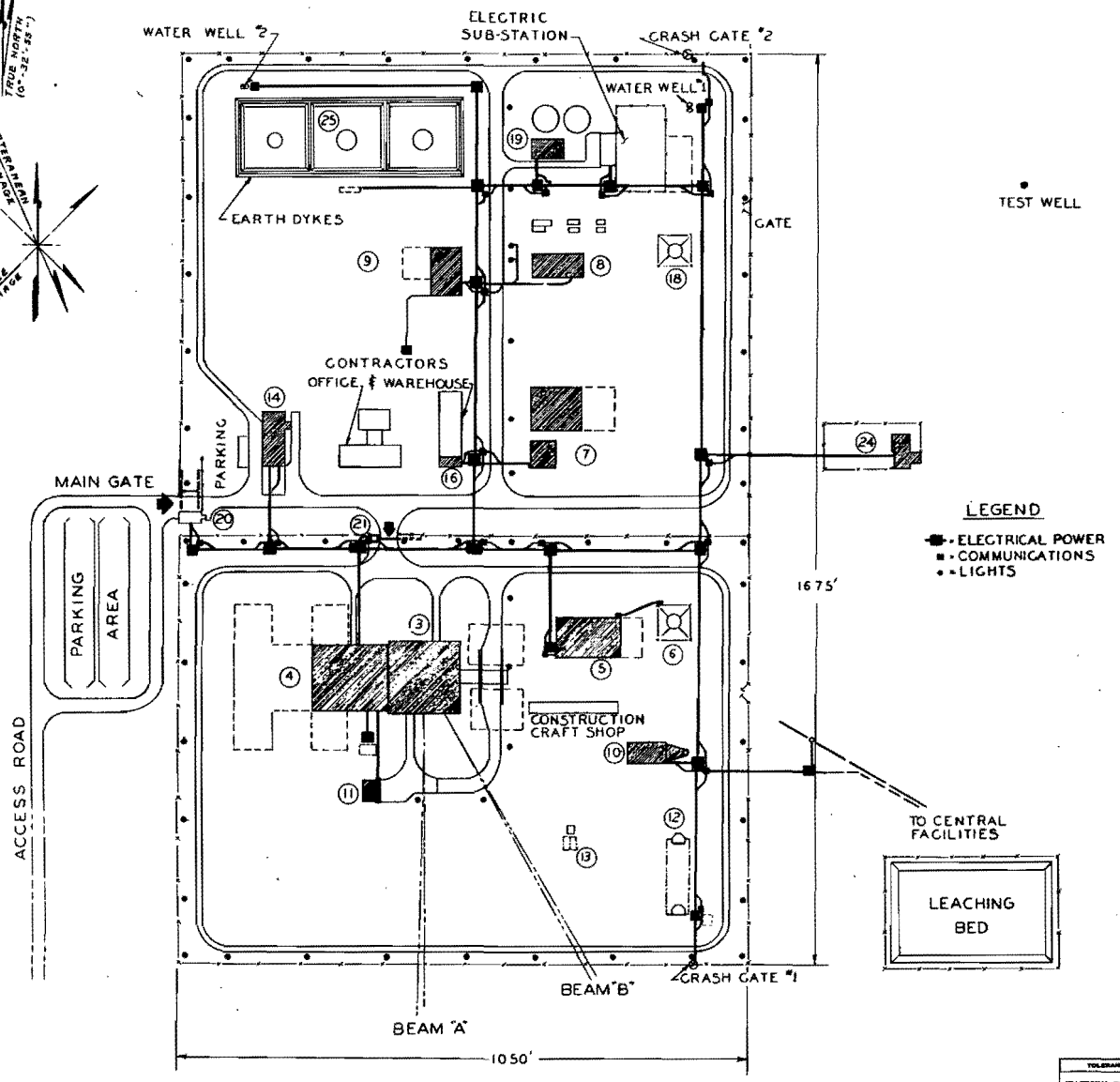
DWG. #10761




## LIST OF AREAS

| NO. | DESCRIPTION |
|---|---|
| 1. | YARD (OUTSIDE FENCE) |
| 2. | YARD (INSIDE FENCE) |
| 3. | REACTOR BUILDING |
| 4. | REACTOR BUILDING WING |
| 5. | PROCESS WATER BLDG. |
| 6. | WORKING RESERVOIR |
| 7. | COOLING TOWER & PUMP HOUSE |
| 8. | DEMINERALIZER BLDG. |
| 9. | STEAM PLANT |
| 10. | FAN HOUSE & STACK |
| 11. | PLUG STORAGE |
| 12. | RETENTION BASIN |
| 13. | EFFLUENT CONTROL |
| 14. | SERVICES BLDG. |
| 15. | |
| 16. | CANTEEN |
| 17. | |
| 18. | EMERGENCY O.H. WATER TANK |
| 19. | RESERVOIR & WATER PUMP HOUSE |
| 20. | CONTROL HOUSE-LIMITED AREA |
| 21. | CONTROL HOUSE EXCLUSION AREA |
| 22. | |
| 23. | |
| 24. | SANITARY SEWAGE TREATMENT PLANT |
| 25. | OIL STORAGE |

—·—·— UNDERGROUND
– – – – – FUTURE

| | | | |
|---|---|---|---|
| TOLERANCES: FRACTIONAL TOL. ± 1/32, DECIMAL TOL. ± .005, ANGLE ± 1/4°, ALL DIMENSIONS IN THIS DRAWING ARE IN INCHES UNLESS OTHERWISE NOTED | ARGONNE NATIONAL LABORATORY — This drawing is the property of the Argonne National Laboratory | SCALE 1"=100' | ELECTRICAL UNDERGROUND DISTRIBUTION & SITE LIGHTING |
| DRAWN F.T. 12-13-56 | | | FIG. 10.2.A |
| APPROVED HUTTER | | | |

The perimeter lighting is part of the area lighting system for the site which includes the interior roadway lighting. These parts of the area lighting system are similar series circuits operated in two groups, each from its own constant-current transformer. One circuit provides power to every second light along the site boundary fence and roadway, while the other circuit includes all other area lights. Since there are no overhead service installations within the site, the feeders from the transformers to each of the lighting circuits pass through underground ducts. Each of the series lighting circuits passes underground from light to light, except around the perimeter where overhead service is carried by the poles that support the lights.

The underground system of fiber ducts is arranged in two groups in the concrete encasement. A large group is provided for power and control cables, with a small group provided for communication and instrument signals. As required for installation of the cables, manholes located between sections of the ducts are provided with a large section for power and a small section for communications. Each section of the combination manhole is accessible from above through a top cover plate.

The transmission of normal electric power is by a dual feeder system, wherein a feeder is extended from each of the two 2400-volt substation bus sections to each of a number of site load centers. In addition to breaker protection at each end of the feeders, there is a breaker between the feeders at each load center to prevent circulatory power transmission. Thus each load center has in effect a two-section bus. Wherever economy of design and division of processes or facilities has permitted, dual power feed is carried onto the load so that faulting of one part of the distribution system affects only part of a given load.

The emergency power generated by the stand-by unit in the steam plant is transmitted by single feeders operating at 2400 or 440 volts. These feeders pass through the underground ducts to the automatic switches of the different emergency load centers. The automatic switches operate in interlocked sequence to transfer their associated loads from normal power circuits to the emergency feeders only when there is an interruption of normal power and when emergency power is available. The transfer switching arrangement precludes any necessity for synchronizing since the load is removed from one power circuit before connecting to the other. With the return of normal power, these switches must be returned to the normal position by manual control before the emergency stand-by unit can be shut down.

**10.2.4 Load Power Factor.** Electrical power consumption in the project is principally by electric motors. Since the resultant overall load is predominantly inductive, power factor correction has been studied with regard to the effects on voltage regulation and penalty costs for excessive amounts of reactive power. In the absence of power contract data and other information pertinent to design for a good operating power factor or low reactive power demand, the use of synchronous motors was abandoned in favor of other means of correction.

Arrangements have been made for power factor correction at some time in the future if required. This is simply the provision of available space for the addition of capacitor banks to the operating circuits of the large motors and to the main substation bus. The "bus correcting" capacitor bank at the substation will in effect correct for the reactive power demand of the numerous small motors in use throughout the site.

**10.2.5 Electrical Grounding.** Adequate protection to personnel and equipment has been provided by extensive grounding systems, including approved methods of grounding throughout the site to prevent electrical hazards. Additional provisions have been made for establishing a constant radio-frequency ground reference for electronic instrumentation within the Reactor Building and Reactor Building wing.

The electrical-hazard grounding arrangements are provided for all electrical services, static discharge, and high potential surges due to lightning. A general grounding system provides appropriate grounding of electrical equipment and metallic structural members of buildings, fences, etc., to underground cables extending to the casings of the water wells as zero ground reference. Additional grounding means for surge absorption of static or lightning are provided in the form of underground grids. These grids, composed of buried mats or driven arrays of grounding rods cabled directly to the zero ground reference, are located in the immediate vicinities of the electrical substation and elevated structures such as the stack, water towers, and certain buildings equipped with lightning rods.

The radio-frequency grounding arrangements for precise electronic instrumentation to be used within the Reactor Building and Reactor Building wing are designed to maintain instrument background interference at a very low level by restricting the generation, transmission, and radiation of extraneous

voltages within or near these buildings. The generation of these unwanted voltages, which is inherent to some degree in many types of electrical equipment, may be amplified by varying potential gradients in partially grounded structural members. These electrical resistance variations between closely adjacent metallic members in the structure of the buildings and within the buildings have been eliminated by isolating or bonding these members. Radiation of disturbances into the buildings is reduced by a Faraday cage effect obtained by grounding the electroweld reinforcing grids of the precast concrete walls and roof panels to the structural steel.

Even though the electrical equipment associated with dust collections, commutation, and the opening or closing of simple circuits includes design to prevent the transmission or radiation of extraneous voltages, it is anticipated that additional filtering and shielding will be designed when complete installation test data are available.

## 10.3 WATER

**10.3.1 Introduction.** The various kinds of water used at the MTR site are classified according to their treatment and use. Demineralized water is used in the MTR process-water system and for experimental purposes. A blended, partially softened well water is in general use throughout the MTR site. The softened water minimizes scaling in hot-water systems and facilitates washing. The water used in the cooling-tower water system is well water which is acid neutralized in order to minimize scaling and corrosion. All these treatments are made in the water-treatment building. Raw water is used in the canal. A flow sheet of the entire water-distribution system is shown in Fig. 10.3.A.

**10.3.2 Water Requirements.** The water-flow requirements of the several different kinds of water used on the MTR site are summarized below. Breakdowns of the uses of each kind of water are given where they are appropriate. The designs of the water-treatment facilities are based on the total flow requirements given in this summary.

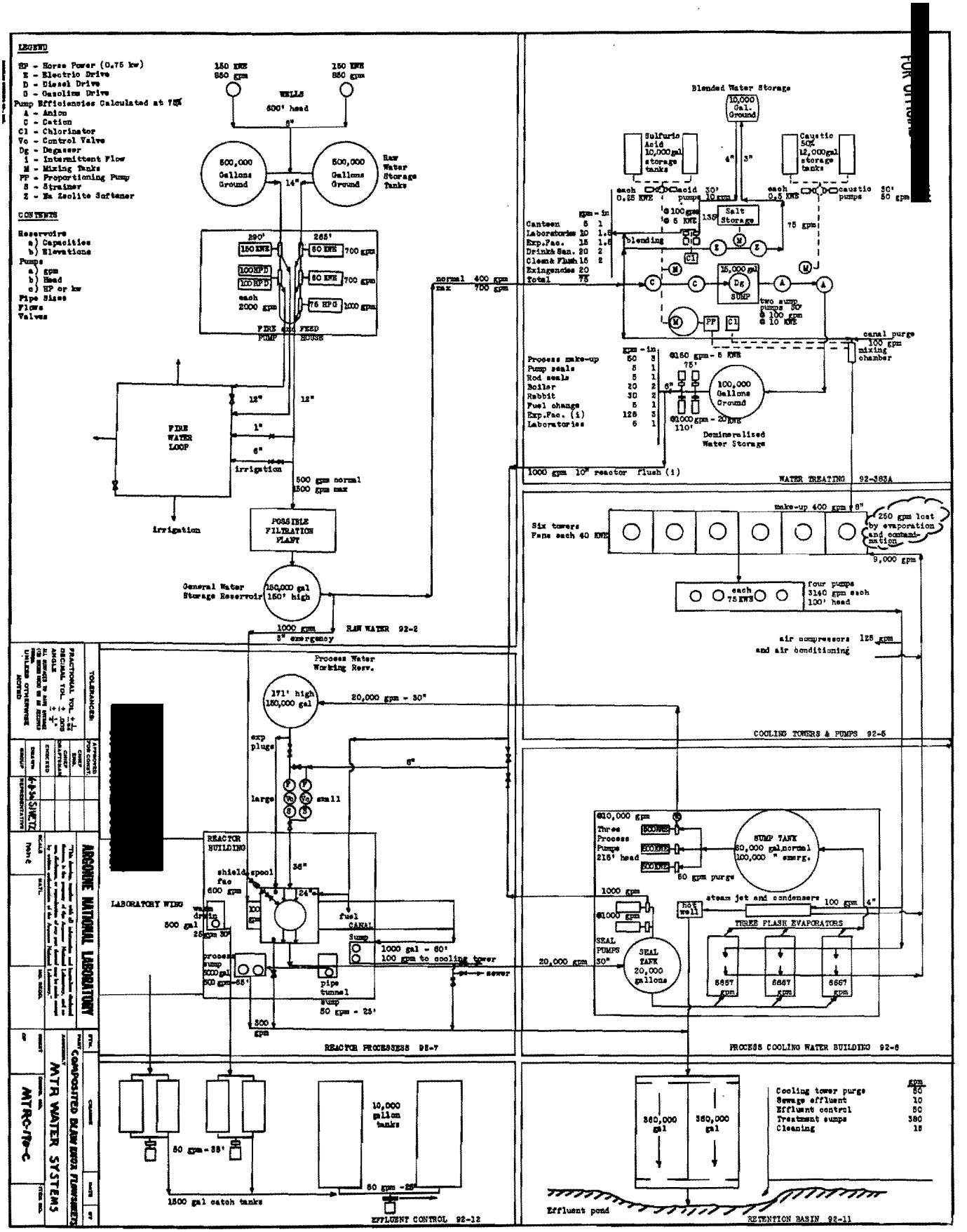



FIGURE 10.3A

MTR WATER SYSTEMS

## TABLE 10.3.A

### Total Water Requirements

| | |
|---|---|
| **Demineralized water** | |
| Process water make-up | 50 gpm |
| Boiler | 20 |
| Rod and pump seals | 10 |
| Rabbits | 30 |
| Laboratory | 5 |
| Experimental purposes and contingencies | 35 |
| Total | 150 gpm |
| **Softened water** | |
| Canteen | 5 gpm |
| Laboratories | 10 |
| Experimental facilities | 15 |
| Drinking and sanitary | 20 |
| Cleaning and flushing | 15 |
| Contingencies | 10 |
| Total | 75 gpm |
| **Cooling-tower water** | |
| Evaporation | 250 gpm |
| Purge | 50 |
| Contingencies | 100 |
| Total | 400 |
| **Raw water** | |
| Canal purge | 100 gpm |

**10.3.3 Water Analysis.** The raw water analysis is given in Table 10.3.B. The analysis was made by the U. S. Department of the Interior. The sample was obtained from the MTR test well.

## TABLE 10.3.B

## Raw Water Analysis

Analysis by the United States Department of the Interior, Geological Survey, Water Resources Branch

Analytical Statement

SW NE ¼ Sec. 14 T3N R29E

(Parts per million)

LOCATION: Butte Co., Arco, Idaho

SOURCE: Well (test)

AEC MTR WELL

DEPTH: 588 ft

DIAM: 8 in. ID

WATER LEVEL: 453.9 ft below TC

YIELD: 80 G.M. pump

OWNER: AEC, Idaho Falls

CHEMIST: W. M. Webster

LAB. NO.: 3896

COLLECTOR: RWK

DATE OF COLLECTION: November 30, 1949

USE: Observation

TEMPERATURE: 55 degrees

COLOR: Clear pH: 7.2

SUSPENDED MATTER:

HARDNESS AS $CaCO_3$

N.C.: 31 TOTAL: 208

IGNITION LOSS: 27

DISSOLVED SOLIDS: 240

SPECIFIC CONDUCTANCE AT 25°C (MICROMHOS): 398

T/A FT: 0.32

*TOTAL FE: 0.15

MN: 0.0

| | |
|---|---|
| $SiO_2$ | 19 |
| *FE(SOL) | 0.06 |
| CA: | 52 |
| MG: | 19 |
| NA: | 2.7 |
| K: | 2.2 |
| $CO_3$: | 0 |
| $HCO_3$: | 216 |
| $SO_4$: | 20 |
| CL: | 12 |
| F: | 0.2 |
| $NO_3$ | 4.1 |
| B: | 0.1 |
| SUM: | 238 |

Lab. No. 3896

| | | |
|---|---|---|
| rCa | 2.595 | |
| rMg | 1.562 | |
| rNa | 0.117 | 4.371 |
| rK | 0.056 | -4.330 |
| | 4.330 | 0.041 |
| $rCO_3$ | | 8.701 |
| $rHCO_3$ | 3.540 | |
| $rSO_4$ | 0.416 | |
| rCl | 0.338 | +0.5% error |
| rF | 0.011 | |
| $rNO_3$ | 0.066 | 3% Na |
| | 4.371 | |

HYPOTHETICAL COMPOUNDS

| Raw Water | | Hardness $CaCO_3$ | (TDS) | Alkalinity $CaCO_3$ |
|---|---|---|---|---|
| $Ca(HCO_3)_2$ | 2.6 epm | 130 ppm | 130 | 130 |
| $Mg(HCO_3)_2$ | 1.0 | 50 | 42 | 50 |
| $MgSO_4$ | 0.4 | 20 | 24 | |
| $MgCl_2$ | 0.2 | 10 | 9 | |
| NaCl | 0.1 | | 6 | |
| $KNO_3$ | 0.1 | | 10 | |
| | 4.4 | 210 | 221 | |
| | | $SiO_2$ | 19 | |
| | | | 240 | |

DATE COMPLETED: 12-13-49

CHECKED BY: E. L. Singleton

PROJECT: AEC MTR well

TRANSMITTED: 12-14-49

REMARKS:

*Demineralized Water.* The desired demineralized-water analysis is given in Table 10.3.C. The design of the demineralization system is based on obtaining water that satisfies this analysis.

**TABLE 10.3.C**

**Demineralized-water Analysis**

| CONSTITUENT | MAXIMUM |
|---|---|
| TDS | 3 ppm |
| Fe | 0.05 |
| Al | 0.05 |
| Ca | 0.5 |
| Mg | 0.5 |
| Na | 2 |
| $CO_2$ | 2 |
| $SiO_2$ | 1 |
| Cl | 0.25 |
| $SO_4$ | 0.50 |
| Cu, Pb, Ni, Co, B | None |
| Process water pH | 5.5 to 6.5 |
| Specific resistance at 18°C | 300,000 to 500,000 ohms |

*Soft Water.* The soft water has the same analysis as the raw water except that all the Ca and Mg are replaced by sodium. This completely softened water is blended with raw water to give the blended partially softened water of 50 ppm hardness that is used throughout the MTR site.

**10.3.4 Demineralization System.** *Cation Exchangers, Acid Tanks, and Pumps.* Sufficient cation capacity must be available to continually process 150 gpm of well water. Two cation exchanger units are provided for this purpose; each is 5½ ft in diameter and 9 ft high. Each has about 2 ft of height available for Anthrafilt base, about 3 ft of head room for resin expansion during backwash, and 4.5 ft of resin. The units are regenerated about every 12 hr. A 60-day supply of 98% sulfuric acid for cation resin regeneration is stored out of doors in two 10,000-gal steel storage tanks. The acid is pumped from these storage tanks to a dilution or mixing tank in the water-treatment building where it is prepared for cation regenerative use.

*Degasser and Sump.* In order to remove the $CO_2$ from the decationized water the water is sprayed down over wooden slats in a wooden tank and falls into a lined sump below. The water is pumped from the sump through the deanionizers.

*Anion Exchangers, Caustic Tanks, and Pumps.* Sufficient anion capacity must be available to process the 150 gpm of decationized and degassed well water. Two deanionizers of about the same physical size as the decationizers are provided. However, only about half the depth of resin is required. A 60-day supply of 50% NaOH liquid caustic for anion resin regeneration is stored out of doors in two 12,000-gal steel storage tanks; its temperature is thermostatically controlled and it is heated by steam. The caustic is pumped from these tanks to a dilution or mixing tank where it is prepared for anion regenerative use. The units are regenerated at about 24-hr intervals.

*Demineralized-water-Storage Tank and Pumps.* After demineralization the water is stored at ground level in a 100,000-gal storage tank. All linings in the demineralized-water system are of stainless steel or inert organic base so as to contribute no minerals to the water. This consequently helps to reduce scale formation on the reactor fuel assemblies in the process-water system. Either of two 75-psig 160-gpm centrifugal pumps continually supplies demineralized water for general use while the other pump serves as a spare. A 75-psig 1000-gpm centrifugal pump supplies reactor flush water during shutdown while a second identical pump serves as a spare.

**10.3.5 Softeners.** Sufficient zeolite resin capacity must be available to continually process 75 gpm of softened, blended water. Two zeolite beds about 2.5 ft in diameter and with a 3-ft resin depth are provided for this purpose. These units are regenerated automatically every 12 hr. A 60-day supply of salt (16 tons) is maintained in a brine pit leached with water. Pumps deliver the saturated brine to the dilution-regeneration tank.

The softened water is blended with raw water to yield a blended water of about 50 ppm hardness. It is stored in a 10,000-gal lined steel ground tank so that about a 3-hr supply is available. It is chlorinated before being sent to its points of use. Two pumps are available to keep the blended water line under continual pressure; one of these pumps serves as a spare.

**10.3.6 Acid-neutralization System for Cooling-tower Make-up Water.** Well water is partly neutralized and mixed with dilute sulfuric acid before being sent to the cooling towers, where scaling is controlled in accordance with *p*H records and Langelier saturation indexes. Chlorine is added occasionally to eliminate growth of algae. The chlorine is stored in cylinders in a separate room inside the building. The 98% acid is measured manually into a 50-gal tank and is propeller-mixed and diluted in a 2500-gal tank. The diluted acid is fed into the make-up water line by means of a flow-controlled proportioning pump; this pump delivers the dilute acid and well water in the proper proportions.

**10.3.7 Design and Operating Data Summary.** The pertinent design and operating data for the softening, decationizing, and deanionizing units are summarized in Table 10.3.D.

**TABLE 10.3.D**

**Water System Design and Operating Data**

| | SOFTENING | DECATIONIZING | DEANIONIZING |
|---|---|---|---|
| Resin capacity (grains/cu ft) | 20,000 | 10,000 | 7,000 |
| Resin bed height (ft) | 3 | 5 | 2.5 |
| Resin bed area (sq ft per bed) | 5 | 20 | 20 |
| Resin bed diameter (ft) | 2.5 | 5 | 5 |
| Number of resin beds | 2 | 2 | 2 |
| Design resin bed volume (cu ft per bed) | 15 | 100 | 50 |
| Calculated resin bed volume (cu ft per bed) | 13.3 | 92.8 | 49.4 |
| Resin bed design safety factor (%) | 12.8 | 7.8 | 1.4 |
| Flow rate (gpm/sq ft) | 6 | 5 | 5 |
| Throughput/unit for two units (gpm) | 30 | 100 | 100 |
| Hardness, cations + anions (ppm) | 210 | 219 | 41 |
| Time between regenerations (hr) | 12 | 12 | 24 |
| Amount of water treated (gpd) | 86,400 | 288,000 | 288,000 |
| Regenerant used | NaCl | 93% $H_2SO_4$ | 50% NaOH |
| Regenerant capacity (lb/cu ft) | 10 | 2.5 | 7 |
| Amount of regenerant storage | 16 tons | 6,000 gal* | 4,160 gal* |

*This figure includes 2500 gal of $H_2SO_4$ needed for neutralizing the make-up water to the cooling tower.

**10.3.8 Instrumentation, Laboratory, and Housing.** All demineralization, softening, and water-neutralization processes are carried out in the water-treatment building, and all related storage tanks are just outside. A water-analysis laboratory, as well as a water supervisor's office, washroom, lockers, and lavatory are provided.

Instruments are furnished to record the production flow of (1) make-up water to the cooling towers, and (2) blended and demineralized water to the storage tanks.

Level indicators and alarms are provided on the blended-water and demineralized-water tanks. The acid tank has a level indicator, but the caustic tank does not. Normal flow through the cation and anion exchangers, backwash flow, and rinse flow are indicated, and the total flow is recorded. As quality checks on FMA (Free Mineral Acidity) break-through, the *p*H values of the water leaving the cation units are recorded. A final check is provided by recording the *p*H of the water passing to the demineralized-water storage tank. Resistivity of the treated water is also recorded. Sodium and silica break-through require manual periodic tests.

## 10.4 EFFLUENT CONTROL

**10.4.1 Introduction.** The purpose of the effluent control system is to collect, hold, and dispose of both process and nonprocess radioactive fluid wastes. The radioactive fluid wastes expected from the MTR site are generally classified and disposed of as follows:

1. The 50-gpm purge from the process-water system is sent direct to the retention basin for a holdup period of about 90 hr. This holdup time reduces the activity, which is due primarily to $Na^{24}$, from 1200 dis/ml/sec to about 20 dis/ml/sec. The discharge from the retention basin flows to the effluent pond, where it percolates into the earth.
2. Mildly active experimental and laboratory fluid wastes are drained to underground holdup tanks. When the tanks are full, the activity and decay trends of the contents are evaluated. According to the activity, the contents are then sent to the effluent pond, retention basin, or the underground permanent storage tanks.
3. "Hot" fluid wastes are collected locally in shielded containers. The contents are either sent to the underground permanent-storage tanks or to the Chemical Processing Plant for disposal.

**10.4.2 Retention Basin.** *General Description.* The retention basin is a tank for holding flowing radioactive water for a period of time that is long enough to allow the radioactivities of the short-lived water components to decay to tolerable values. The entering water must not mix with the leaving water, and, ideally, there should be no mixing at any intermediate points. The time necessary for decay of $Na^{24}$ radioactivity and the given effluent flow are the main factors determining the size of the basin. Effluent waters from the Process Water Building and the Reactor Building process sump mix in the line going to the retention basin. The 1200 dis/ml/sec activity based on 50 gpm of solution containing 5 ppm of $Na^{23}$ plus a recoil equivalent of 1 ppm requires a 266,000-gal basin to provide the 88-hr holdup necessary for decay to a tolerance value of 18.5 dis/ml/sec. In order to have storage space available for intermittent flushes, future expansion, and also for the 320,000 gal of process water in case it becomes overly contaminated, there are two 360,000-gal basins provided. Thus, one of these is large enough to store the entire contents of the process-water system. During normal operation, one basin is in service while the other is available for the emergency discharge of radioactive process water. If the reactor power level is doubled, the load on the retention basin will not appreciably increase and the two 360,000-gal basins will easily accommodate the effluent flow.

*Shielding.* The roof over the basin is concrete with several feet of earth covering. This reduces radiation directly over it to less than a tolerance value of 50 mr/hr.

*Effluent Flows.* The water flows that the retention basin is required to accommodate are summarized in Table 10.4.A. The normal operating flows are those listed under "Continuous Flows," while the "Intermittent Flows," constitute the flows encountered during periodic reactor operations and during emergencies.

*Size and Structure.* The retention basin has a capacity of 720,000 gal and is made up of two 360,000-gal sections, each about 125 by 20 by 20 ft deep, placed side by side. An inlet trough with adjustable sluice gates distributes the water entering the basin. Outlet weirs measure the discharge into the exit trough. Each of the two sections of the basin can be drained separately and thoroughly, and may be entered for inspection and cleaning. The underground feature of the basin obviates any algae growths.

**TABLE 10.4.A**

**Water Flows to Retention Basin**

| | CONTINUOUS FLOWS (gpm) | INTERMITTENT FLOWS | |
|---|---|---|---|
| | | RATE (gpm) | DURATION (hr) |
| Effluent control | | | |
| Catch tanks* | | 50 | 0.5 |
| Reactor Building process sump | | | |
| Experimental plugs | | 90 | |
| Control rod seal | 2 | | |
| Monitoring tubes | 5 | | |
| Normal sump overflow in water-cooling building | | | |
| Process purge | 50 | | |
| Steam jet condensate | 2 | | |
| Reactor Building canal sump | | | |
| Canal flow* | | 100 | |
| Canal flush | | 400 | 6 |
| Emergency overflow from sump tank in water-cooling building | | | |
| Fuel assembly change | | 200 | 10 |
| Emergency overflow | 2 | 20,000 | 0.25 |

*Normally little or no activity expected here.

*Ventilation.* The operating level of water in the retention basin is fixed at roof level by the overflow weir. Thus there is essentially no air over the water. This design eliminates the accumulation of radioactive argon, and/or an explosive mixture of hydrogen and oxygen. The basin effluent inlet must be ventilated to remove any $A^{41}$ activity. This is done by means of wind-induced turret vanes on 20-ft-high stacks.

*Instrumentation.* The activity of incoming and outgoing effluents is monitored continuously. The activity is measured by a gamma-radiation recorder a few feet above water level at the entrance and exit of the basin. The exit water flow is measured at the discharge weir and recorded.

*Emergency.* In the event of a fission break or high radioactive contamination in the cycling 320,000-gal process-water system, the entire contents of the system can be emptied into the retention basin for decay and holdup until disposal action can be taken. The 130,000-gal contents of the canal can be handled similarly if so desired.

**10.4.3 Effluent Pond.** Since there are no rivers available for water disposal near the MTR site, it is necessary to dispose of the water from the retention basin by allowing it to percolate through the ground, sand, and gravel until it reaches the water table below. A percolation rate of 0.3 cu ft per square foot per day was determined by laboratory tests of ground cores obtained at the MTR site. This figure is exceedingly low and a field test was later made in which the percolation rate was found to be 100 cu ft per square foot per day. Since the higher figure was so different from the smaller one, it was decided from the experimental data that a rate of 1 cu ft per square foot per day would be realistic and conservative for design purposes. Based on this percolation rate and a continuous discharge flow of 179 gpm, an effluent pond 130 by 240 ft, giving an area of approximately 30,000 sq ft, is provided. The sides of the effluent pond basin are sloped at about a 45° angle. This basin is deepest at the water inlet and from there the bottom slopes upward in all directions. The highest point of the basin floor is 4 ft above the water inlets. The water inlets are at the low point of the basin in order to eliminate freezing at the inlets. The basin is enclosed by a fence to prevent animals or unauthorized persons from entering the area. Although little activity is expected to accumulate in the pond, the barriers are furnished as a safety precaution.

**10.4.4 Nonprocess Active Effluents.** In addition to process wastes, the effluent control system must handle nonprocess liquid wastes originating from experimental work or accidental spills in the Reactor Building and laboratory wing. Large quantities of very "hot" wastes are not anticipated, and therefore provisions are only made for the handling of moderate quantities of "warm" wastes.

Since the bulk of the wastes was increased and the methods of subsequent waste treatment were made more difficult by neutralization of acid wastes in the past, the wastes at the MTR site are not neutralized. Thus the storage tanks and handling systems are such that they can withstand acidities of *p*H 2.

The type of wastes expected and the planned disposal of them are discussed briefly in this paragraph. The "hot" wastes from the reactor area will be transferred to the Chemical Processing Plant for disposal. Radioactive material from experimental equipment will be collected in shielded pots and, according to whether it is "hot" or "warm," will be sent to the Chemical Processing Plant or to the permanent-storage tank farm. Alpha particle wastes and slightly active wastes and solvents will be collected in stainless steel drums and, depending on the activity, will be sent either to the Chemical Processing Plant, permanent-storage tank farm, or the underground holdup tanks. It is expected that the largest volumes of wastes will come from accidental spills and laboratory sinks, and that only about 5% of these will be radioactive. The description of the system as given in the following paragraphs refers particularly to this large volume of wastes from spills and laboratory sinks.

The wastes are accumulated in four 1000-gal underground holdup tanks located near the Reactor Building and laboratory wing. Two of these tanks receive effluent from the basement and main floor drains of the Reactor Building via a Reactor Building sump tank and pump; the other two tanks receive radioactive effluent from the sinks in the laboratories. While the contents of one tank of each pair are being monitored, the companion tank is filling. The nonradioactive effluents caught in these tanks are sent direct to the effluent pond. The highly radioactive wastes are pumped to the permanent-storage tank farm. The tank farm, based on a one-year capacity, initially consists of two 10,000-gal underground storage tanks.

Each holdup tank is constructed on a concrete pad that drains to a dry well so that monitoring for leaks is possible. Each pair of tanks is supplied with one self-priming 50-gpm pump located above it so that leakage from the pump drains back to the tanks. The pump and tank piping is such that (1) the tank can be stirred by recirculation, (2) a sample can be taken, or (3) the contents can be delivered to the retention basin, the effluent pond, or the 10,000-gal permanent-storage tanks. The tanks in each pair have overflow lines to each other. Each tank is provided with a level gauge and a high-level alarm.

The water table at the MTR site is about 600 ft below the surface. It is estimated that it takes 150 years for water to percolate through the lava beds under the Snake River Plains back to the Snake River. If this is true, the activity of wastes that leak from the effluent system and reach the Snake River will be greatly reduced.

## 10.5 COOLING TOWERS

**10.5.1 General Description.** A water-cooling tower indirectly dissipates most of the reactor heat load to the atmosphere. The 30,000 kw, or 1,700,000 Btu/min, developed in the reactor active section passes directly to the process water and raises its temperature from 100 to 110°F. The process water yields this acquired heat to the cooling-tower water circulating through the condenser tubes of the flash evaporators. The 200 gpm of flashed process-water vapors is condensed and returned to the process-water stream. The cooling-tower water passes through the cooling towers, collects in the cooling-tower basin, and is again pumped through the flash-evaporator condensers to complete its cycle. Water is provided to the basin to make up for the quantity lost by evaporation and drift and to dilute the concentrating water solids so as to prevent scaling. Provision is made to add chlorine solution intermittently to the circulating cooling-tower water to inhibit algae growth. Dilute acid solution is mixed with the make-up water so as to stabilize the water by neutralizing part of the well-water alkalinity. Up to 100 gpm of the cooling-tower make-up water comes from the canal purge. Strict control of the Langelier saturation index is kept on the circulating cooling tower water to minimize corrosion, scaling, and delignification of the wood. Particular design consideration was given to operation of the cooling towers during icing weather and under heavy dusting conditions.

**10.5.2 Heat Load.** If the air leaving the tower is saturated with water vapor at 95°F and the air entering is saturated with water vapor at 72°F, then 66,500 lb of dry air per minute is needed to handle the heat load as follows:

| TEMPERATURE (°F) | *H* (Btu per lb dry air) | DRY AIR (lb/min) | WATER (lb/min) | OPERATING CONDITIONS | |
|---|---|---|---|---|---|
| | | | | (cfm) | (cu ft/lb) |
| 95 | 60.7 | 66,500 | 2,420 | 1,000,000 | 14.8 |
| 72 | 35.1 | 66,500 | 1,120 | 920,000 | 13.8 |
| 23 | 25.6 | | 1,300 | | |

Following is a summary of the heat data:

| | |
|---|---|
| Latent, 1,300 lb of water | 1,300,000 Btu/min |
| Sensible, 1,300 lb of water | 15,000 |
| Sensible, 66,500 lb dry air | 382,000 |
| Total | 1,697,000 Btu/min |

**10.5.3 Operation and Description of Cooling Towers and Basins.** The cooling-tower operations are based on a 7°F wet-bulb approach to the highest summer wet-bulb temperature (65°F). A medium-height tower of 28 ft requires, according to accepted practice, about 600 sq ft of ground area per 1000 gpm of throughput. Thus, five 30- by 40-ft (ground area) towers provide the 6000 sq ft needed. Dust conditions require especially deep basins for collecting mud and silt, while severe winter conditions require ice defrosting by means of reversible fan gear drives. For a pressure drop of ½ in. of water across the fans and a fan efficiency of 60%, the entire fan air load is 115 hp. The fans have waterproof and dustproof motor drives for outdoor operations. Each cell has an independently drainable basin section which is 4 ft deep and holds about 175,000 gal. Make-up is estimated as 0.1% of the circulated flow times the cooling range, or 210 gpm. The drift losses should be about 0.1 to 0.2% of the circulating flow and thus are estimated as 10 to 20 gpm. The make-up requirements are then about 250 gpm during the summer, plus a 50-gpm basin purge. Normal purge is accomplished by means of an adjustable overflow weir. With a 300-gpm make-up, and assuming 250 gpm is vaporized, a sixfold increase in concentration of dissolved solids in the cooling-tower water will occur.

A brief summary of an analysis of the raw water from the wells is given in the following table:

**MTR Raw Well Water**

| CATIONS | AMOUNT (epm) | ANIONS | AMOUNT (epm) | HARDNESS AS $CaCO_3$ (ppm) | ALKALINITY AS $CaCO_3$ (ppm) | TDS (ppm) |
|---|---|---|---|---|---|---|
| Ca | 2.6 | $HCO_3$ | 3.6 | 205 | 172 | 239 (19 ppm $SiO_2$) |
| Mg | 1.6 | $SO_4$ | 0.4 | | | |
| Na | 0.1 | Cl | 0.3 | | | |
| K | 0.1 | $NO_3$ | 0.1 | | | |
| | 4.4 | | 4.4 | | | |

Partly neutralizing with sulfuric acid or 3.0 epm of $SO_4$ gives the following analysis for the cooling-tower make-up water:

**Acid-neutralized Well Water for Cooling Tower**

| CATIONS | AMOUNT (epm) | ANIONS | AMOUNT (epm) | HARDNESS AS $CaCO_3$ (ppm) | ALKALINITY AS $CaCO_3$ (ppm) | TDS (ppm) |
|---|---|---|---|---|---|---|
| Ca | 2.6 | $HCO_3$ | 0.6 | 205 | 0 | 219 (19 ppm $SiO_2$) |
| Mg | 1.6 | $SO_4$ | 3.4 | | | |
| Na | 0.1 | Cl | 0.3 | | | |
| K | 0.1 | $NO_3$ | 0.1 | | | |
| | 4.4 | | 4.4 | | | |

The 2.6 epm of $CaSO_4$ gives a concentration of about 170 ppm. A sixfold increase would increase this to about 1000 ppm of $CaSO_4$, which is still far below its solubility of about 1700 ppm at 0 °C. $MgSO_4$ is very water soluble and thus it offers no problem with regard to crystallizing out of solution. A sixfold increase in concentration would also give about 1400 ppm of TDS.

**10.5.4 Pumping Station.** The pump house is some distance from the cooling towers so as to facilitate air circulation and minimize the amount of water spraying the pumping station. Acidulation and chlorination control of the cooling-tower make-up water are performed in a mixing chamber in the water-treatment building. Four vertical flooded turbine type pumps, each of 3300 gpm at about a 100-ft head, are used. Three pumps are in continual operation, while one is held as a spare or alternate. It is recommended that the pump-house and cooling-tower-area equipment be inspected hourly, but that no one be stationed there permanently. The inlet ventilating air is filtered and the building is pressurized so as to prevent the infiltration of dust and thereby protect electrical and other equipment.

**10.5.5 Instrumentation.** At the pump house each of the four cooling-water pump motors is provided with an ammeter. A voltmeter is provided across the power supply feeder to these motors. A two-point *p*H recorder in the water-treatment building helps the operator guide the acid-neutralization dosage by recording the *p*H of the entering acidulated make-up water and the circulatory

water streams. Each pump has a pressure gauge. The adjustable weir overflow is calibrated so that the purge flow is indicated visually. At the water-treatment building, the cooling-tower circulatory flow and inlet and outlet water temperatures are recorded. A low water level in the basin is indicated by an alarm in the water-treatment building; a dry-wet bulb two-point outdoor recorder is also located here. When the water level in the basin is adequate, the make-up from the general water reservoir is shut off but the make-up from the canal continues.

**10.5.6 Emergency Operation.** In the event of power failure, the cooling-tower system is shut down completely, and the flow of make-up water is automatically stopped. Make-up water is also shut off automatically if the general water reservoir content is low.

**10.5.7 Design Data.** Following is a summary of the design data:

| | |
|---|---|
| Heat load | 1,700,000 Btu/min or 30,000 kw |
| Maximum wet-bulb temperature | 65°F |
| Approach temperature | 7°F |
| Altitude | 5,000 ft above sea level, 25 in. Hg, 28 ft water |
| Water flow | 9,000 gpm, 100 ft water, 120 hp each pump |
| High temperature | 95°F |
| Basin temperature | 72°F |
| Air flow | 66,500 lb/min, ½ in. water, 110 hp total |
| Dust | Moderate-to-strong "dust devils" |
| Winds | Moderate-to-strong, SW and NE prevailing 2:1 |

ARCO TEMPERATURES (°F)[6]

| 1948 | Jan | Feb | Mar | Apr | May | June | July | Aug | Sept | Oct | Nov | Dec | Annual |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Average max. | 28 | 34 | 43 | 57 | 66 | 73 | 84 | 83 | 73 | 60 | 45 | 31 | 56 |
| Average | 14 | 20 | 31 | 43 | 51 | 56 | 66 | 64 | 55 | 44 | 32 | 19 | 41 |
| Average min. | 10 | 16 | 24 | 31 | 39 | 45 | 52 | 49 | 40 | 31 | 23 | 13 | 31 |

POCATELLO RELATIVE HUMIDITY (%)[6]

| 1948 | Jan | Feb | Mar | Apr | May | June | July | Aug | Sept | Oct | Nov | Dec | Annual |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 8 A.M. | 80 | 79 | 73 | 69 | 67 | 60 | 55 | 54 | 59 | 68 | 73 | 79 | 68 |
| Noon | 71 | 67 | 56 | 45 | 38 | 31 | 27 | 29 | 32 | 47 | 60 | 70 | 48 |
| 8 P.M. | 73 | 67 | 54 | 42 | 37 | 30 | 24 | 25 | 30 | 45 | 60 | 72 | 47 |

ABSOLUTE HUMIDITY (grains water/lb dry air)

| | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 8.5 | 10 | 13 | 20 | 20 | 20 | 26 | 27 | 20 | 22 | 23 | 20 | |

## 10.6 PROCESS RADIATION INSTRUMENTATION AND AREA MONITORING

**10.6.1 Process Radiation Instrumentation.** In detecting the radioactivity of the air and water systems some commercially available instruments are applicable. For those installations in which such instruments prove inadequate, modifications of currently used instruments or entirely new designs are required.

The methods used in detecting radiation give only relative values. Radiochemical analyses are to be used when accurate quantitative determinations are necessary.

*Cooling-air and Contaminated-air Systems.* In the reactor cooling-air system the instrumentation is set up primarily to measure the argon activity and the particulate activity of the air before it is discharged into the atmosphere. The monitoring system extracts air from the main air duct, passes it first through a particulate monitoring device, then through a gas-monitoring chamber, and finally discharges it back into the main air stream.

To detect the particulate activity, a modification of the continuous room-monitoring system used at Brookhaven National Laboratory is used. The instrument consists of a strip of filter paper moving across a series of openings through which air is pumped. The paper, after collecting the dust, passes in front of a mica end-window Geiger counter, where the counts are transmitted

through a scaler into a recording system. The major alterations of the Brookhaven design are (1) elimination of the alpha scintillation counter, (2) addition of an air inlet tube that leads directly to the filter paper strip from the main air duct, and (3) the encasement of the unit in an airtight container.

The air, after passing the dust-monitoring device, is passed through a C.W.S. filter and then into the annular type ionization chamber. The purpose of the second filter is to further ensure dust-free air. The annular type chamber consists of two concentric cylinders and utilizes maximum geometry by allowing the radioactive air or gas to pass through the innermost cylinder while the outer chamber is filled with argon gas at a pressure of 20 atm.

Normally, radioactive gases and particles will be vented through the off-gas system and discharged to the stack. To detect and measure this activity, a similar arrangement as described above for the reactor air system is used.

*Process-water System.* In the water system the annular type ionization chamber is used. By-pass valves and conduits allow the radioactive water or gas to pass through the chamber. Activities are measured by this method at the seal tank, liquid vent seal, and in the demineralized water system.

The measurement of the water activity at the seal tank tends to serve a dual purpose. First, it records the total water activity for the process-water system at approximately 60 sec flow time from the active lattice; secondly, the reading is used as a check on the background activity for the scanning unit which is to detect a fuel plate rupture.

The liquid vent seal evicts gases such as argon, oxygen, neon, and fluorine, which are present or formed in the process water. These gases are radioactive and a measurement of the radiation is required before venting to the atmosphere to serve as a check on the release of activity.

Ionization chambers are placed on the demineralized-water system to detect a possible back-up of radioactive water from the reactor. The back-up of radioactive water may occur through the leakage of valves or the failure of check valves.

Calibration tests performed on the annular type ionization chamber, using radioactive $Na^{24}$ in an aqueous solution, yielded a sensitivity of $1.55 \times 10^2$ curies/cc-amp and $1.12 \times 10^2$ curies/cc-amp for gas pressures of 29.9 atm and

20.75 atm, respectively. Similar tests for the (activated) $Co^{60}$ aqueous solution yielded a sensitivity of $2.22 \times 10^2$ curies/cc-amp at a gas pressure of 20.1 atm. Since $Na^{24}$ will be the main radioactive constituent of the exit process water, the sensitivity of the actual installation should be essentially that indicated above.

*Canal.* The water in the canal may become contaminated with minute radioactive particles from the fuel and beryllium assemblies that are handled and stored in the canal, and possibly from fission products in the event of a fuel assembly rupture. When this water is not contaminated it is pumped to the cooling-tower make-up water. If the water becomes contaminated it is directed to either the sewer or the retention basin, depending on the amount of activity. These measurements are also made with the annular type ionization chamber.

*Retention Basin.* The radiation emitted by the radioactive water in the retention basin is measured by placing G.E. air-wall type chambers above the inlet and outlet troughs.

**10.6.2 Area Monitoring.** Area monitoring will be required wherever the radiation might exceed levels considered to be safe for personnel. In general, monitoring will be required in the following areas: (1) Reactor Building wing, (2) Reactor Building, (3) Process Water Building and (4) Fan House. A variety of types of instruments for area monitoring can be classified according to their respective functions as follows:

| CLASSIFICATION | FUNCTION | RADIATION | INSTRUMENT |
|---|---|---|---|
| *a* | Health physics, personnel monitoring | Gamma<br>Beta<br>Alpha | Hand and foot counters |
| *b* | Survey (portable) | Gamma<br>Gamma | Zeus<br>Zeus with probe, Cutie Pie |
| *c* | Area monitoring (stationary) | Gamma<br>Alpha<br>Beta or gamma | Beckman MX3A or equivalent<br>Air monitoring - air particle |
| *d* | Monitoring of experiments | Fast neutron<br>Slow neutron<br>Alpha and beta | Cheng and Eng<br>Neut<br>Zeuto |

In Table 10.6.A are listed the numbers of each type of instrument to be used in each of the four general areas requiring monitoring. It will be noted that the greatest number of instruments is required for the Reactor Building and wing.

**TABLE 10.6.A**

**Area Radiation Monitoring Instruments**

| TYPE | USE | NUMBER REQUIRED | | | | |
|---|---|---|---|---|---|---|
| | | REACTOR BLDG. | WING BLDG. | PROCESS WATER | FAN HOUSE | TOTAL |
| Hand and foot | Health physics | | 2 | | | 2 |
| Zeus | Survey | 5 | 10 | 2 | 2 | 19 |
| Zeus with probe | Survey | 4 | 2 | | | 6 |
| Cheng and Eng | Experimental | 4 | | | | 4 |
| Air particle | Air monitoring | 4 | | | 1 | 5 |
| Slow neutron Monitron neut | Experimental | 6 | | | | 6 |
| Gamma Detector for stationary installation | Area monitoring | 7 | | 2 | 2 | 11 |
| Alpha | Experimental | 4 | | | | 4 |
| | | | | | | 57 |

The only instruments to which definite locations in the various buildings can be assigned are the area-monitoring instruments, classification "c" above. The locations selected for these instruments have been arbitrarily chosen, and actual survey when the reactor is in operation may result in these locations being changed.

## 10.7 REFERENCE REPORTS


(1) Lennox, D. H., *Retention Basin,* ANL-EKF-11, November 8, 1949.

(2) Lennox, D. H., *Shielding for the Retention Basin,* ANL-EKF-26, December 22, 1949.

(3) Persiani, P. J., *Activity of $H_2O$ as Discharged from Retention Basin,* ANL-EKF-33, February 2, 1950.

(4) Lennox, D. H., *Retention Basin,* ANL-EKF-41, February 27, 1950.

(5) Persiani, P. J., *Activity of the Process Water Effluent,* ANL-EKF-56, July 13, 1950.

(6) Smith, Hinchman, and Grylls, Inc., *Survey Report - Fort Peck, Montana; Pocatello, Idaho Sites,* March 26, 1949.

## 10.8 REFERENCE DRAWINGS

(Blaw-Knox Construction Company Drawings)

| DWG. NO. | TITLE |
|---|---|
| 3150-9-1 | Layout of Boiler House |
| 3150-9-2 | Boiler House - Sections |
| 3150-9-3 | Mezzanine Plan and Sections |
| 3150-902-1 | Electrical, Underground Distribution System |
| 3150-902-5 | Electrical, 2300 Volt - One Line |
| 3150-902-20 | Electrical, Road and Fence Lighting |
| 3150-92-2 | Flow Diagram, Filtered Water System |
| 3150-92-3 | Flow Diagram, Water Treating System |
| 3150-92-3A | Flow Diagram, Water System Demineralizer and Water System |
| 3150-12-3 | Retention Basin, Plan and Section |
| 3150-13-3 | Effluent, Control, Plan and Sections |
| 3150-92-11 | Water Retention System - Flow Diagram |
| 3150-92-12 | Effluent Control - Flow Diagram |
| 3150-7-1 | Cooling Tower and Pump House - Pump House Layout |
| 3150-7-2 | Cooling Tower and Pump House - Cooling Tower Layout |
| 3150-92-5 | Flow Diagram - Cooling Tower |

## Appendix 1

# CRITICALITY STUDIES ON ENRICHED URANIUM—HEAVY WATER SYSTEMS[1]

### A1.1 EXPERIMENTAL ARRANGEMENT

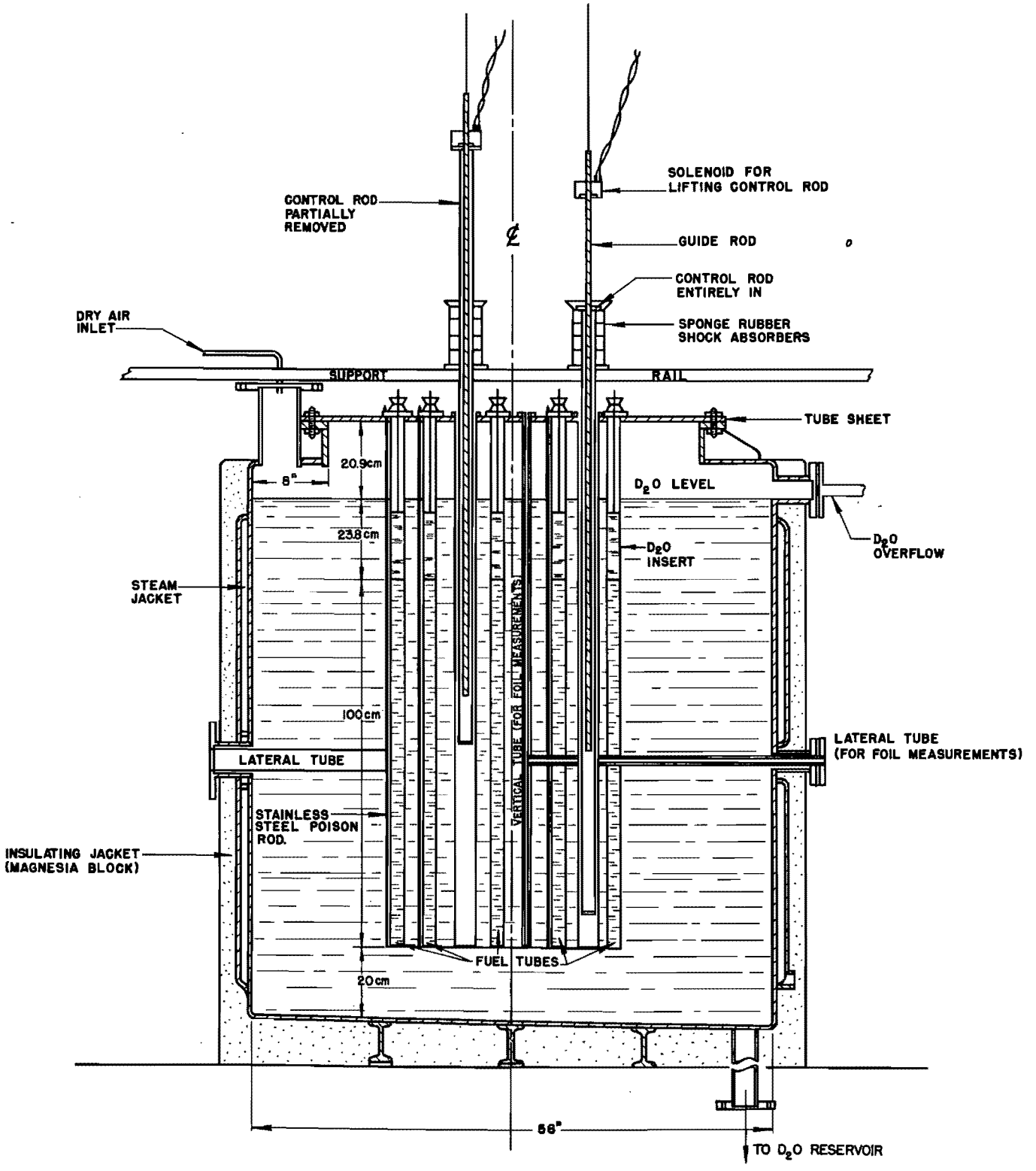
The design and construction of the apparatus for these experiments were carried out by members of the Technical Division (of Clinton Laboratories). The reactor was a vertical welded stainless steel cylindrical tank 140 cm in diameter and built to be filled to a depth of 144 cm with heavy water. The top of the tank was covered by a perforated stainless steel tube sheet which was bolted and gasketed on the rim of the tank proper (Fig. A1.A). "Fuel tubes" containing $U^{235}$ could be lowered through the tube sheet into the heavy water, and thus a lattice could be built up by adding fuel tubes until the assembly became critical. The perforations in the tube sheet and the amount of $U^{235}$ in the fuel tubes were arranged so that if contiguous holes were filled, the mean concentration of $U^{235}$ in the region of the lattice so built up in the heavy water moderator was 10.35 g/liter. Specifically, the holes were in a square lattice of 8.4 cm spacing, and the fuel tubes, containing 45.45 g of $U^{235}$ in the form of a heavy water solution of uranium oxyfluoride ($UO_2F_2$), were filled to a depth of 61.79 cm. Because neutrons have such a large mean free path for diffusion in heavy water, the lattice arrangement can be treated as a homogeneous distribution of $U^{235}$ without large error provided the foreign material in the fuel tubes is considered as well. By omitting alternate holes, it would be possible to obtain a lattice spacing of 16.8 cm or a mean concentration of 2.58 g/liter using the same fuel tubes and tube sheet, and, by using the square lattice at 45° to the foregoing, the mean concentration of 5.17 g/liter could be easily achieved.

Five vertical control and safety rods penetrated the active lattice region through the tube sheet. Each was composed of a sleeve of cadmium in an otherwise empty aluminum thimble which ran down through the heavy water to the bottom of the active region of the lattice. The thimbles and rods could be moved individually as desired. The cadmium rods were supported by electromagnets hung from piano wires, which in turn passed over pulleys on the ceiling to counterweighted controls in a control room. Electrical safety circuits

Drawing # 4805

FIG. A1. A

[redacted]




[redacted]

were so arranged that if either of two Monitrons* near the reactor were to trip, the electromagnets would become deactivated and the cadmium rods would drop into their thimbles upon shock absorbers.

A typical tube sheet lattice arrangement is given in Fig. A1.B. A blacked-in spot on the position of a tube sheet hole means that the hole contained a fuel tube with a full complement of $U^{235}$, viz., 45.45 g for solution depth of 61.8 cm or 73.57 g for a depth of 100 cm. Fractional fuel tubes were also made up in order to permit close bracketing of the critical mass in an assembly. One phase of the experimentation called for the calibration of the control rod position against known amounts of poison distributed in the lattice.

Besides the reactor tank, an elaborate stainless steel system was provided for handling the heavy water. The entire experimental arrangement occupied a two-level concrete cell enclosure so that the heavy water could be raised from a reservoir tank on the lower level into the reactor tank on the upper level by means of dried compressed air. The heavy water could be dropped quickly, if necessary, to stop a chain reaction. Side holes in the reactor tank made possible the insertion of foils to be activated by the neutron flux for flux distribution measurements on a horizontal line through the reactor, and vertical flux measurements were conveniently made in an empty fuel tube hole. Typical horizontal and vertical flux distributions are given in Figs. A1.C and D.

During changes in the lattice arrangement, boron trifluoride counters and Geiger counters with registering circuits and audible signals kept operators informed on reactor activity levels. The progress of the critical reactor was observed by means of an Esterline-Angus recorder driven by the Monitron.

## A1.2 CRITICAL MASS EXPERIMENTS

The main purpose of these measurements was to ascertain the amounts of $U^{235}$ required for criticality as a function of average concentration. The

* A Monitron is an ionization chamber and amplifier device which trips a relay when the ionizing radiation falling upon it exceeds a predetermined level.

Dr 5055

[redacted]

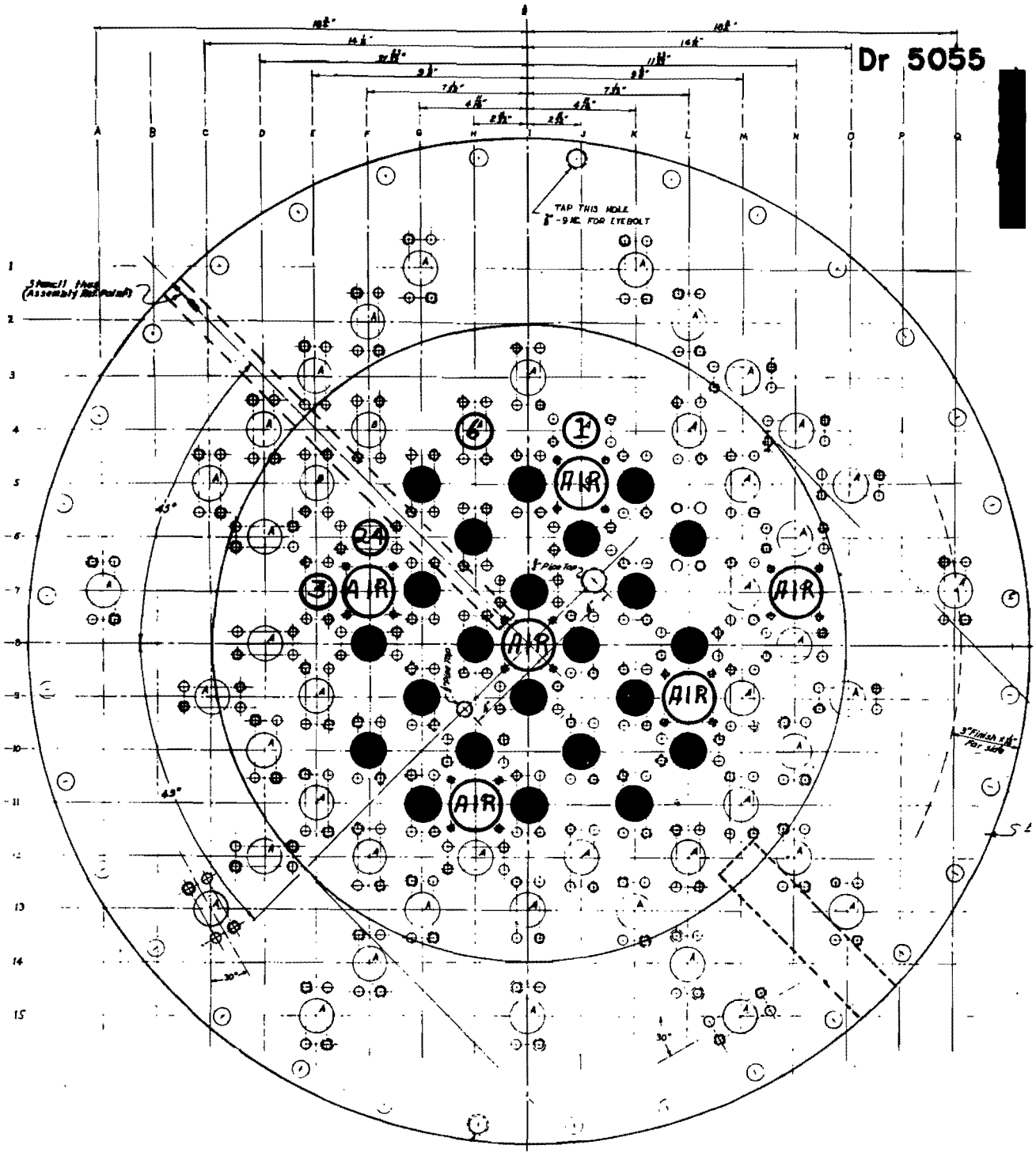



FIG. A1.B h= 61.8 cm.

Critical Mass: 1078 g.

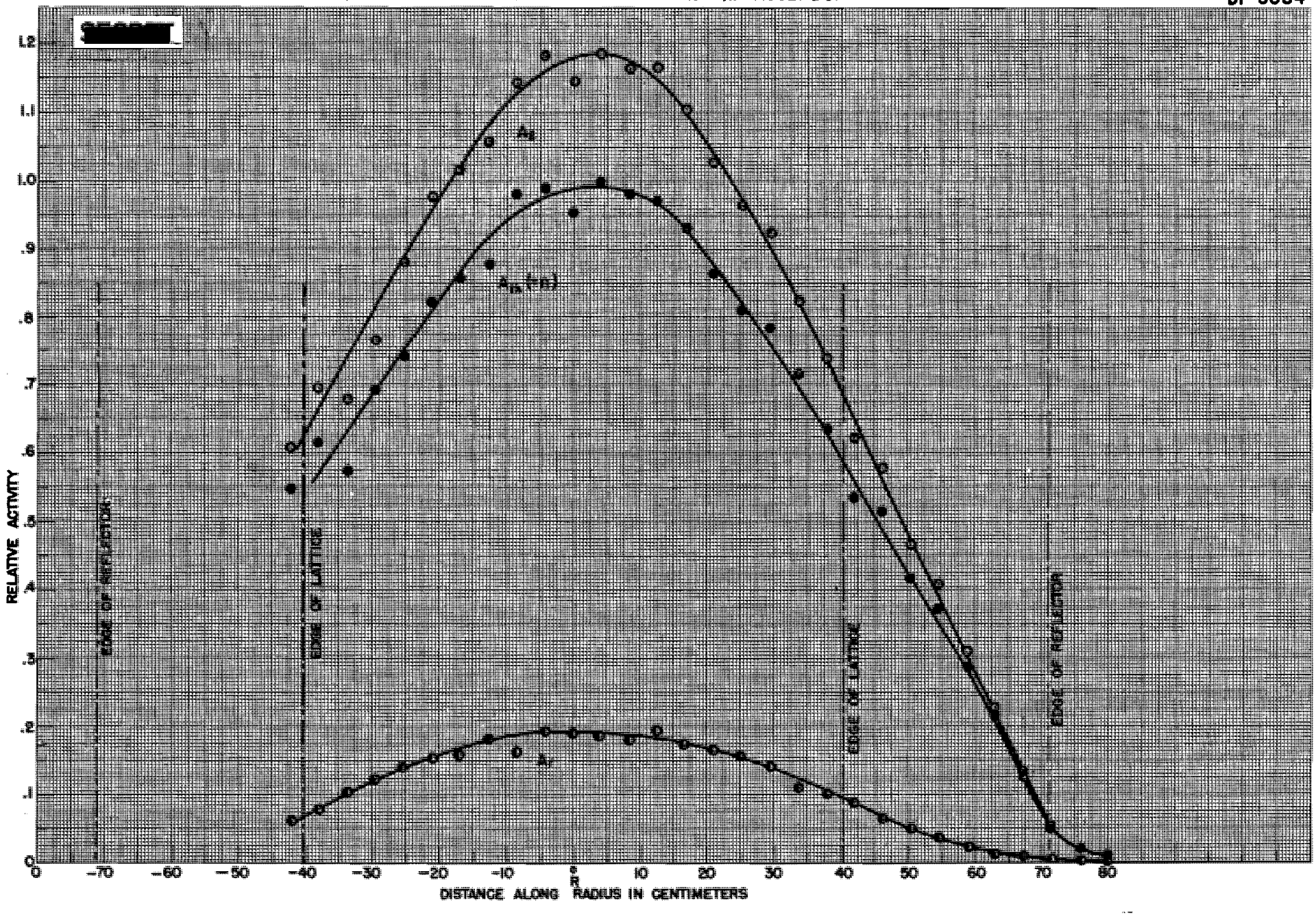
[redacted]



CNL-P821

FIG. A1.C RADIAL NEUTRON DISTRIBUTION IN 2.58 g/l ASSEMBLY

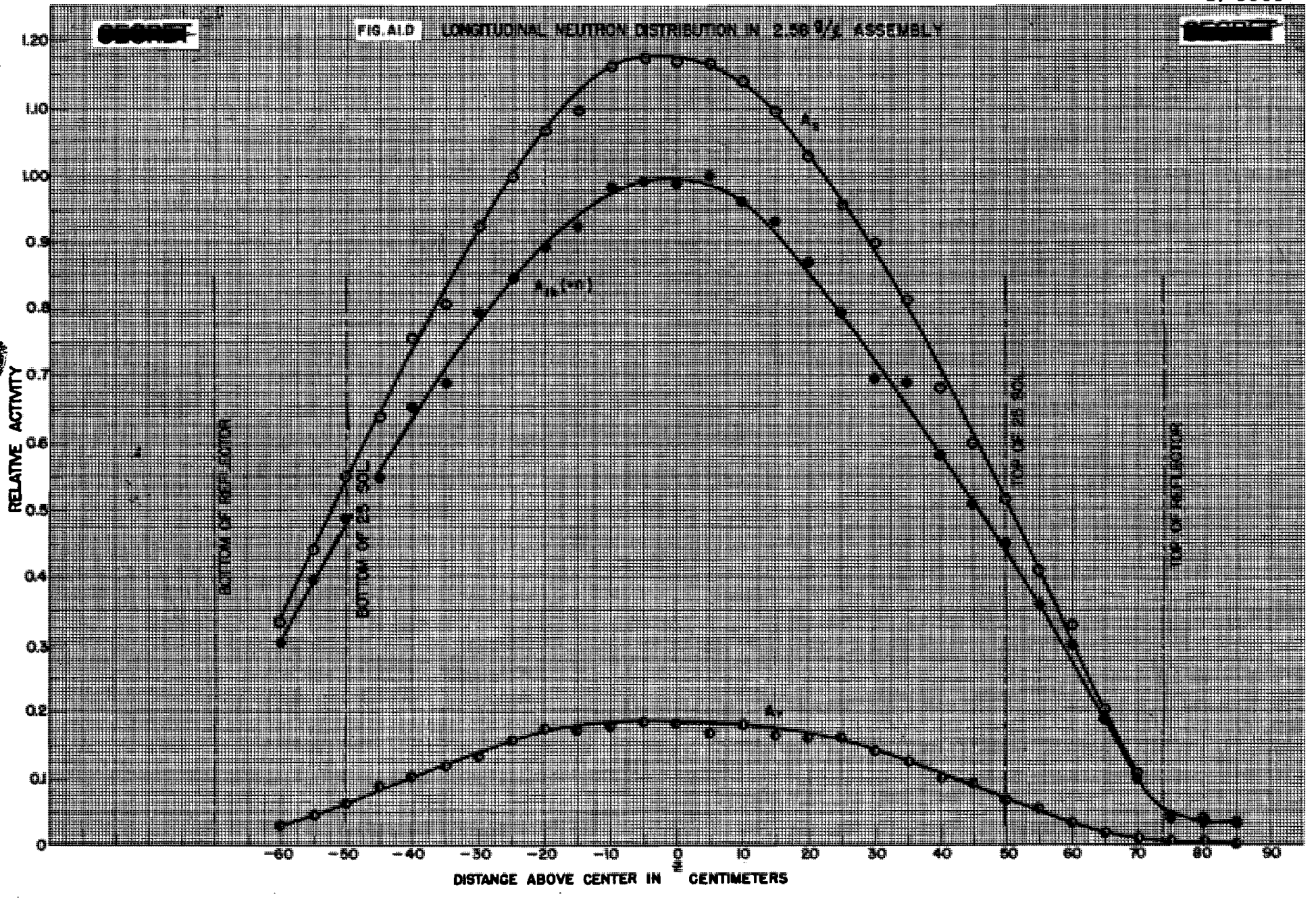
Dr 5084



[redacted]

FIG. A1.D LONGITUDINAL NEUTRON DISTRIBUTION IN 2.56 g/l ASSEMBLY

[redacted]



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initial approach to criticality was made by adding successive fuel tubes to the heavy water tank, meanwhile following the neutron multiplication of a polonium-beryllium source quantitatively by plotting the reciprocal activity, as measured by several detectors, against fissionable mass present. When experience had been gained, it was sufficient to look for a certain amount of multiplication of the source as indicated on the Monitrons, and then to remove the source while the apparatus was in operation and retract the control rod a little more to get the critical condition. The amount of $U^{235}$ was then adjusted so that the reactor was just critical when the cadmium was all out. Table A1.A gives a concise summary of the critical mass data as they were obtained. A general summary of the dimensions and constants of the experiments is given in Table A1.B.

## TABLE A1.A

### Amounts of $U^{235}$ Required for Criticality in Various $D_2O$ Systems

| ITEM | MEAN CONCENTRATION (g/liter) | HEIGHT OF LATTICE, $h_c$ (cm) | TEMPERATURE (°C) | $D_2O$ REFLECTOR THICKNESS (cm) | | | MEAN LATTICE RADIUS, $R$ (cm) | LATTICE VOLUME, $V$ (liters) | CRITICAL MASS OF $U^{235}$ (g) | LATTICE AS IN FIGURE |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | BELOW | ABOVE | LATERAL | | | | |
| 1 | 1 control rod thimble at $r$ = 0; 4 control rod thimbles at $r$ = 18.8 cm; 1 control rod thimble at $r$ = 30.3 cm; 2 radial tubes through reflector; 1 1¼-in.-O.D. radial tube to axis | | | | | | | | | |
| | 10.35 | 61.8 | 20 | 20 | 62 | 47.9 | 23.1 | 104 | 1078 | 5a |
| 2 | 1 control rod thimble at $r$ = 0; 3 control rod thimbles at $r$ = 18.8 cm; 1 thimble at $r$ = 30.3 cm; 2 radial tubes as in Item 1 | | | | | | | | | |
| | 10.35 | 61.8 | 20 | 20 | 62 | 48.2 | 22.8 | 101 | 1046 | 5b |
| 3 | 1 control rod thimble at $r$ = 0; 2 control rod thimbles at $r$ = 18.8 cm; 1 thimble at $r$ = 30.3 cm; 2 radial tubes as in Item 1 | | | | | | | | | |
| | 10.35 | 61.8 | 20 | 20 | 62 | 48.6 | 22.4 | 97.5 | 1009 | 5c |
| 4 | 2 control rod thimbles at $r$ = 18.8 cm; 1 thimble at $r$ = 30.3 cm; 2 radial tubes as in Item 1 | | | | | | | | | |
| | 10.35 | 61.8 | 20 | 20 | 62 | 49.2 | 21.8 | 92.7 | 959 | 5d |
| 5 | 1 control rod thimble at $r$ = 18.8 cm; 1 thimble at $r$ = 30.3 cm; 2 radial tubes as in Item 1 | | | | | | | | | |
| | 10.35 | 61.8 | 20 | 20 | 62 | 49.6 | 21.4 | 89.3 | 924 | 6a |
| 6 | 1 control rod thimble at $r$ = 18.8 cm; 2 radial tubes as in Item 1 | | | | | | | | | |
| | 10.35 | 61.8 | 20 | 20 | 62 | 49.7 | 21.3 | 87.8 | 909 | 6b |
| 7 | 1 control rod thimble at $r$ = 18.8 cm | | | | | | | | | |
| | 10.35 | 61.8 | 20 | 20 | 62 | 49.8 | 21.2 | 87.3 | 904 | 6c |
| 8 | No-hole extrapolation | | | | | | | | | |
| | 10.35 | 61.8 | 20 | 20 | 62 | 50.2 | 20.8 | 84.0 | 869 | |
| 9 | 1 control rod thimble at $r$ = 18.8 cm; no other holes of consequence | | | | | | | | | |
| | 5.17 | 61.8 | 20.8 | 20 | 62 | 40.1 | 30.9 | 185 | 954.5 | 6d |
| 10 | No-hole extrapolation (estimated) | | | | | | | | | |
| | 5.17 | 61.8 | 20.8 | 20 | 62 | (40.6) | (30.4) | (180) | (930) | |
| 11 | 2 control rod thimbles at $r$ = 18.8 cm | | | | | | | | | |
| | 2.58 | 100 | 22.7 | 20 | 24 | 29.0 | 42.0 | 552 | 1425 | 7a |

TABLE A1.A (Cont'd)

| ITEM | MEAN CONCENTRATION (g/liter) | HEIGHT OF LATTICE, $h_o$ (cm) | TEMPERATURE (°C) | $D_2O$ REFLECTOR THICKNESS (cm) | | | MEAN LATTICE RADIUS, $R$ (cm) | LATTICE VOLUME, $V$ (liters) | CRITICAL MASS OF $U^{235}$ (g) | LATTICE AS IN FIGURE |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | BELOW | ABOVE | LATERAL | | | | |
| 12 | 1 control rod thimble at $r$ = 18.8 cm | | | | | | | | | |
| | 2.58 | 100 | 22.7 | 20 | 24 | 29.8 | 41.2 | 532 | 1374 | 7b |
| 13 | No-hole extrapolation | | | | | | | | | |
| | 2.58 | 100 | 22.7 | 20 | 24 | 30.6 | 40.4 | 512 | 1323 | - |
| 14 | 2 control rod thimbles at $r$ = 18.8 cm; 1 control rod thimble at $r$ = 30.3 cm | | | | | | | | | |
| | 10.35 | 100 | 23.5 | 20 | 24 | 52.4 | 18.6 | 109 | 1126 | 7c |
| 15 | 2 control rod thimbles at $r$ = 18.8 cm | | | | | | | | | |
| | 5.17 | 100 | 23.5 | 20 | 24 | 45.0 | 26.0 | 212 | 1098 | 7d |
| 16 | 2 control rod thimbles at $r$ = 18.8 cm; circle of iron at $r$ = 51.9 cm | | | | | | | | | |
| | 5.17 | 100 | 21.6 | 20 | 24 | 41.6 | 29.4 | 272 | 1404 | 8a |
| 17 | 1 control rod thimble at $r$ = 18.8 cm | | | | | | | | | |
| | 2.86 | 61.8 | 21.2 | 20 | 62 | 20.7 | 50.3 | 491 | 1405 | 8b |

## TABLE A1.B

### Collected Dimensions and Constants of the Experiments

| | |
|---|---|
| $U^{235}$ | Average isotopic concentration = 95.4% $U^{235}$ |
| $D_2O$ | Average isotopic concentration = 99.7% D |
| Mean concentration of $U^{235}$ | (*a*) 2.58 g of $U^{235}$ per liter<br>(*b*) 5.17 g of $U^{235}$ per liter<br>(*c*) 10.35 g of $U^{235}$ per liter |
| Constants of theory | $f = 0.963$; $L^2 = 90.42$ cm$^2$; $k = 2.09$ |
| Fuel tubes | O.D. = 1.500 in.; 2S aluminum; wall thickness = 0.035 in.; bottom thickness = 1/8 in.; length below top of tube sheet = 4 ft 9¾ in.; thermal-neutron cross-section = 0.89 cm$^2$ per tube for solution depth of 61.8 cm |
| Control rod thimbles | 2S aluminum; 2-3/8 in. O.D.; 1/8-in. wall; 1/8-in. bottom; thermal-neutron cross-section = 4.62 cm$^2$ |
| Control rods | O.D. = 1¾ in.; type 304 stainless steel No. 16 BWG; length 6¾ in. overall; 0.020-in. Cd sleeve cemented inside |
| $UO_2F_2 + D_2O$ solution | Height in fuel rods (*a*) 61.8 cm, (*b*) 100 cm; amount of fuel per tube (*a*) 45.45 g, (*b*) 73.57 g; thermal-neutron cross-section = 74.5 cm$^2$ per tube for 61.8 cm solution height |
| Reactor tank | 142 cm I.D.; ¼-in. wall; type 304 stainless steel |
| Tube inserts | 1.25-in. O.D.; wall thickness = 0.035 in.; 24ST aluminum; bottom thickness = 1/8 in.; overall length = 10 in.; about 9 in. filled with $D_2O$ |
| Poison rods | Length below top of tube sheets = 4 ft 8¾ in.; type 304 stainless steel; diameters = 3/16 and 1/4 in.; thermal-neutron cross-section = 0.0273 cm$^2$ per gram, giving for 61.8 cm depth, 2.58 cm$^2$ and 4.58 cm$^2$ per rod for 3/16- and 1/4-in. rods, respectively, and 4.18 cm$^2$ and 7.41 cm$^2$ for 100 cm depth, respectively, for the rods |

| Square lattice spacing | 2.59 g/liter | 5.17 g/liter | 10.35 g/liter |
|---|---|---|---|
| | 16.86 cm | 11.92 cm | 8.43 cm |

| | |
|---|---|
| Iron rods simulating thorium | Armco iron; diameter 3.65 cm; length overall = 139.7 cm; spaced 14.8 cm, center to center on circle 51.9 cm in radius |
| Thorium insert in iron rod | Diameter 4.00 cm; length 7½ in; centrally located in iron rod |

Total neutron absorption cross-section of materials in active volume of assembly

(1) For depth of 61.8 cm:

| | |
|---|---|
| Vertical experimental tube (Al) | 0.63 cm$^2$ |
| Radial experimental tube (Al) | 2.61 cm$^2$ |
| Control rod thimble (Al) | 4.62 cm$^2$ |
| Fuel Tubes (Al), each | 0.89 cm$^2$ |
| Solution, per fuel tube | 74.53 cm$^2$ |

## TABLE A1.B (Cont'd)

Thorium insert in iron rod (cont'd) (2) For depth of 100 cm:

| | | |
|---|---|---|
| Vertical experimental tubes | | 1.02 $cm^2$ |
| Vertical experimental tubes, | small | 0.08 $cm^2$ |
| | large | 1.64 $cm^2$ |
| Control rod thimble | | 7.48 $cm^2$ |
| Fuel tubes, each | | 1.44 $cm^2$ |
| Solution, per fuel tube | | 120.6 $cm^2$ |

## A1.3 REFERENCE REPORT

(1) Snell, A. H., *Critical Studies on Enriched Uranium—Heavy Water Studies*, MonP-454, December 15, 1947.

## Appendix 2

# WORK ON CRITICAL EXPERIMENTS, JULY 1946 TO DECEMBER 1948

### A2.1 CRITICAL EXPERIMENTS WITH Al + $H_2O$ MODERATOR AND HEAVY WATER, LIGHT WATER, AND BERYLLIUM REFLECTORS(1)

The experiments with Al + $H_2O$ as a moderator and heavy water, $D_2O$, as reflector were a direct continuation of those initially undertaken and described in Appendix 1 relevant to the "heterogeneous" reactor design, p. A2.1. While the decision for a beryllium reflector had been made, a sufficient quantity could not be available soon enough to justify the removal of the heavy water system. Consequently a broad experimental program on enriched small reactors was initiated to include reflectors of ordinary water, heavy water, pure beryllium, and beryllium plus a small percentage of ordinary water. The last case was in anticipation of cooling water in a beryllium reflector for the MTR.

The first group of experiments in the enriched reactor program, beginning in the summer of 1946, involved the heavy water reflector, and it is these which are now to be described.

**A2.1.1 Experimental Facilities.** Some extensive changes were made in the physical arrangement of the equipment described in Appendix 1 which increased safety, ease of operation, and control of the critical assembly. The general arrangement is sketched in Fig. A2.A. Not shown is an exit from the control room to the outside as an emergency exit and security gate.

With respect to instrumentation, new and faster Monitrons (ion chambers and associated circuits) were added so that three were in operation at all times and all were connected to the control rod safety system. The time constants of the safety circuits were about 0.2 to 0.5 sec. Esterline-Angus recorders installed in the control room were driven by two of the Monitrons so that visual monitoring of the pile was provided. In addition, the usual array of counters, gamma ray and neutron, were in use with mechanical recorders for counting rate purposes and loud speakers for continuous auditory monitoring.

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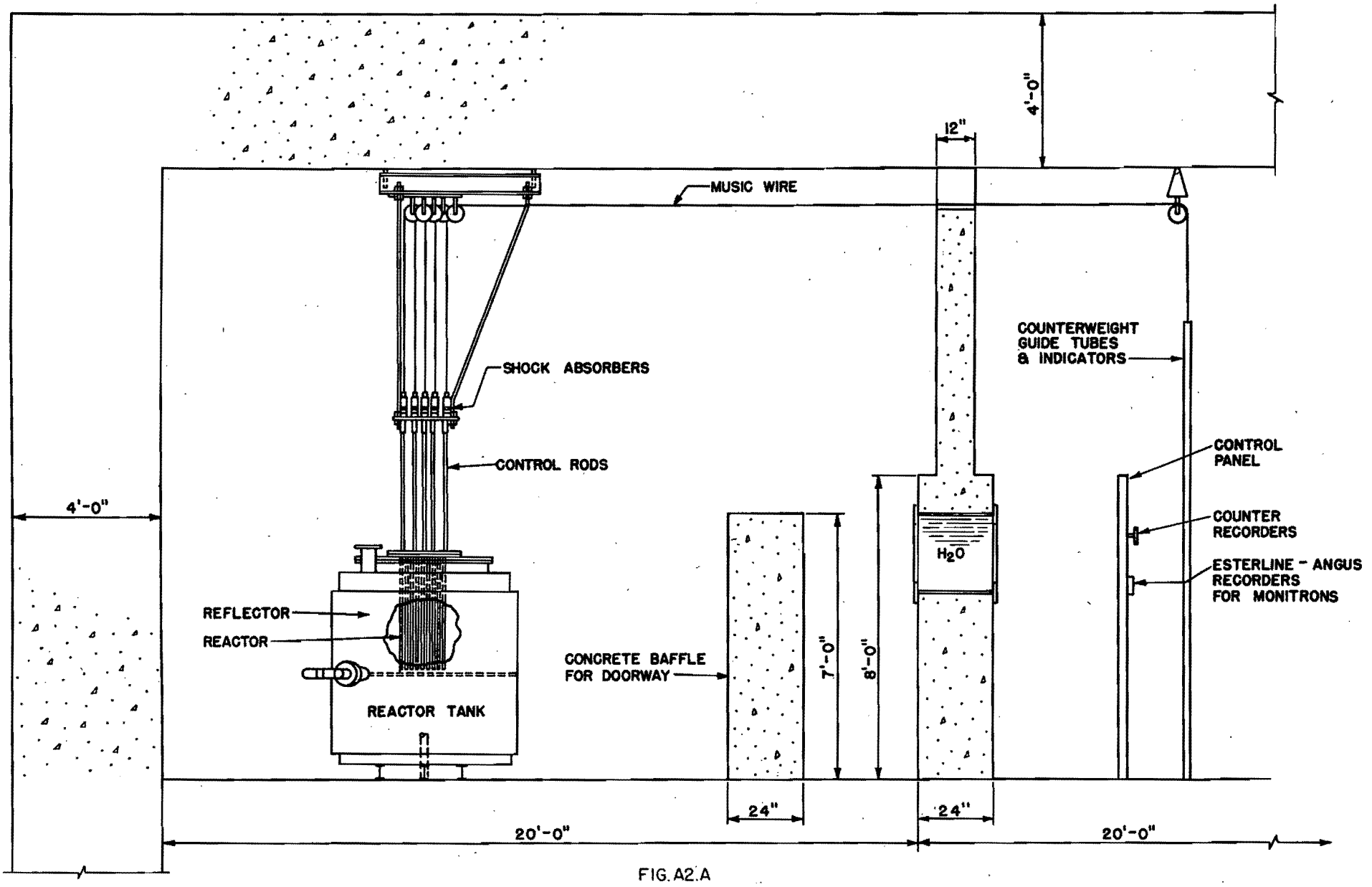


FIG. A2.A

SECTION THROUGH ASSEMBLY & CONTROL ROOMS

A2.2

~~SECRET~~

**A2.1.2 Pile Geometry and Design.** In the design of the high-flux reactor going on during 1946 and 1947 the core would contain a considerable quantity of aluminum in alloy with uranium or otherwise in structural form. It therefore became desirable to study, in the critical experiments, the effect of various volume ratios of aluminum to light water on the fluxes and critical masses. The new high-flux reactor design called for such a close spacing of fuel plates, that, to a very good approximation, the reactor core could be treated as a homogeneous mixture of uranium, aluminum, and water. Consequently, experiments were undertaken in which the concentration of $U^{235}$ was held constant while the volume ratio of aluminum to water was varied for the reasons that (1) engineering design depended upon the allowable quantities of aluminum in the reactor, and (2) the dependence of critical mass, fast flux to slow flux ratio, and neutron "age" on the $Al/H_2O$ ratio is of both theoretical and practical interest. The $Al/H_2O$ ratios studied were 0.88, 0.76, 0.65, and 0.303.

The problem of constructing and assembling a reactor that was essentially homogeneous in reasonably small integral parts was solved by the use of square aluminum tubes, each of which contained about 20.3 g of $U^{235}$ in the form of $UO_2F_2$ in light water solution. A square tube satisfies the requirement of close packing and approximation to homogeneity, and a reasonable size was available, namely 1 by 1 in. with 1/16-in. wall thickness.

The pertinent physical data for the $D_2O$ reflector experiment were as follows:

Concentration of $U^{235}$ = 35.6 g per liter of reactor volume
Height of active portion = 70 cm
Size of individual fuel tube = 2.54 by 2.54 by 90 cm
$U^{235}$ Content of fuel tube = 20.3 g per tube
$Al/H_2O$ volume ratios attainable = 0.88, 0.76, 0.65, 0.303

The actual pile geometry utilized was cylindrical with no reflection above and below the core but $D_2O$ reflection on the cylindrical side. For this purpose a concentric inner tank was built into the $D_2O$ tank described in Appendix 1, p. A1.1. The final arrangement is given in Fig. A2.B.

The procedure during assembly was as follows: (1) the reactor tank has a full complement of tubes at all times; (2) some tubes initially are fuel

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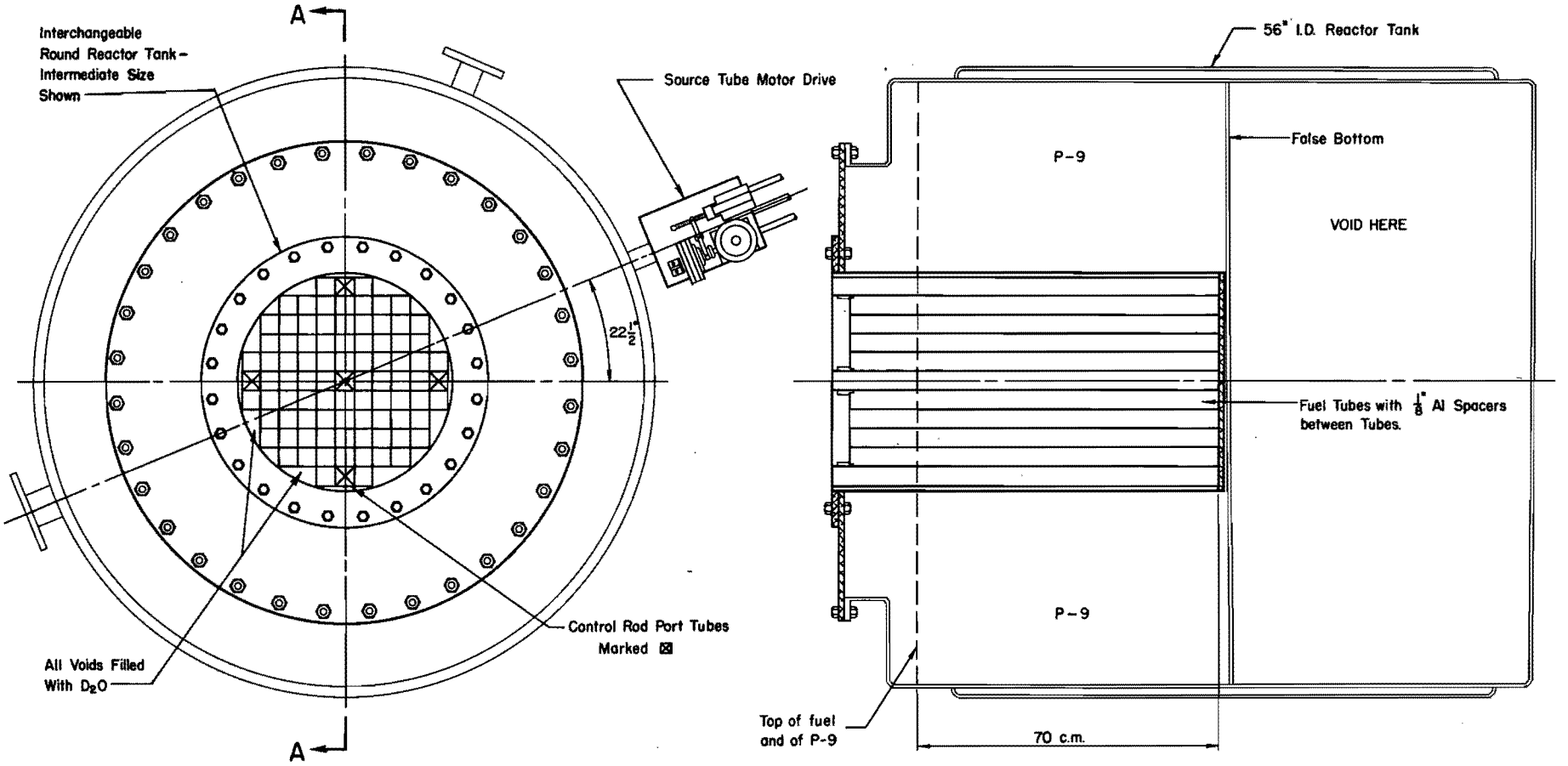


PLAN

SECTION A-A

FIG. A2.B

# ASSEMBLY FOR $D_2O$ REFLECTOR

~~SECRET~~

tubes and the rest are $D_2O$; and (3) the gaps between the square tubes and the cylindrical tank are filled by $D_2O$ introduced into the core tank for this purpose, i.e., to prevent a gap between core and reflector. However, the rather large percentage of aluminum in the reflector region near the core tank essentially required a three-region theoretical treatment of the experiment.

All the above statements relating to the $D_2O$ experiment apply equally well to the arrangement and procedure used in the case of the $H_2O$ reflector. The experiments in which $H_2O$ was employed as reflector were carried out for two reasons: (1) the results would provide information regarding the effect of the water that would be present on top and bottom of the high-flux pile, and (2) the study of a pile in which both the moderator and the reflector are of the same substance is of considerable interest to theoreticians.

When beryllium was used for the reflector the experiment was somewhat "cleaner" with a square arrangement in that it was possible to do away with the irregular region between reactor and reflector. This will be evident from Fig. A2.C. It turned out that the beryllium-reflected pile became critical just four fuel tubes short of a nine-tube square, and the reactor was built out to completely fill this square geometry, the excess $k$ being taken up by poison. Figure A2.D is a photograph looking down on the beryllium assembly.

**A2.1.3 Results and Discussion.** Experiments were performed with $D_2O$, beryllium, and $H_2O$ reflectors to find (1) critical mass, (2) neutron distributions, and (3) values of control rods, cavities, etc., in terms of fuel. Various ratios of Al to $H_2O$ were used. These experiments served as a check on the theoretical calculations of critical masses under the same conditions. The values for the critical mass found in both theory and experiment are compared in Table A2.A.

The situation in spatial neutron distribution is shown in Fig. A2.E, which contains both experimental and theoretical radial distributions. These distributions were measured at a height of 35 cm above the bottom of the pile.

It will be noted that in general the observed densities of both thermal and epithermal neutrons in the reflector are somewhat greater than expected from a two-group calculation. The rise in thermal-neutron density just outside the reactor is quite pronounced, the maximum experimental value being 40% higher than the minimum, whereas the rise for a two-group picture is 15%.

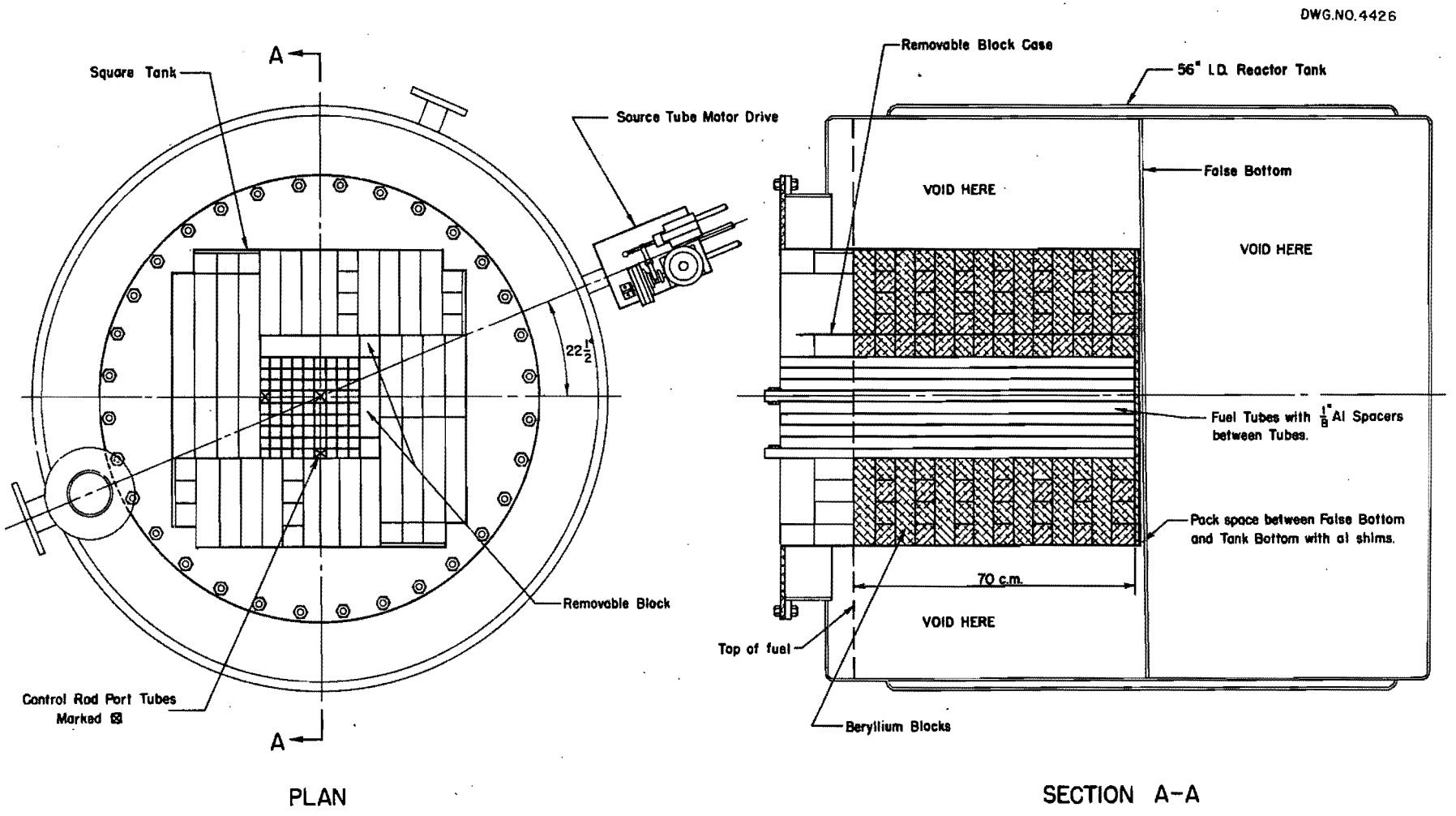


FIG.A2.C

ASSEMBLY FOR
BERYLLIUM REFLECTOR

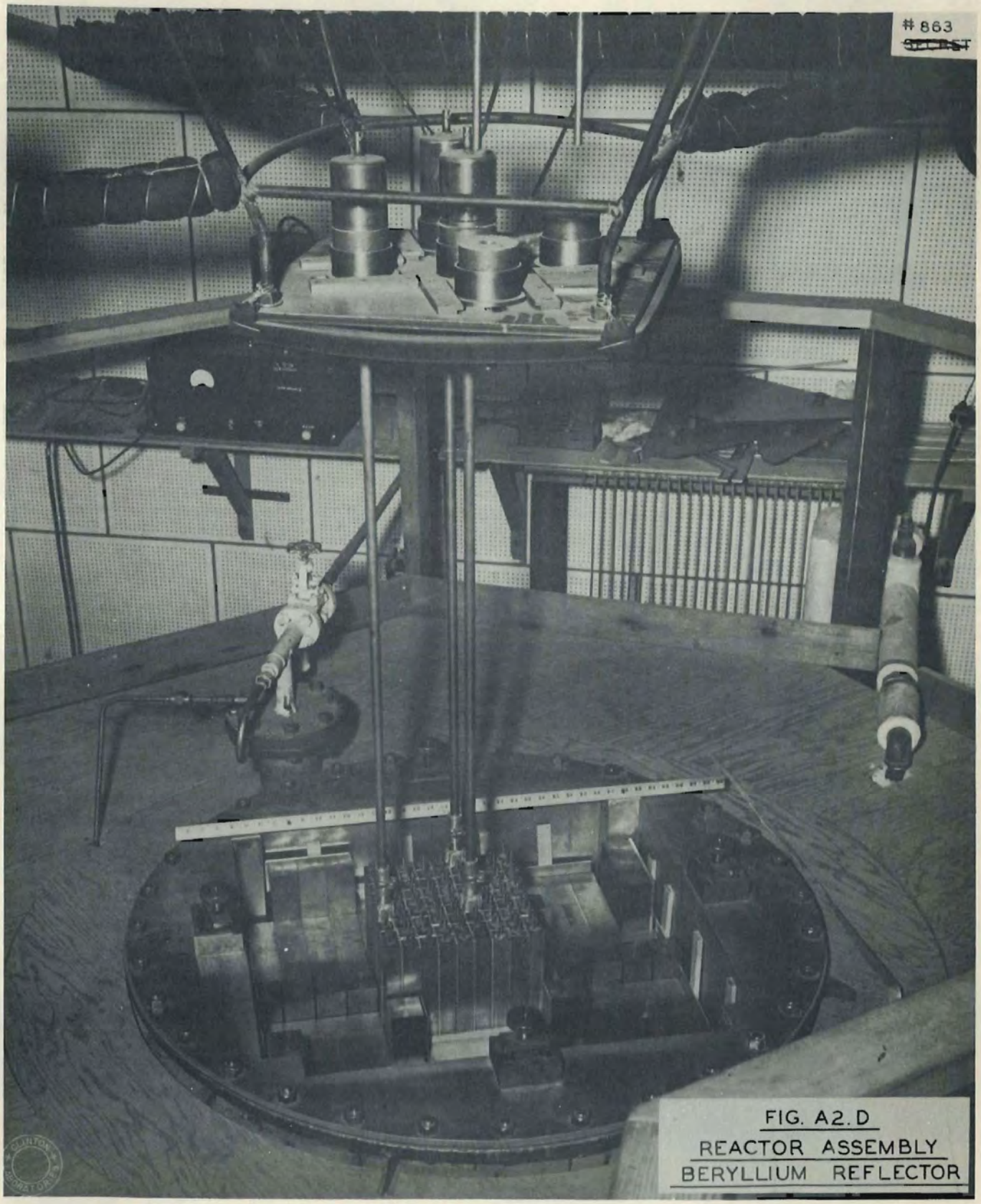


FIG. A2.D
REACTOR ASSEMBLY
BERYLLIUM REFLECTOR

FIG. A2.E

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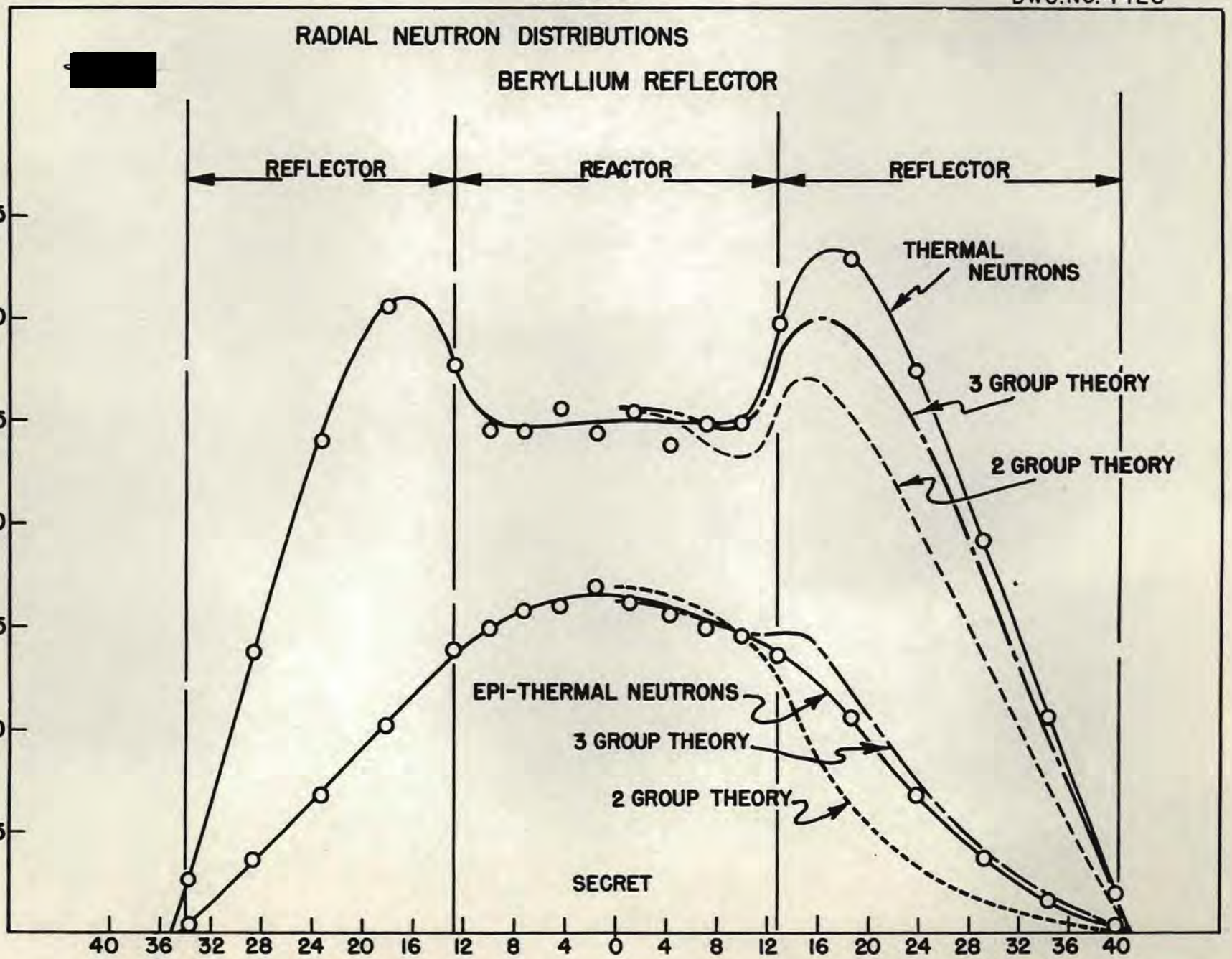


A2.8

**TABLE A2.A**

**Critical Masses in Kilograms of $U^{235}$**

| REFLECTOR | VOLUME RATIO $Al/H_2O$ | OBSERVED | | CALCULATED, 2-GROUP | |
|---|---|---|---|---|---|
| | | GROSS | CORRECTED FOR EMPTY CONTROL ROD PORTS, ETC. | 2-REGION | 3-REGION |
| $D_2O$ | 0.88 | 2.48 | 2.40 | 2.10 | 2.64 |
| $D_2O$ | 0.76 | 2.23 | 2.15 | | 2.43 |
| Be | 0.76 | 1.54 | 1.42 | 1.54* | |
| $H_2O$ | 0.76 | 3.29 | 3.22 | | 3.34 |
| $H_2O$ | 0.66 | 2.82 | 2.72 | | 3.04 |

*Where critical $K = 0.988 \times k$ of pile without poison.

One might be tempted to conclude that beryllium exhibits advantages as a reflector which were unsuspected, but it seems probable that the experiment merely emphasizes known shortcomings of a two-group theory. A three-group calculation is shown in Fig. A2.E. The source of the remaining discrepancy between theory and experiment was not yet clear. Age measurements in beryllium were in progress at Argonne.

Vertical neutron distributions were studied also; typical results will be found in Fig. A2.F. It will be recalled that no reflector was present at top or bottom of the experimental pile, and the solid lines are cosine curves fitted to points near the center by a least-squares method. The slight lateral shift of the data from center is apparently due to the presence of a portion of a control rod in the top of the pile. It is clear that the extrapolation distances for thermal and epithermal neutrons are the same, within the experimental limits.

A few rough measurements of the values of control rods were made. In the beryllium case, the rod near center was found equal to 160 g of $U^{235}$ and a cavity (empty tube) approximately halfway from the center to the periphery was equal to 18 g. On this basis the center cavity equal 35 g.

FIG. A2.F

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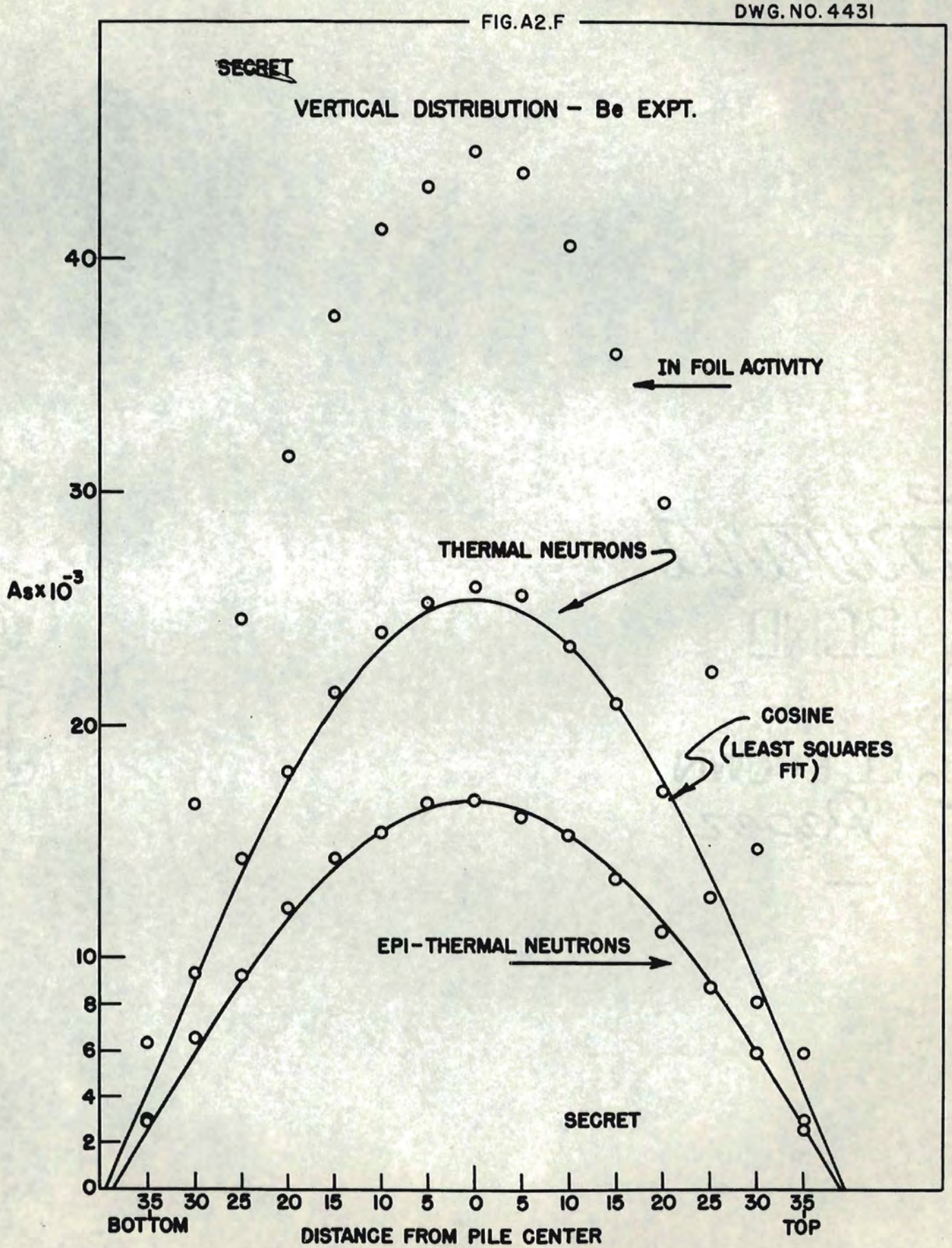

## A2.2 FURTHER CRITICAL EXPERIMENTS ON ENRICHED REACTORS WITH Al + $H_2O$ MODERATOR AND SOLID BERYLLIUM REFLECTOR[2]

With the termination of the experiments dealing with liquid reflectors and in preparation for a series of experiments on reactors having solid reflectors, it was decided that it would be worth while to make certain major changes in the experimental arrangement in the criticality cell in order to provide for greater flexibility and greater convenience in stacking solid reflectors. The entire liquid storage and piping system was removed from the cells, and in place of the tank assemblies which had been used for $D_2O$ and $H_2O$ reflectors, a low platform framework of steel was erected. A ½-in. plate of 2S aluminum was placed over the steel framework to serve as a floor on which to build critical assemblies. The center portion of the plate was cut out, and flanges were provided to hold a center plate of ½-in. aluminum, through which was drilled a rectangular array of holes accurately spaced to position the lower ends of the fuel tubes for $Al/H_2O$ ratios equal to or greater than 0.65. Thus the bottom of the reactor was relatively clean, the core and a considerable portion of the reflector having only the ¼-in. perforated aluminum plate immediately under them.

**A2.2.1 Experimental Facilities.** Figure A2.G shows a core of square cross-section with the beryllium reflector assembled on the platform. The level of the fuel was at the same height as the top of the reflector in these assemblies, the remaining upper portion of the fuel tubes being empty up to the rubber stoppers.

Practically all the radiation detection and monitoring instruments were replaced by improved circuits of more recent design. In particular, it was found that the vibrating-reed Monitrons, two of which can be seen in Fig. A2.G, are a decided improvement over the older designs in stability, sensitivity, and response time.

The operation of the control rod system was improved by replacing the hand-operated control rod counterweights and clamps by small gear boxes (one for each control and safety rod), which were designed to limit the maximum rate of withdrawal of the rods. While the operation of the rods was still basically manual, the use of the gear boxes ensured a reasonably slow and constant rate of withdrawal of the rod and also facilitated changing the control rod position in small increments with accuracy.

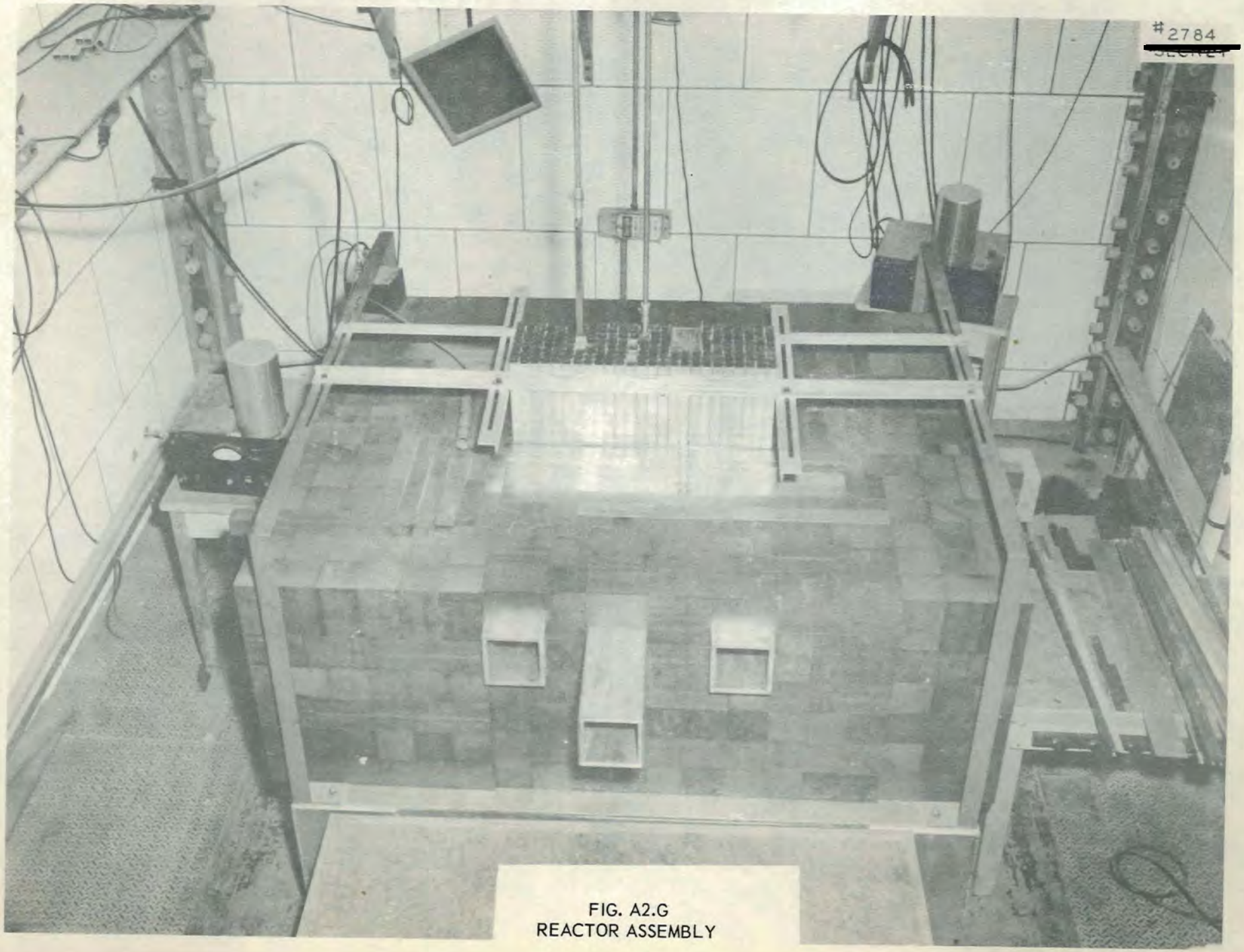

FIG. A2.G
REACTOR ASSEMBLY

**A2.2.2 Experimental Results.** *Bare Pile.* While waiting for the arrival of sufficient beryllium to continue the studies for the high-flux pile, it was decided to try to obtain a figure for the bare critical size as a check on the fundamental characteristics of this reactor. A bare assembly was built with cadmium sheets around the sides of the reactor to reduce the effect of nearby objects, and, although criticality was not finally reached, a multiplication of about 11.6 was obtained at a load of 5050 g. The extrapolation of the reciprocal counting rate versus mass curve (Fig. A2.H) indicates a critical mass of about 5.5 kg. After correcting for the presence of holes for control rods, it was estimated that the bare critical mass was 5.3 ± 0.1 kg. The $Al/H_2O$ volume ratio in this case was 0.65, the height of the core was 66 cm, the maximum loading attained was 48 by 43 cm in area, approximating square geometry.

*Beryllium-reflected Reactors.* The description will be limited to a discussion of the experiments which have been done on relatively "clean" beryllium assemblies, i.e., assemblies which have simple, calculable geometries.

The first experiment in this series on beryllium-reflected reactors was that on the measurement of the critical mass of a square assembly having an $Al/H_2O$ volume ratio in the core of 0.65. In this experiment the geometry of sources and $BF_3$ counters was apparently very good since the counting rate versus mass plots turned out to be approximately straight lines, showing no tendency to either turn downward or curve upward near the critical mass, as happens in many cases. Two Po-Be sources were used, one ($10^7$ neutrons per second) located directly beneath the lower surface of the reactor below the center of the core, and another ($10^5$ neutrons per second) suspended in an empty tube at the core reflector boundary at mid-height of the reactor. Two $BF_3$ proportional counters were used to observe the multiplication of the reactor during the build-up to criticality. These counters were located in such a manner that a large portion of the core lay between the sources and the counters, i.e., the counters must "look through" the multiplying medium to "see" the sources. One counter was placed on the top surface of the reflector in the immediate vicinity of the core (Fig. A2.G), the other was placed about 15 ft from the pile at an elevation about 4 ft higher than the top surface of the assembly.

It was found that the square assembly became critical just three fuel tubes short of an 8- by 8-tube array. In order to complete the square geometry,

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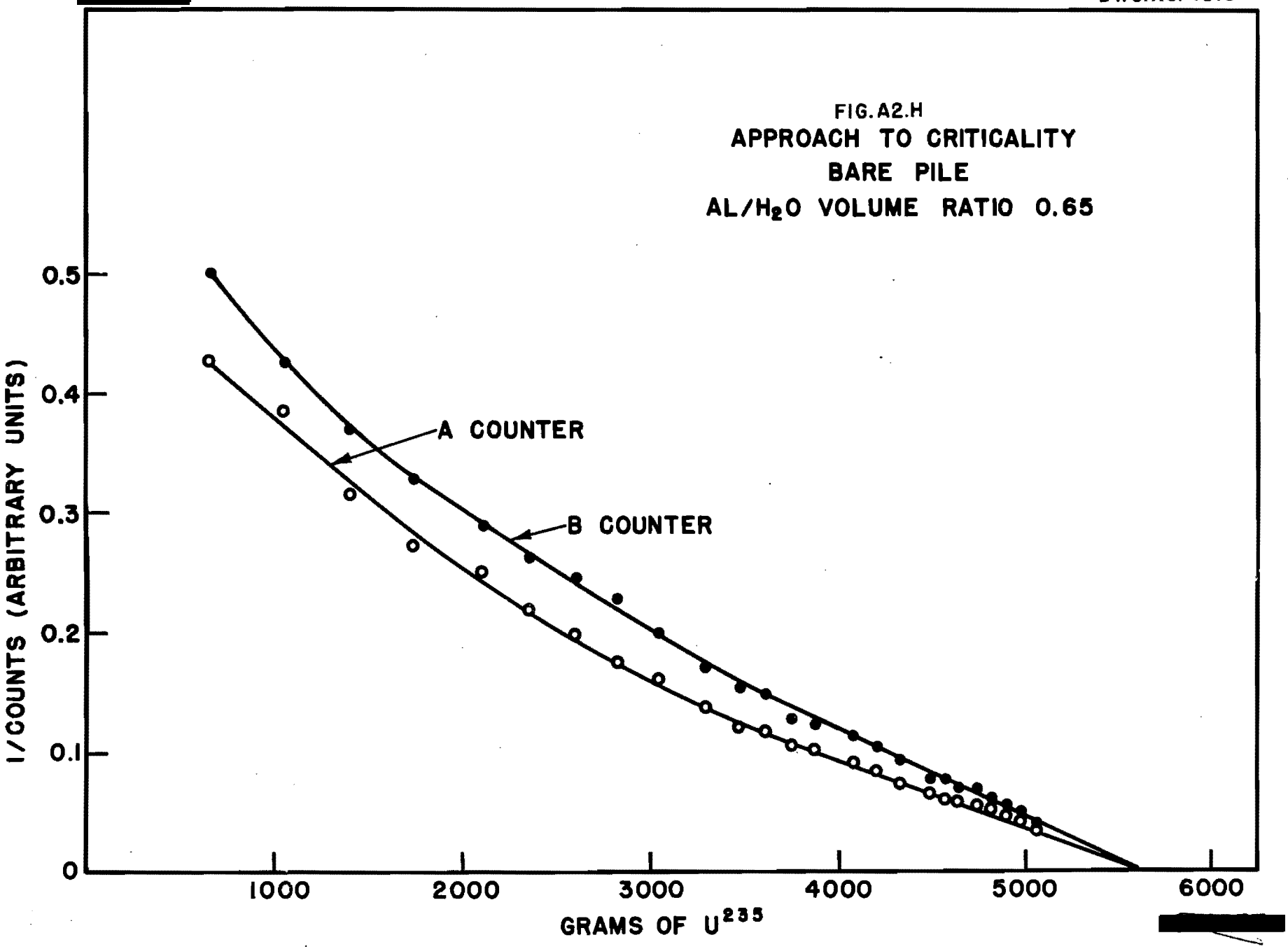
FIG. A2.H
APPROACH TO CRITICALITY
BARE PILE
AL/$H_2O$ VOLUME RATIO 0.65




[redacted]

it was necessary to add a small amount of poison (about 60 $cm^2$) in the form of thirty-one 0.030-in.-diameter gold wires, uniformly distributed throughout the core. The resulting reactor then has a core 22.5 by 22.5 cm in cross-section and 66 cm in height and containing 1.21 kg of $U^{235}$ at a fuel concentration of 38 g of uranium per liter of reactor volume. The $Al/H_2O$ ratio in the core was 0.65. The core was reflected on all sides, the reflector consisting of more than 30 cm thickness of beryllium, but there was no reflector on the top or bottom. When corrections were made for the presence of control rod ports (air holes) and for distributed gold wires in the core, the minimum critical mass of a square reactor of this type turned out to be 1.07 kg.

The second clean beryllium-reflected pile to be built was a thin-slab assembly. It was desired to find the critical mass of a rectangular geometry having a length-to-width ratio of approximately 5. The width selected for the core of the pile was four fuel tubes, or 11 cm, and the assembly became critical at a length of 19 fuel tubes (51 cm) with 30 $cm^2$ of excess poison in the core. The mass of $U^{235}$ in this case was 1.35 kg, and when corrections were made for excess poison and control rod ports, the minimum critical mass for this thin-slab geometry was 1.28 kg. The height of the reactor, reflector thickness, fuel concentration, and the $Al/H_2O$ volume ratio in the core were the same in this case as for the square assembly.

The plot of the reciprocal counting rate versus fuel aboard, shown in Fig. A2.I, indicates the linearity of results that can be obtained with reasonably good source and counter geometry.

Spatial distributions of thermal and epithermal neutron flux were measured in both the square reactor and the thin-slab reactor. The measurements were made with indium foils, used alternately bare and cadmium covered. Figure A2.J shows the distributions of thermal and epithermal neutron flux along a perpendicular bisector of the side of the square reactor. Figure A2.K shows the same distributions along a diagonal of the square assembly. All these foil measurements were taken at mid-heights of the assemblies.

The ratio of the thermal-neutron flux at certain points in the pile to that at the center of the core is of interest; these data are presented in Table A2.B.

[redacted] Dr 5195

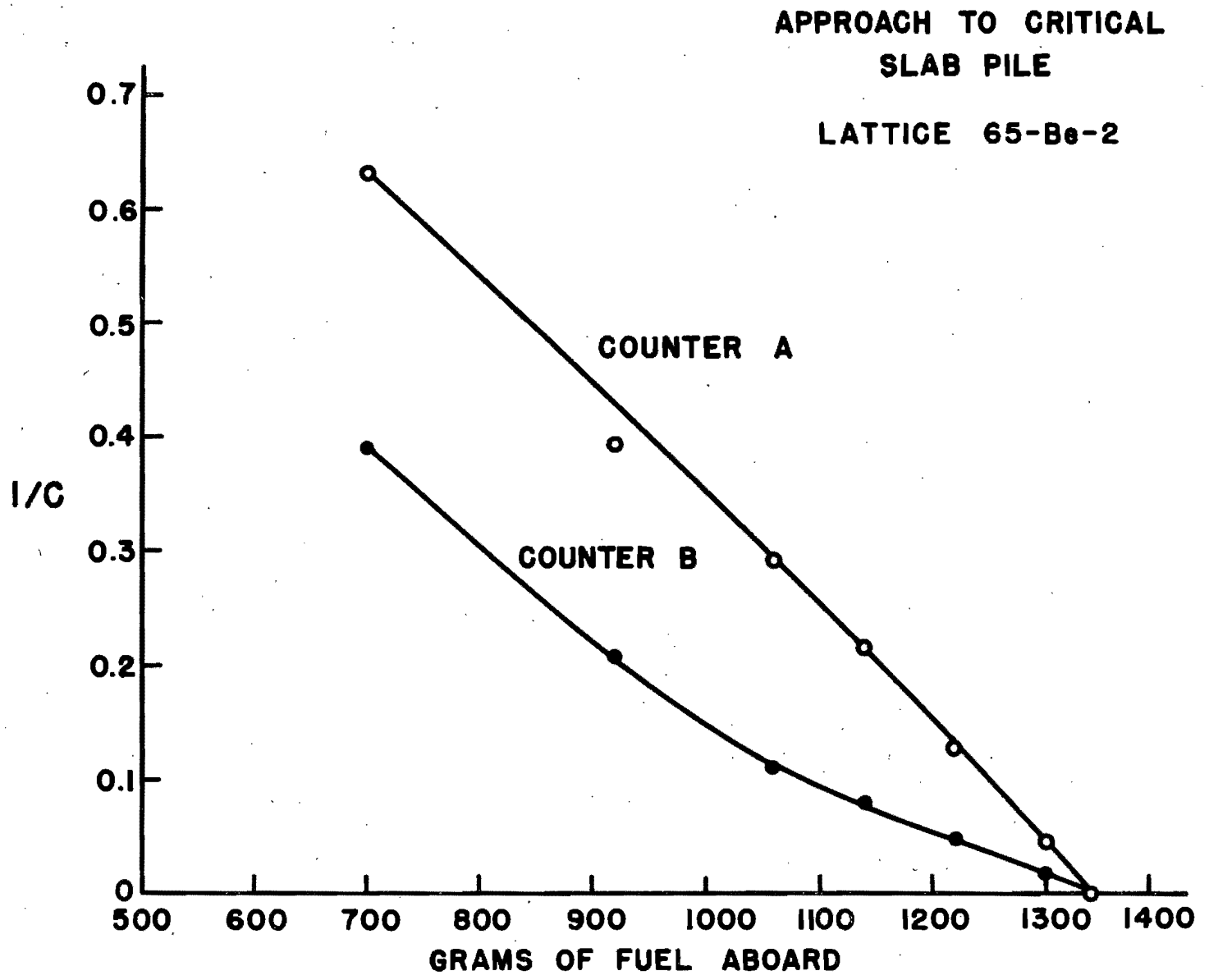



FIG. A2.1

Dr 5196
[redacted]

SQUARE Be PILE
LATTICE 65-Be-1
LATERAL NEUTRON DISTRIBUTIONS

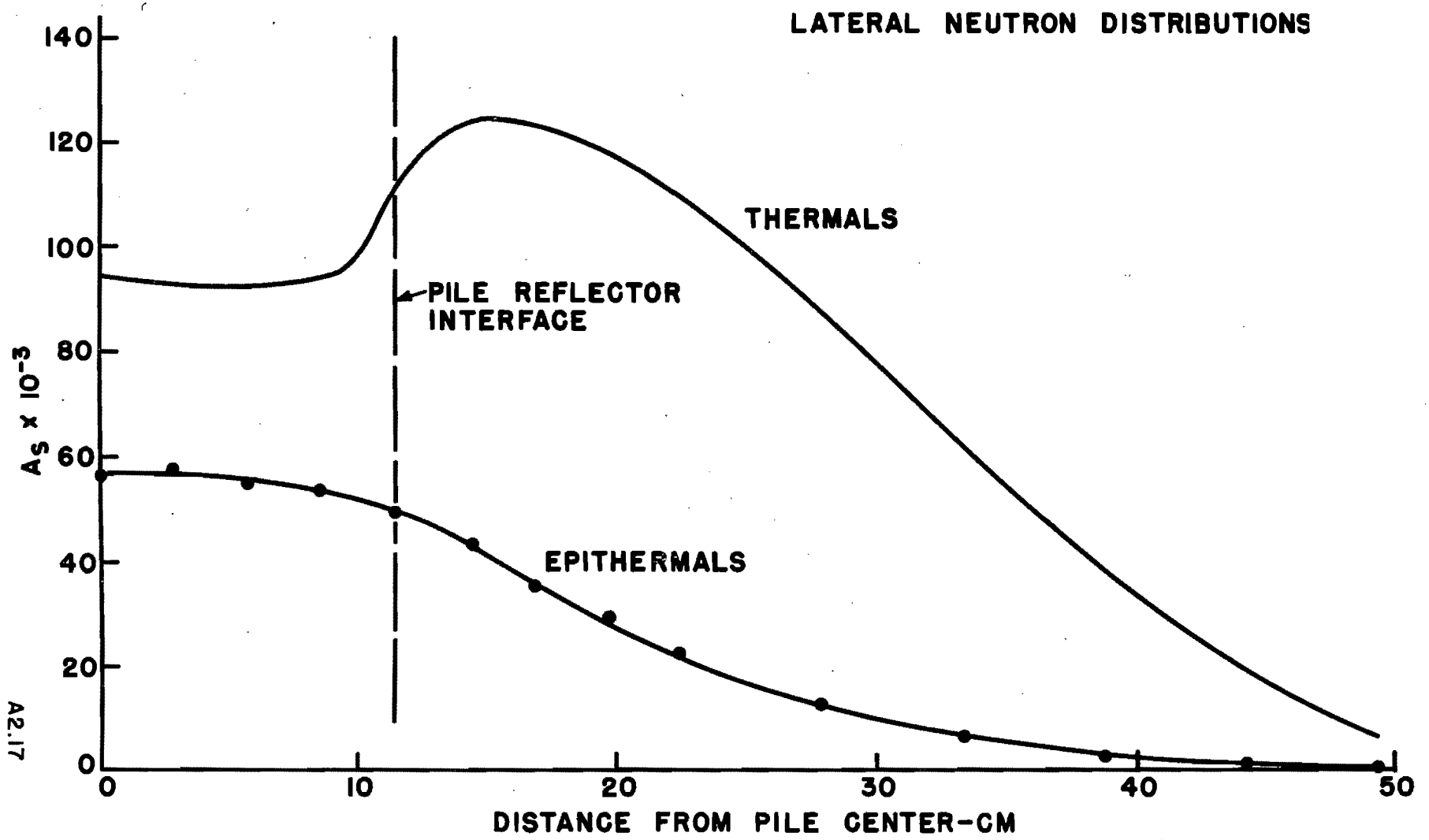



FIG. A2.J

A2.17

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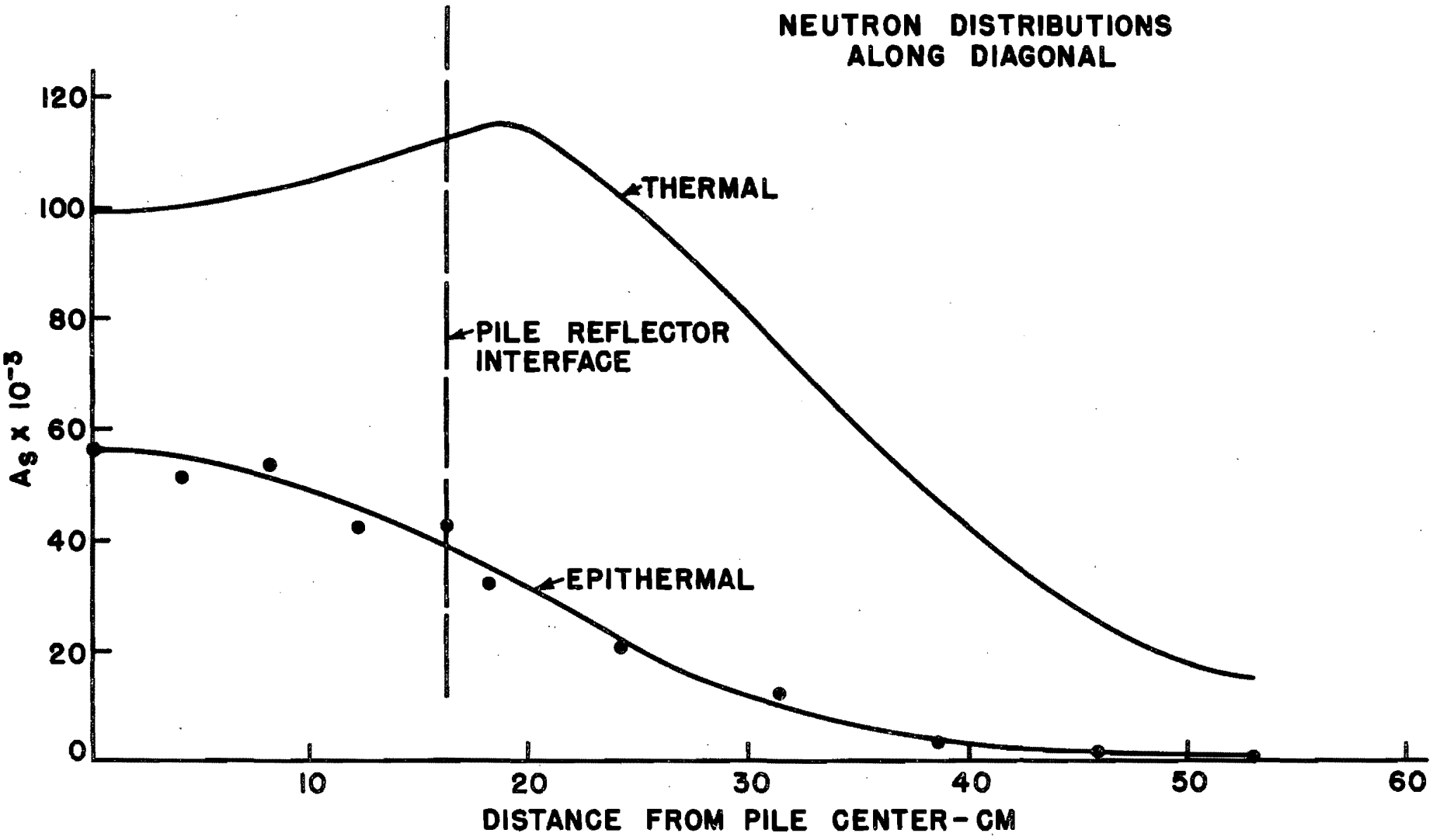
Dr 5199

SQUARE Be PILE
LATTICE 65-Be-1
NEUTRON DISTRIBUTIONS
ALONG DIAGONAL




A2.18

TABLE A2.B

**Relations Between Thermal Flux at Certain Locations in the Reactor and That at Center of Core**

| ASSEMBLY NO. | $\frac{\phi_{th}\ (\text{peak})}{\phi_{th}\ (\text{center})}$ | $\frac{\phi_{th}\ (\text{edge})}{\phi_{th}\ (\text{center})}$ | DISTANCE TO FALL TO CENTER VALUE (cm) | |
|---|---|---|---|---|
| | | | FROM CENTER OF CORE | FROM EDGE OF CORE |
| 1 (lateral) | 1.32 | 1.19 | 26.5 | 15 |
| 1 (diagonal) | 1.16 | 1.13 | 25 | 9 |
| 2 (lateral) | 1.42 | 1.24 | 22 | 16.5 |
| 2 (longitudinal) | 0.90 | 0.85 | | |

The ratio of the peak thermal flux to the thermal flux at the center is an indication of the maximum slow flux, for a given operating level, which the assembly will provide for experimental purposes. The ratio of a thermal flux at the edge of the core to that at the center is a measure of the maximum heat production per unit volume to be expected in the core. The extent to which the thermal flux holds up in the reflector is of interest in planning experiments which will make use of this flux.

A few preliminary experiments were carried out which yield an estimate of fast-neutron flux in these reactors. A threshold type fission chamber[3] was constructed using 145 mg of uranium from K-25 which was depleted in $U^{235}$ content to approximately 1 part in 100,000.

The fission chamber was first placed in a graphite column containing an Sb-Be source at a position where the thermal-neutron flux and the Cd ratio were known from standard indium foil measurements. By exposing the chamber alternately bare and Cd covered, it was determined that 93% of the fissions in $U^{235}$ were caused by neutrons of energy below the Cd cutoff at a position in the graphite column approximately 10 cm from the Sb-Be source where the Cd ratio was 3.7.

The chamber was then placed at the center of the active lattice of the thin-slab pile where the counting rate, when corrected for thermal fissions in $U^{235}$, indicated that the flux of neutrons having energies above the $U^{238}$ threshold (>1 Mev) is 0.50 $(nv)_{th}$. This result is in reasonably good agreement with the calculated virgin flux in this core, 0.60 $(nv)_{th}$ (see MonP-272). The correction for thermal fissions in $U^{235}$ in this case amounts to only 2% of the total counting rate.

## A2.3 REFERENCE REPORTS


(1) Mann, M. M. and Martin, A. B., *Critical Experiments on a Small Reactor of Enriched U-235/Al-$H_2O$ Moderator and $D_2O$, Beryllium, and $H_2O$ Reflectors*, MonP-357, August 18, 1947.

(2) Mann, M. M. and Martin, A. B., *Further Critical Experiments on a Small Reactor of Enriched U-235/Al-$H_2O$ Moderator and Beryllium Reflector*, ORNL-79, September 16, 1948.

(3) Jordan, W. H. and Bell, P. R., *Instructions for Use of Al Amplifier and Preamplifier*, MonP-323, July 29, 1947.

## Appendix 3

## CRITICAL EXPERIMENTS ON SIMULATED MTR STRUCTURE[1]

In particular, the third set of critical experiments was designed to study the following:

1. The critical mass (or, more directly, the excess of $\Delta k/k$) of slab reactors as a function of the length-to-width ratio of the core.
2. The effect of holes on the $k$ of the reactor.
3. The heat load in the reflector arising from gamma-ray absorption.
4. Spatial distributions of thermal, indium resonance, and fast ($E > 1$ Mev) neutron flux in the core and in the reflector, particularly the distributions of thermal and indium resonance neutrons in the experimental holes.
5. The effectiveness of control rods of various materials and of various sizes in terms of $\Delta k/k$.

### A3.1 THE EXCESS REACTIVITY OF U-Al-$H_2O$ REACTORS, WITH Be REFLECTOR, AS A FUNCTION OF GEOMETRY AND REFLECTOR COMPOSITION

The starting point for these experiments was the small clean-slab reactor which is described in Appendix 2, p. A2.15. The core was 51 by 11 cm in cross-section and 66 cm high. It had a 30-cm beryllium reflector on each face except the top and bottom. The volume ratio of Al to $H_2O$ was 0.65, and the fuel concentration in the core was 37.1 g of $U^{235}$ per liter of core.

The size of the reactor was increased in a series of discrete steps, the final objective being a core of dimensions 71 by 22.5 by 66 cm, with a reflector on all sides (except top and bottom) consisting of 30 cm of beryllium, diluted with about 2% water (imitated by plexiglas). This final assembly approximated very closely the dimensions and composition of one of the probable configurations of the high-flux reactor.

The major steps in building up the small slab to final size were the following:

1. Increase in length of the slab from 51 to 71 cm and addition of a 30-cm graphite layer to one side (long side of the beryllium reflector).
2. Increase in thickness of the slab from 11 to 17 cm.
3. Increase in thickness of the slab from 17 to 22.5 cm.
4. Completion of a simulated high-flux reactor, putting 30 cm of graphite on each side of the beryllium reflector and introducing the seven large experimental holes in the reflector. The final assembly has been shown in Fig. A3.A.

As the size of the reactor was increased over that of the small clean slab, it was necessary to add uniformly distributed poison in the core to hold $k$-effective to a value close to unity. The poison used in these experiments was type 347 stainless steel in the form of strips 1/8 by 15/16 by 126 in. These were inserted in place of aluminum strips of the same dimensions which were used to attain the desired $Al/H_2O$ volume ratio in the core. Although this substitution changed the $Al/H_2O$ ratio to some extent, the magnitude of the change was small and the effect on the neutron age was also small.

Beginning with the 51- by 11- by 66-cm reactor, containing 1.35 kg of fuel, both poison and fuel were added in appropriate increments until a length of 71 cm was reached. At this stage the reactor contained 1.94 kg of fuel, and the excess 0.59 kg (over the clean critical mass of 1.35 kg) required 266 $cm^2$ of uniformly distributed poison to hold the excess reactivity to zero with all control rods removed. Table A3.A shows the excess reactivity, represented by poison content, for each of the assemblies studied. The excess $\Delta k/k$ is obtained by computing the $k$ for each assembly from $k = \sigma_{25}/\sigma_{total}$ where $\sigma_{total} = \sigma_{25} + \sigma_{Al} + \sigma_{H_2O} + \sigma_{poison}$; then $\Delta k/k = (k_0 - k_i)/k_i$, where $k_0$ refers to the clean, unpoisoned core and $k_i$ to the poisoned assembly.

When the reactivity is changed by a means which does not alter the total amount of fuel in the core or the concentration of fuel, it can easily be shown that the fractional change in $k$ is given simply by $\Delta k/k = \Delta\Sigma/\Sigma$, where $\Delta\Sigma$ is the amount of uniform poison added or removed from the core and $\Sigma = N\sigma$ is the total cross-section of the reactor having the smaller amount of poison. For example, when graphite is added to one side of the beryllium reflector of assembly 3 (Table A3.A), 37 $cm^2$ of additional poison is required to offset the increased reactivity, and the effect of the graphite layer is given by

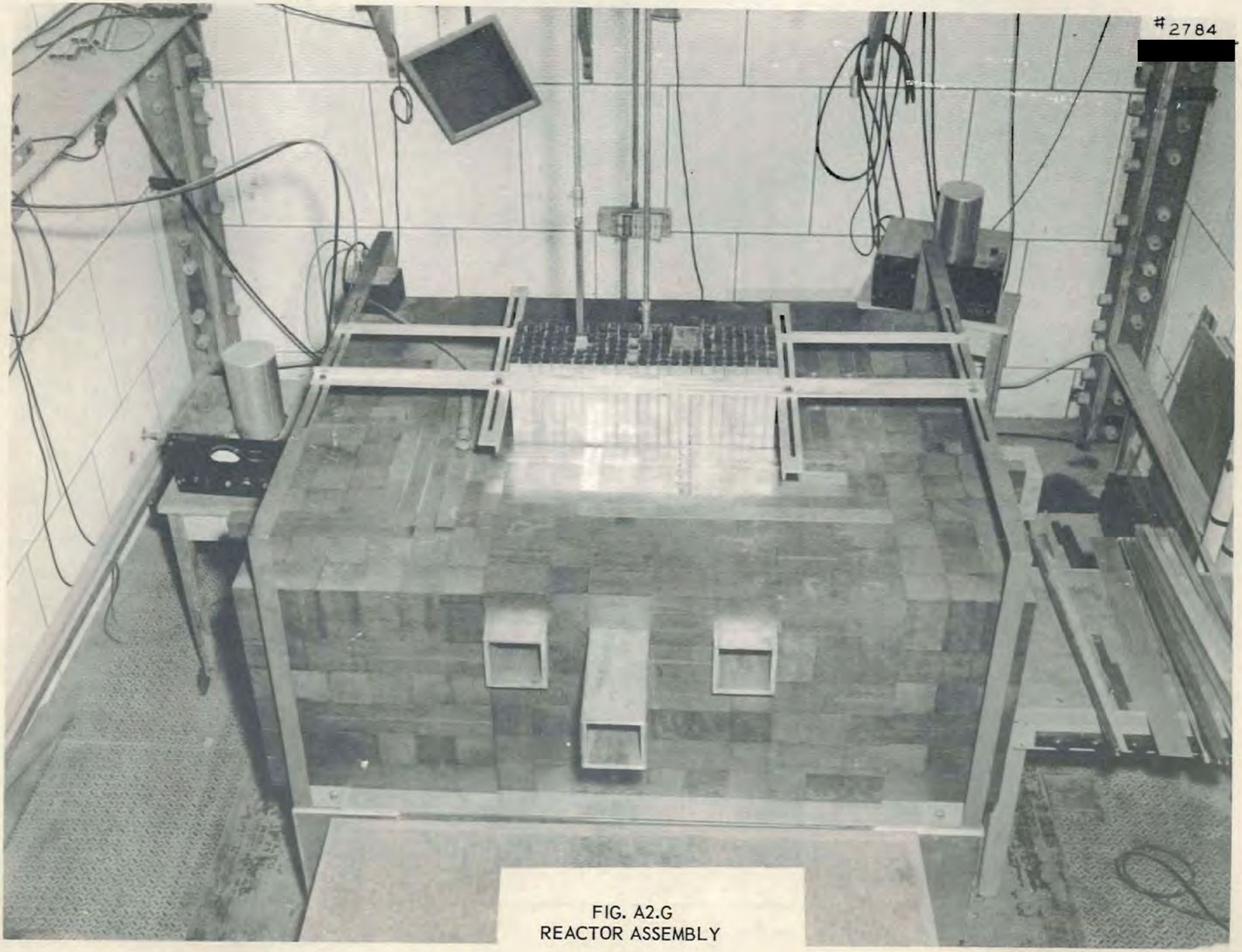



FIG. A2.G
REACTOR ASSEMBLY

$$37/4384 = \Delta\Sigma/\Sigma = \Delta k/k = 0.9\%$$

Table A3.A summarizes the pertinent criticality data for each of the beryllium-reflected reactors in this series. Column 3 indicates the amount of $U^{235}$ required to fill the core having the dimensions shown in Column 2 at a fuel concentration of 37.1 g per liter of core. Column 4 shows the amount of uniformly distributed poison required to hold the excess $k$ of the assembly to zero, i.e., to hold the reactor just at criticality with all control rods withdrawn. For each assembly, the $\Sigma_{pile} = \Sigma_{25} + \Sigma_{Al} + \Sigma_{H_2O}$ is given in Column 5, the experimental $k$ is shown in Column 6, and in Column 7 is shown the excess $\Delta k/k$ which can be made available by the removal of distributed poison.

**TABLE A3.A**

**Critical Mass and Poison Content (Excess $k$) of U-Al-$H_2O$ Reactors with Be Reflector**

($Al/H_2O$ volume ratio = 0.65; Be thickness = 30 cm; reactor bare top and bottom; $k_0$ = 1.6167 for clean core)

| ASSEMBLY NO. | DIMENSIONS $L \times W \times H$ | MASS (kg) | POISON ($cm^2$) | PILE ($cm^2$) | $k$ | $\Delta k/k$ EXCESS (%) |
|---|---|---|---|---|---|---|
| 1 | 22 x 22 x 66 | 1.21 | 66 | 2591* | 1.5626 | 3.5 |
| 2 | 51 x 11 x 66 | 1.35 | 30 | 2882* | 1.5849 | 2.0 |
| 3a | 71 x 11 x 66 | 1.94 | 266 | 4118 | 1.5184 | 6.5 |
| 3b (30 cm graphite on one side) | 71 x 11 x 66 | 1.94 | 303 | 4118 | 1.5057 | 7.4 |
| 4 (30 cm graphite on one side) | 71 x 14 x 66 | 2.44 | 640 | 5179 | 1.4388 | 12.4 |
| 5 (30 cm graphite on one side) | 71 x 17 x 66 | 2.94 | 1140 | 6240 | 1.3670 | 18.3 |
| 6 (30 cm graphite on one side; 2% water in Be) | 71 x 22.5 x 66 | 3.95 | 1848 | 8384 | 1.3247 | 22.0 |
| 7 (30 cm graphite on all sides of Be; seven 6-in. holes in reflector; 2% water in Be) | 71 x 22.5 x 66 | 3.95 | 1580 | 8384 | 1.3603 | 18.8 |

* These contain fractional fuel tubes.

## A3.2 EFFECTS OF THE EXPERIMENTAL HOLES THROUGH THE REFLECTOR

The research facilities of the high-flux reactor as conceived in 1948 included seven large experimental holes, four of which began at the core reflector. The other three holes began 15 cm out in the beryllium and extended through the remainder of the reflector. In connection with these irregularities in the reflector a number of experiments were of interest: (1) the cost of the holes in terms of $\Delta k/k$, or in terms of additional fuel, (2) the effect on the reactivity of the core of filling any one or all of the holes with water, and (3) the effect of various absorbers, such as experimental apparatus, at various positions in the holes. Item 2 is of interest from the standpoint of safety since it is conceivable that an accident might occur in which the wall of the hole would collapse or otherwise fail, causing the hole to fill with water and thereby increasing the reactivity. Furthermore, it is likely that water cooling will be desired for experimental apparatus in one or more of the holes on certain occasions, and the data obtained here will indicate the maximum effect of water cooling on $\Delta k/k$.

The location of the holes in the critical experiments is shown in Fig. A3.A, and also in Fig. A3.B which is a sketch of the assembly 3b of Table A3.A. The so-called 6-in. hole is made up of a hollow aluminum tube of square cross-section, having ½-in. wall thickness and measuring 6 in. overall along the edge. The 8-in. hole is of similar construction and measures 8 in. along the outer edge.

The effects of the various holes were measured by the following procedure: A reactor which was just critical with a given amount of poison and with no holes in the reflector was used. A portion of the graphite and beryllium in the region of the hole to be studied was removed, a square aluminum shell was inserted at the proper place, and the reflector was restacked around it. In order to offset the leakage through the hole and bring the reactor back to the point of criticality, an amount of distributed poison must be removed from the core. The effect of the hole on the reactivity of the reactor is given by $\Delta k/k = \Delta\Sigma/\Sigma$. Before inserting a close-fitting paraffin block into the aluminum shell to simulate water filling, the core was poisoned excessively to offset the expected increase in reactivity. When the hole was filled with paraffin, the distributed poison was again adjusted until the reactor was just critical and the reactivity change due to the introduction of paraffin was determined as before.

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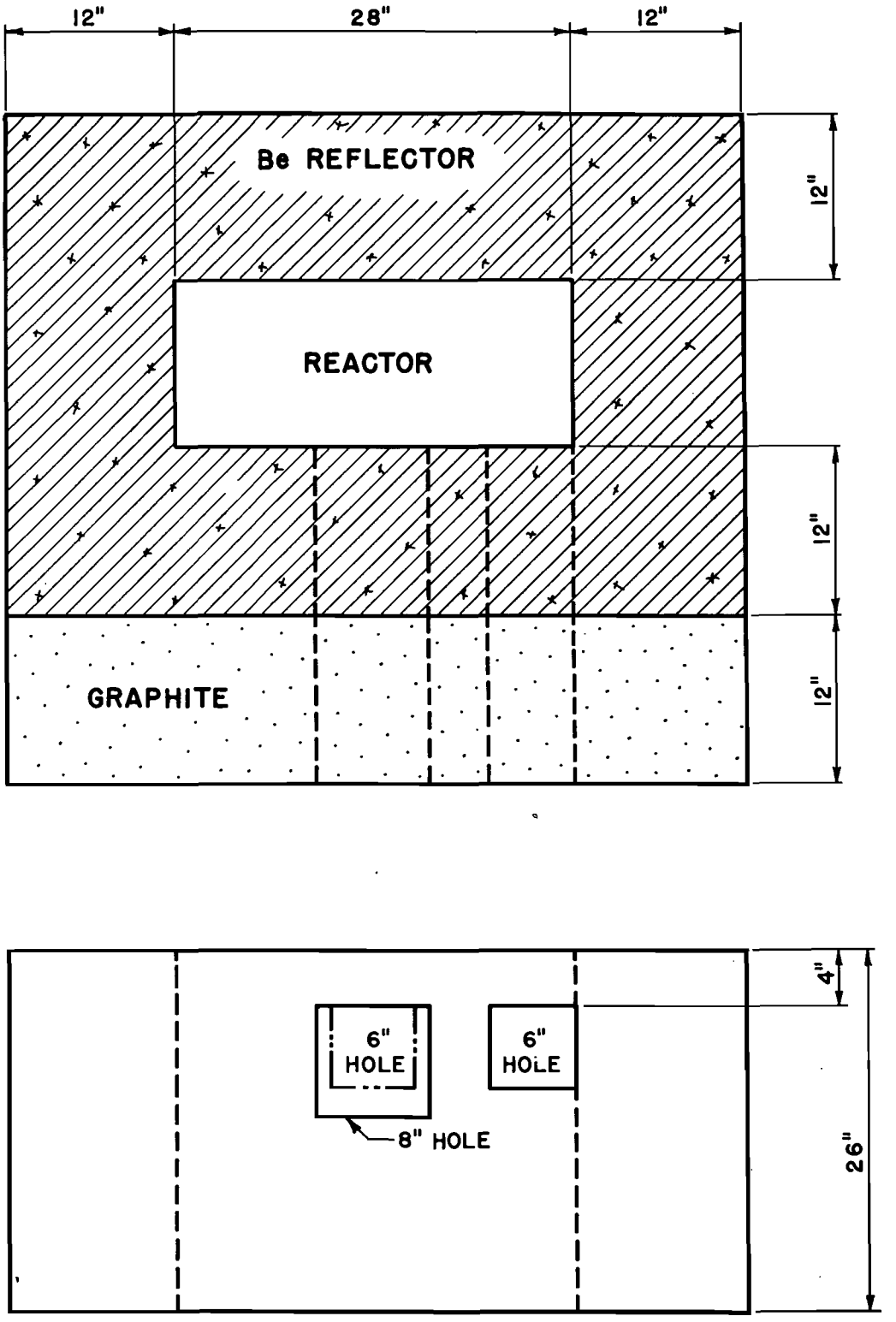


FIG. A3.B

**CRITICAL ASSEMBLIES**

**LOCATION OF HOLES IN REFLECTOR**

Table A3.B presents the results of the experiments just described for assemblies 3, 5, and 6 of Table A3.A. Measurements were taken for 6-in. holes at the edge positions and for both 6- and 8-in. holes at the center position. It was thought at one time that some interaction between the holes might take place, but measurements made on individual holes and on various combinations of two or more holes indicated that the effects were approximately additive.

**TABLE A3.B**

**Effect of Experimental Holes in the High-flux Reactor Critical Assemblies**

| Assembly No. | 3 | 5 | 6 |
|---|---|---|---|
| Reactor dimensions (cm) | 71 x 11 x 66 | 71 x 17 x 66 | 71 x 22.5 x 66 |
| Critical mass (kg) | 1.94 | 2.94 | 3.95 |
| Pile ($cm^2$) | 4,118 | 6,240 | 8,384 |
| Poison ($cm^2$) | 303 | 1,140 | 1,848 |
| Total ($cm^2$) | 4,421 | 7,380 | 10,232 |
| **Effect of the Holes in $\Delta k/k$ (%) Where $k$ Is the $k$ of the Reactor with Solid Reflector** | | | |
| Holes extending to reactor-reflector interface | | | |
| 6-in. hole on edge | 1.17 | 0.75 | 0.49 |
| 6-in. hole in center | 2.20 | | 0.86 |
| 8-in. hole in center | 4.80 | 2.84 | 1.71 |
| Holes beginning 15 cm from reactor | | | |
| 6-in. hole on edge | 0.17 | 0.12 | 0.08 |
| 6-in. hole in center | 0.34 | | 0.13 |
| 8-in. hole in center | 0.68 | 0.40 | 0.22 |
| Holes extending to reactor, water filled | | | |
| 6-in. hole on edge | 0.90 | 0.65 | 0.42 |
| 6-in. hole in center | | | 0.84 |
| 8-in. hole in center | 3.88 | 2.45 | 1.53 |
| Holes beginning 15 cm from reactor, water filled | | | |
| 6-in. hole on edge | | 0.08 | 0.06 |
| 6-in. hole in center | 0.24 | | 0.06 |
| 8-in. hole in center | 0.37 | 0.23 | 0.11 |

It is interesting to note that the effect of the holes decreases percentagewise as the thickness of the slab is increased. This was expected on theoretical grounds.

Table A3.C contains a summary of the total cost in $\Delta k/k$ for two alternate provisions for experimental holes. This table shows the change in $\Delta k/k$ that would be expected if the holes in these two cases should be filled with water. An estimate of the cost of the holes in terms of critical mass can be obtained by use of the empirical relation $\Delta M/M = 4.5\ \Delta k/k$. However, this relation leads to a reasonably accurate result only in the case of fuel $\Delta M$ added to the periphery of a cylindrical core. In the case of a rectangular geometry, the cost in critical mass ($\Delta M$) corresponding to a given $\Delta k$ will depend strongly on the statistical weight of the position at which the additional fuel is added, and the figure obtained from $4.5\ \Delta k/k$ is a reasonable approximation only for positions of average statistical weight.

**TABLE A3.C**

**Summary of Total Effect of Experimental Holes**

| Assembly No. | 3 | 5 | 6 |
|---|---|---|---|
| Reactor dimensions (cm) | 71 x 11 x 66 | 71 x 17 x 66 | 71 x 22.5 x 66 |
| Mass (kg) | 1.94 | 2.94 | 3.95 |
| Cost in $\Delta k/k$ of two 8-in. holes plus five 6-in. holes (%) | 9.33 | 5.73 | 3.56 |
| Cost in $\Delta k/k$ of seven 6-in. holes (%) | 6.39 | 4.23 | 2.62 |
| Effect in $\Delta k/k$ of water filling two 8-in. holes and five 6-in. holes (%) | 2.14 | 0.94 | 0.54 |
| Effect in $\Delta k/k$ of water filling seven 6-in. holes (%) | 1.41 | 0.66 | 0.34 |

The effect of possible absorbers (experimental apparatus) in the holes on the excess $k$ of the pile was studied by taking the extreme case of a cubical Cd box (0.020 in. of Cd, 10 cm along an edge). The effect of this relatively large absorber on the $k$ of the pile when it was placed at various positions in the holes is shown in Fig. A3.C. From these measurements it is estimated that the maximum excess $k$ required to offset the effect of experimental apparatus in the holes is 1.5%.

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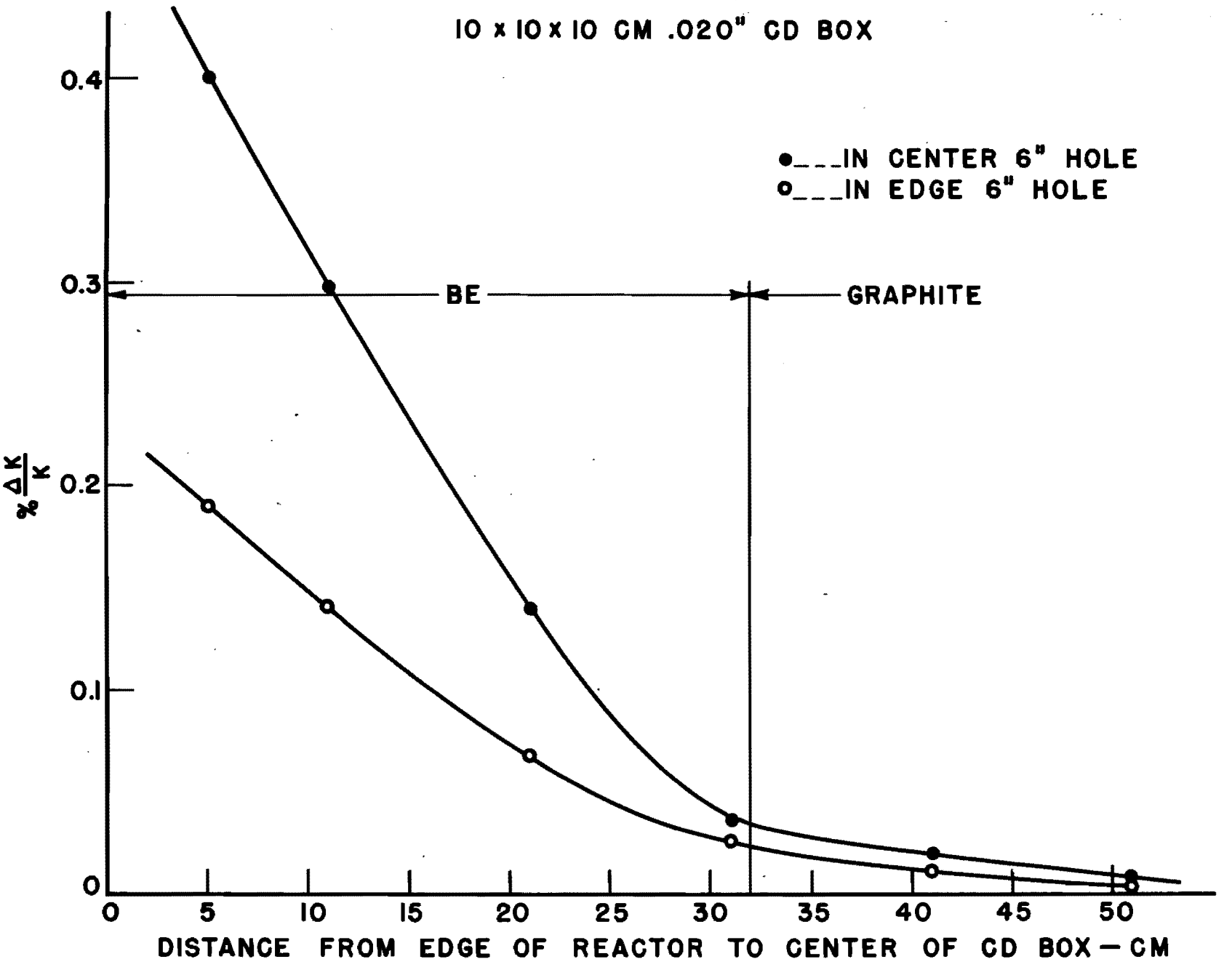
FIG. A3.C

EFFECT OF ABSORBER

10 x 10 x 10 CM .020" CD BOX



Finally the excess $k$ of the 3.95-kg mock-up of the high-flux reactor was measured. The characteristics of this mock-up include: (1) a core of dimensions 71 by 22.5 by 66 cm, containing 3.95 kg of $U^{235}$ in water solution at a concentration of 37.1 g uranium per liter of pile volume, (2) $Al/H_2O$ volume ratio in the core = 0.65, (3) a reflector on all sides consisting of 30 cm of beryllium plus 30 cm of graphite, (4) no reflector top or bottom, (5) lucite distributed throughout the beryllium to simulate 2% in cooling water, and (6) seven 6-in. experimental holes through the reflector as prescribed in the design of the high-flux reactor. The excess $k$ of this mock-up assembly is found to be 18.8%. When allowance is made for the top and bottom water reflectors of the high-flux machine, the total available excess $k$ is increased to about 20% for this loading. Therefore it appears that a comfortable margin exists over the total $k$ requirements, which are estimated (see Chapter 4) to be as follows:

| | |
|---|---|
| Xenon | 9.6% |
| Depletion | 3.5% |
| Temperature coefficient | 0.5% |
| Experimental apparatus in holes | 1.5% |
| Total $\Delta k/k$ required | 15.1% |

## A3.3 NEUTRON DISTRIBUTIONS

The spatial distributions of thermal and epithermal neutron flux in the 2.94-kg pile, the 3.95-kg pile, and in the various experimental holes of the mock-up assembly, as indicated in the listing below, were measured. These measurements were made with indium foils, used alternately bare and cadmium covered, as described previously. A calibration of foils in the standard (sigma) reactor indicated that the absolute flux (nv) is obtained from the measured saturated activities, $A$, shown in the attached figures by the following relations:

$$(nv)_{thermal} = 6.50A_{thermal}$$

$$(nv)_{epithermal} = 1.66A_{epithermal}$$

Several of the neutron distributions were measured as follows:

1. Figure A3.D: A lateral traverse across the 2.94-kg reactor (along the short axis of the slab, at mid-height).

2. Figure A3.E: A series of measurements along the centerline of the 6-in. hole which is located at the edge of the long side of the core: (a) with solid reflector in place, (b) with hole extending to the core-reflector interface, and (c) with the hole beginning 15 cm out from the edge of the core. These curves show the depressions in the "solid reflector" neutron distributions caused by the presence of a single 6-in. hole in the reflector.

3. Figures A3.F, G, H: These show the neutron distributions along the axes of the various experimental holes which were planned for the high-flux reactor. These measurements were made with all seven 6-in. holes (or equivalent) in the reflector.

4. Figure A3.I: The fast-neutron flux ($E$ > 1 Mev) and the indium resonance flux, measured along a line midway between two of the large experimental holes that extend through the reflector.

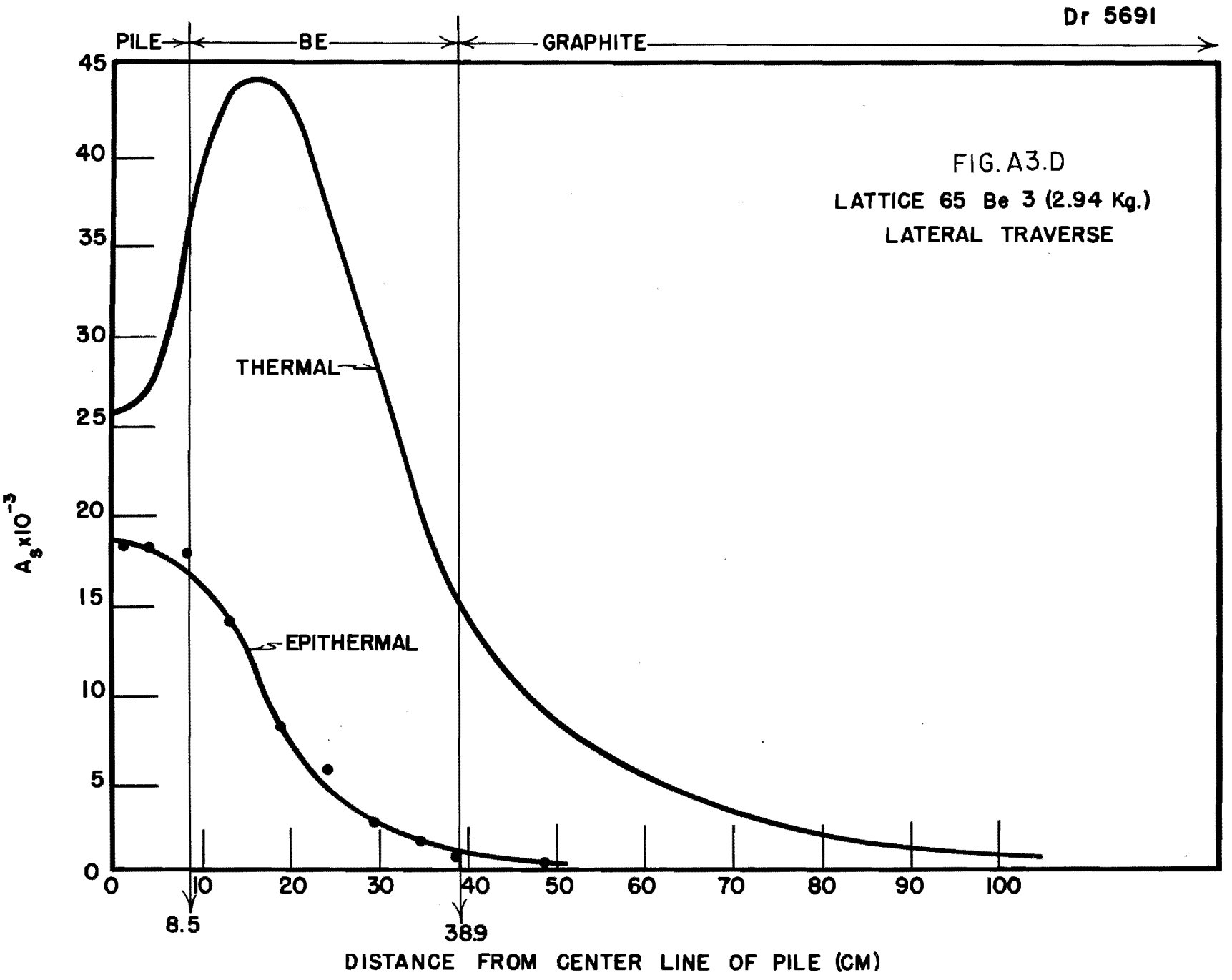
The $U^{238}$ fission chamber used to measure the flux of fast neutrons has been described in previous ORNL reports.[2,3]

The results of these experiments indicate that the $(nv)_{fast}$ near the center of the 3.95-kg assembly is 40% of the $(nv)_{thermal}$, in reasonably good agreement with the calculated value of $(nv)_{virgin}$ = 60% of $(nv)_{thermal}$ given in the report MonP-272.

## A3.4 GAMMA HEAT PRODUCTION IN THE REFLECTOR

In the high-flux reactor the beryllium is water cooled and there is no concern regarding the possibility of excessive heating in this portion of the reflector. The graphite, however, is cooled only by air, and empirical information on the heat production in graphite was desired to substantiate theoretical calculations. In the reflector, heat production is primarily from absorption of gamma rays. Most of these come from the core of the reactor, but some arise from neutron capture in beryllium and in graphite.

The gamma-ray absorption measurements were made in the rectangular assembly having core dimensions 71 by 17 by 66 cm and loaded to 2.94 kg of fuel. Measurements were made in the core and in the reflector along a line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PILE
BE
GRAPHITE
FIG. A3.D
LATTICE 65 Be 3 (2.94 Kg.)
LATERAL TRAVERSE
THERMAL
EPITHERMAL
$A_s \times 10^{-3}$
45
40
35
30
25
20
15
10
5
0
0
10
20
30
40
50
60
70
80
90
100
8.5
38.9
DISTANCE FROM CENTER LINE OF PILE (CM)

[redacted]

FIG. A3.E

4 KG SLAB PILE (71 x 22.5 x 66 CM)

2% $H_2O$ IN BE

NEUTRON DISTRIBUTIONS ALONG AXIS OF 6" HOLE AT EDGE OF REACTOR

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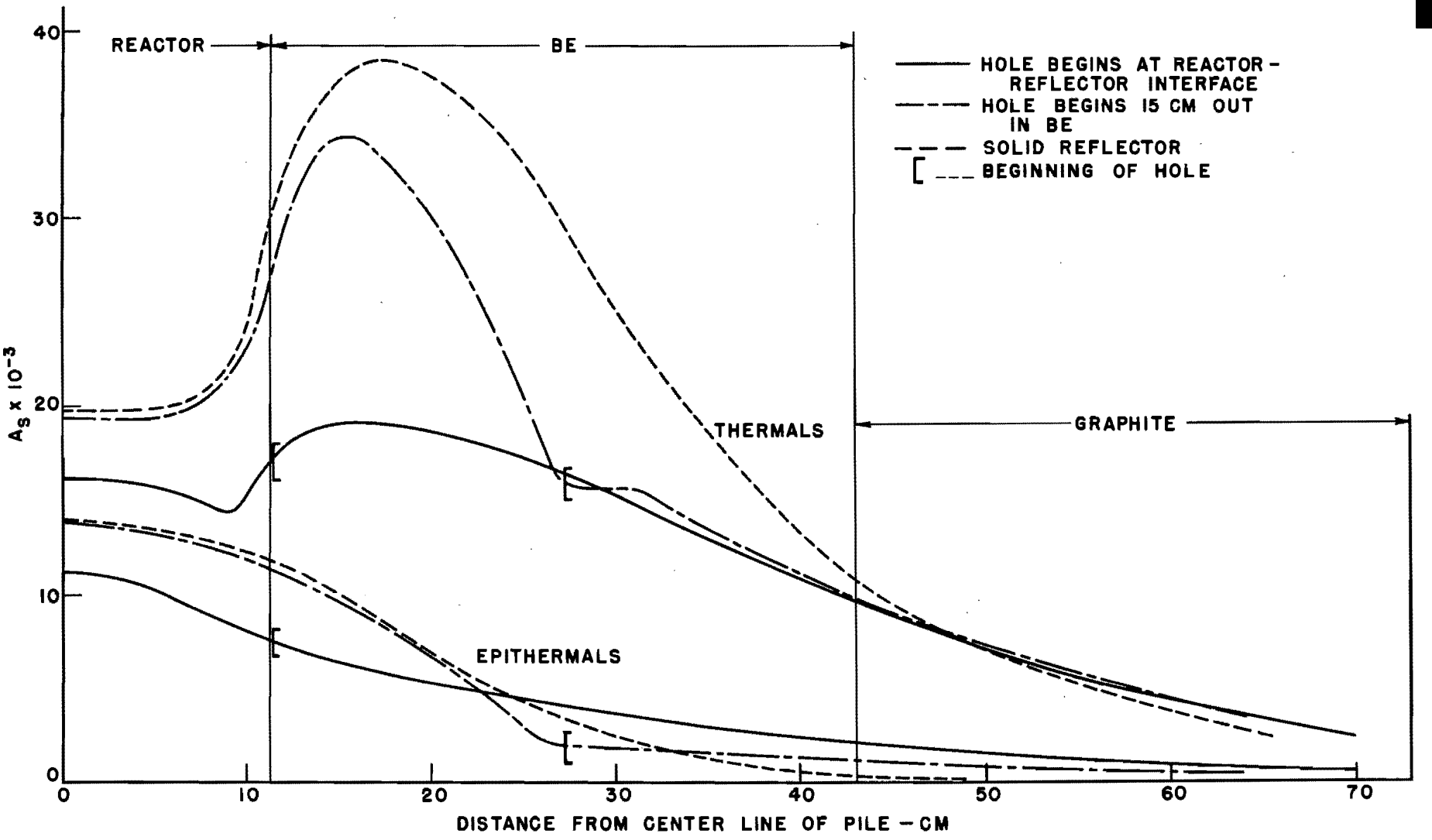


A3.13

[redacted]

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FIG. A3.F

4 Kg SLAB PILE (71 x 22.5 x 66 CM)

7 SIX INCH HOLES IN REFLECTOR

LATERAL NEUTRON DISTRIBUTIONS (ALONG AXIS OF CENTER HOLE)

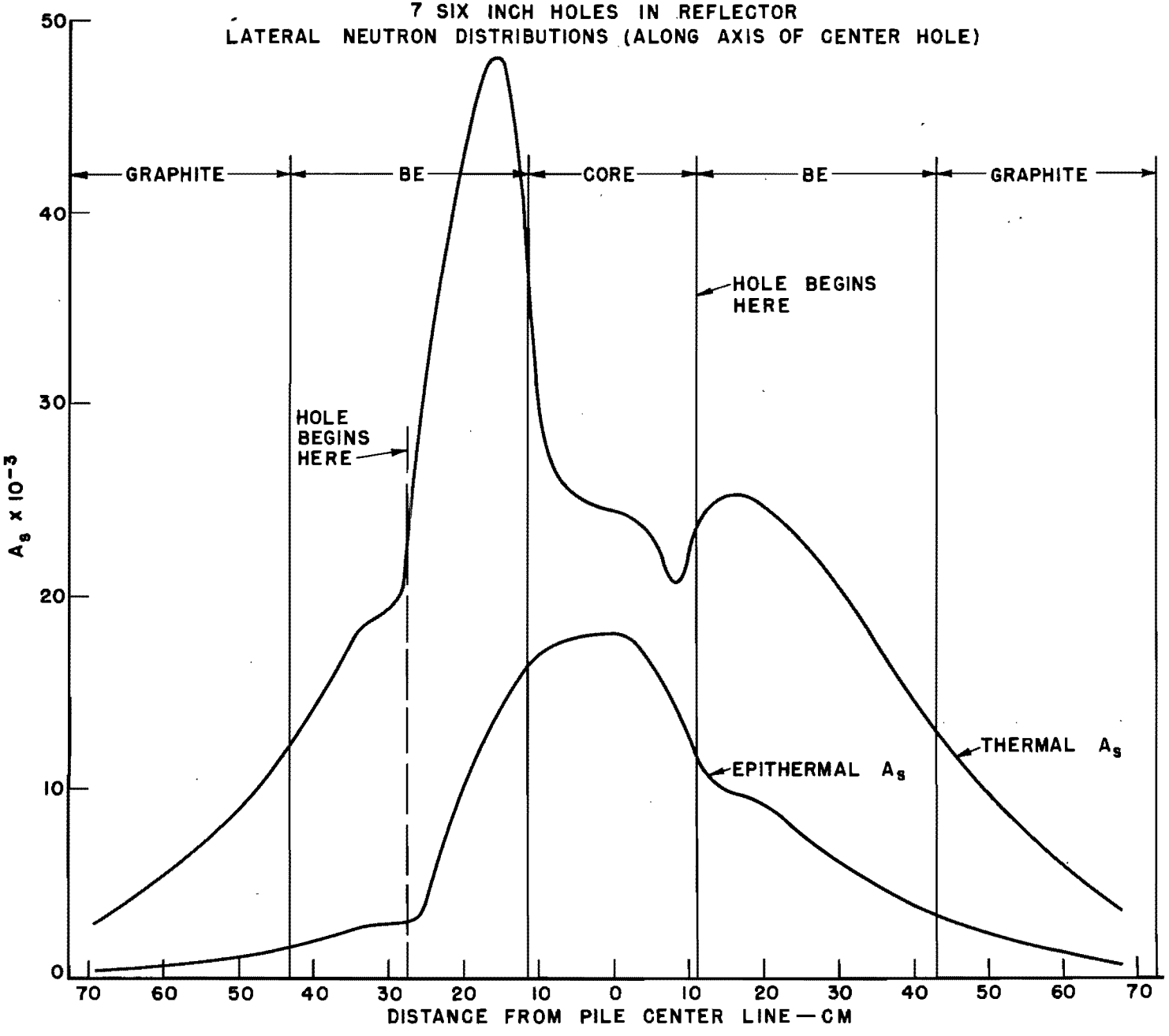

FIG. A3.G

4 KG SLAB PILE (71 X 22.5 x 66 CM)
7 SIX INCH HOLES IN REFLECTOR
LATERAL NEUTRON DISTRIBUTIONS (ALONG AXIS OF EDGE HOLE)

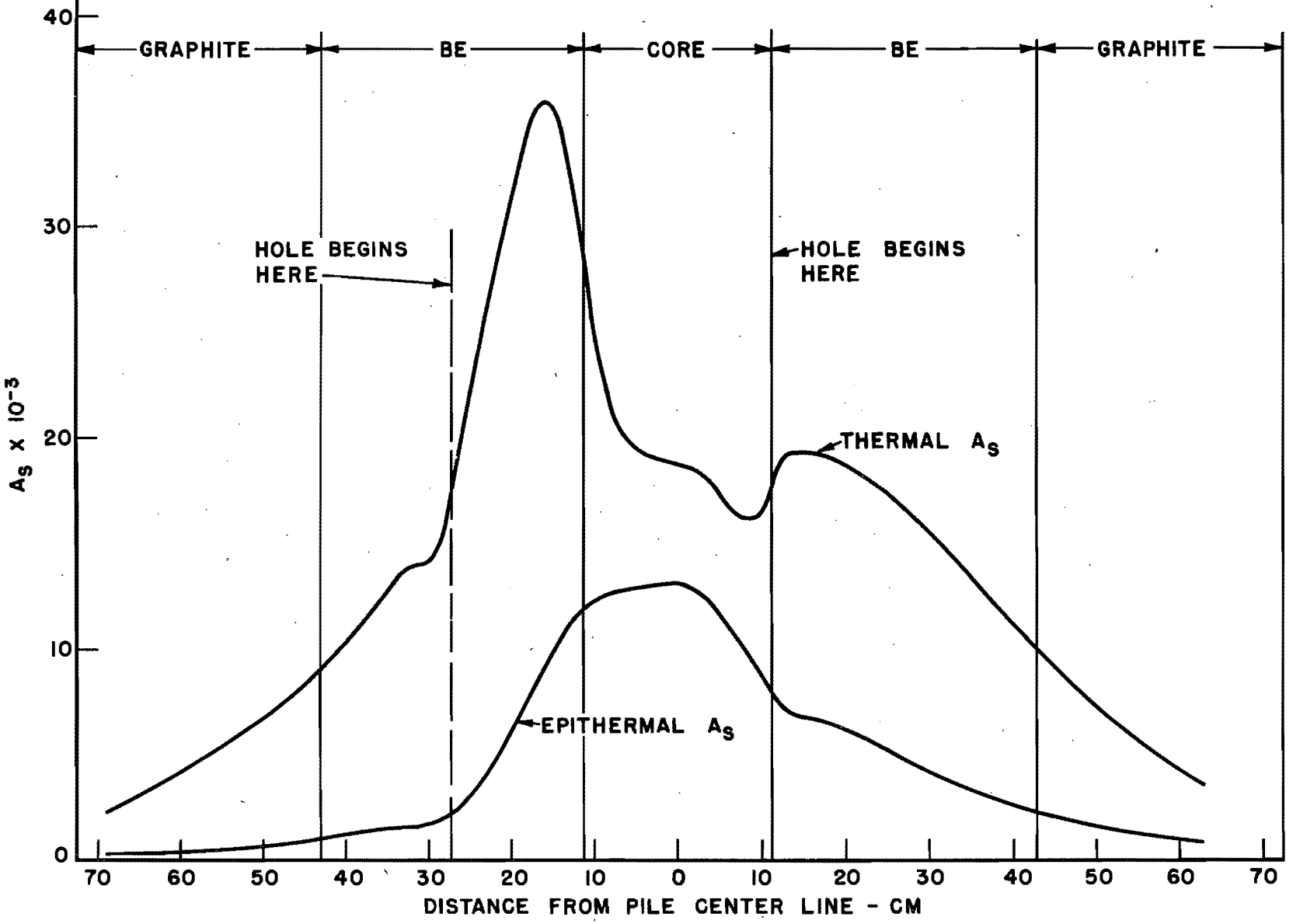

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FIG. A3.H

4 Kg SLAB PILE (71 X 22.5 x 66 CM)
7 SIX INCH HOLES IN REFLECTOR
LONGITUDINAL NEUTRON DISTRIBUTIONS (ALONG AXIS OF END HOLE)

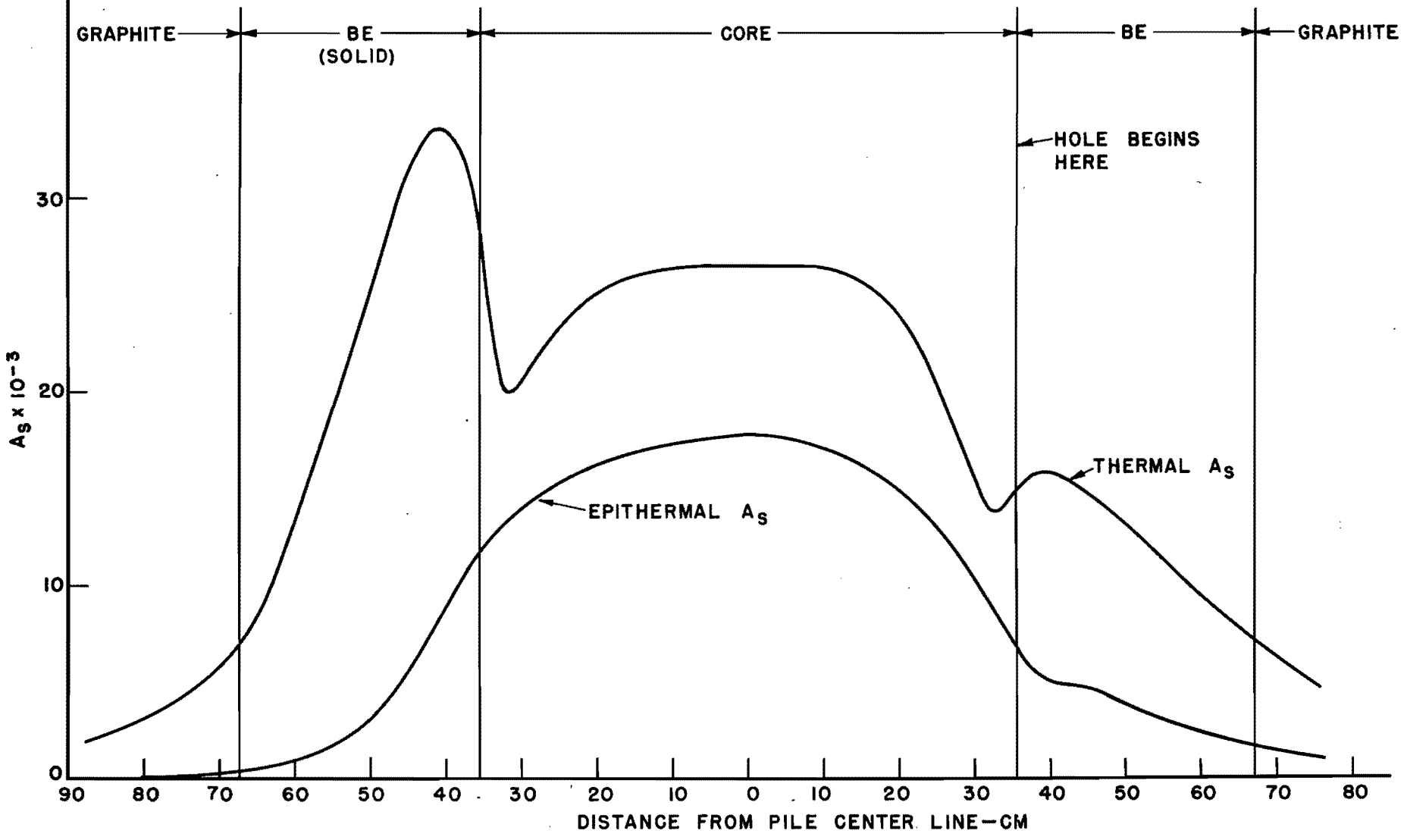

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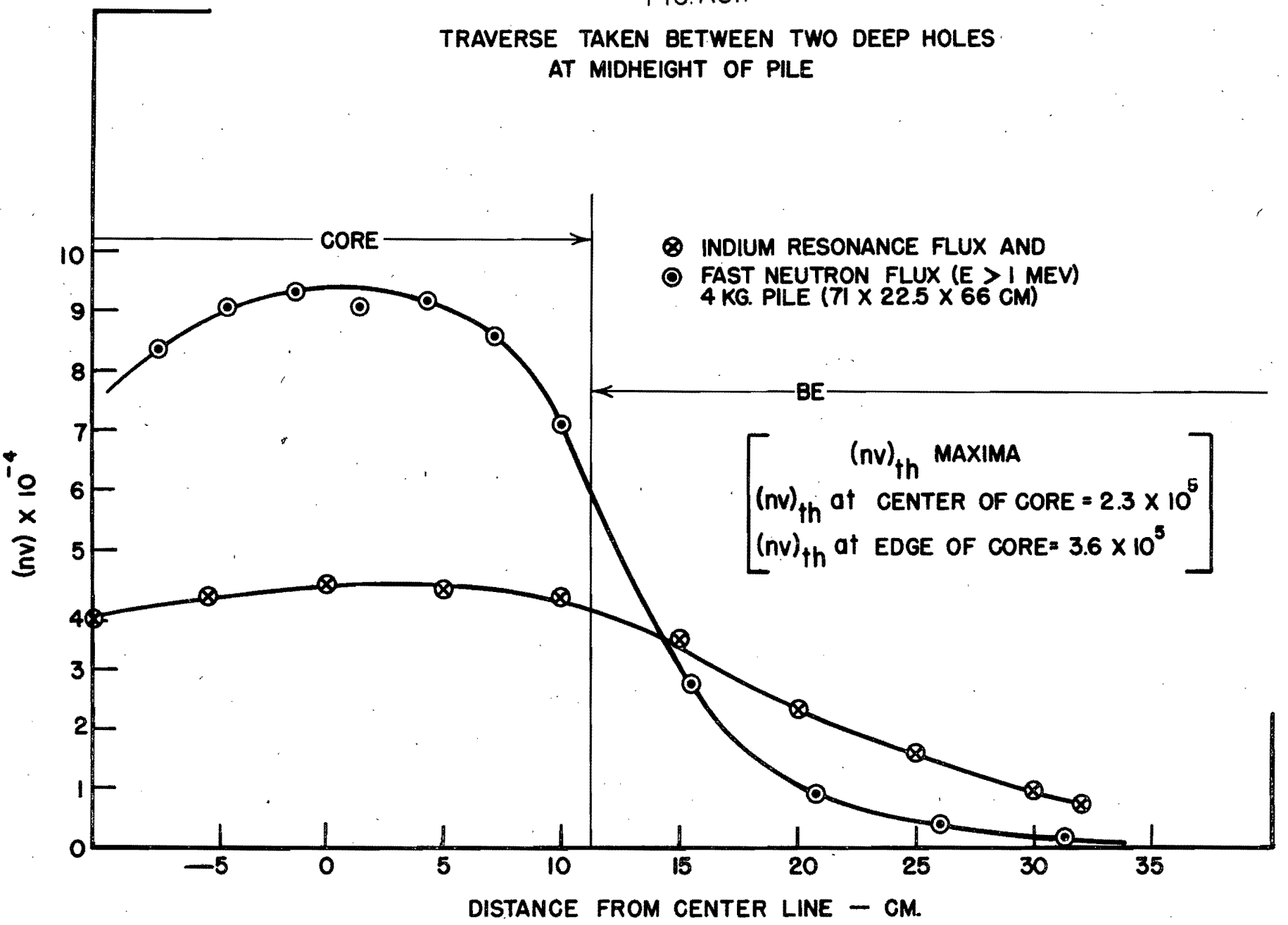
FIG. A3.1

TRAVERSE TAKEN BETWEEN TWO DEEP HOLES AT MIDHEIGHT OF PILE



perpendicular to the long dimension of the reactor. Three graphite ion chambers having volumes of approximately 1, 10, and 200 cc were used to cover the range of gamma flux encountered in going from the interior of the core through the beryllium reflector and out to the edge of the graphite portion of the reflector. In order to minimize the possibility of measuring ionization due to unforeseen neutron reactions in the gas of the ion chamber, the measurements were taken with each of four filling gases: air, argon, $CO_2$, and helium.

The lower solid-line curve in Fig. A3.J represents the total heat production in the reflector calculated from theoretical considerations. This calculation was based on the neutron distribution for a thinner slab pile (70 by 11 by 60 cm) where the heat load in the reflector is somewhat heavier than in the case of the assembly in which our measurements were made (71 by 17 by 66 cm). The measured spatial thermal-neutron distribution in this assembly is shown as a dotted curve.

## A3.5 CONTROL ROD EXPERIMENTS

A number of experiments were carried out to study the effectiveness of control rods of various sizes in the 4-kg mock-up of the high-flux reactor. The experiments are performed in the following way: the uniform poison in the "clean" core (before the control rod is inserted) is adjusted until the assembly is just critical. The control rod is inserted and the amount of uniformly distributed poison in the core is reduced until the assembly is just critical. When the experiment involves no change in the amount or concentration of fuel in the core, the percentage change in $k$, i.e., $\Delta k/k$, is given simply by $\Delta\Sigma/\Sigma$ where $\Delta\Sigma$ is the change in uniform poison and $\Sigma$ is the total cross-section of the active portion of the core after the rod is inserted. In experiments which entail the removal of fuel *and* the insertion of a control rod, the $k$ of the clean core, $k_1$, is calculated, the $k$ of the active portion of the core after insertion of the rod, $k_2$, is calculated, and $\Delta k/k$ is then given directly by $(k_1 - k_2)/k_2$.

The Cd control rods tested in these experiments were essentially replicas of those designed for the high-flux reactor in so far as dimensions and composition are concerned. They were in the form of hollow cylinders of square, 3¼ by 3¼ in., cross-section and 26 in. long and were filled with water to duplicate the conditions that will be obtained in the case of the high-flux reactor.

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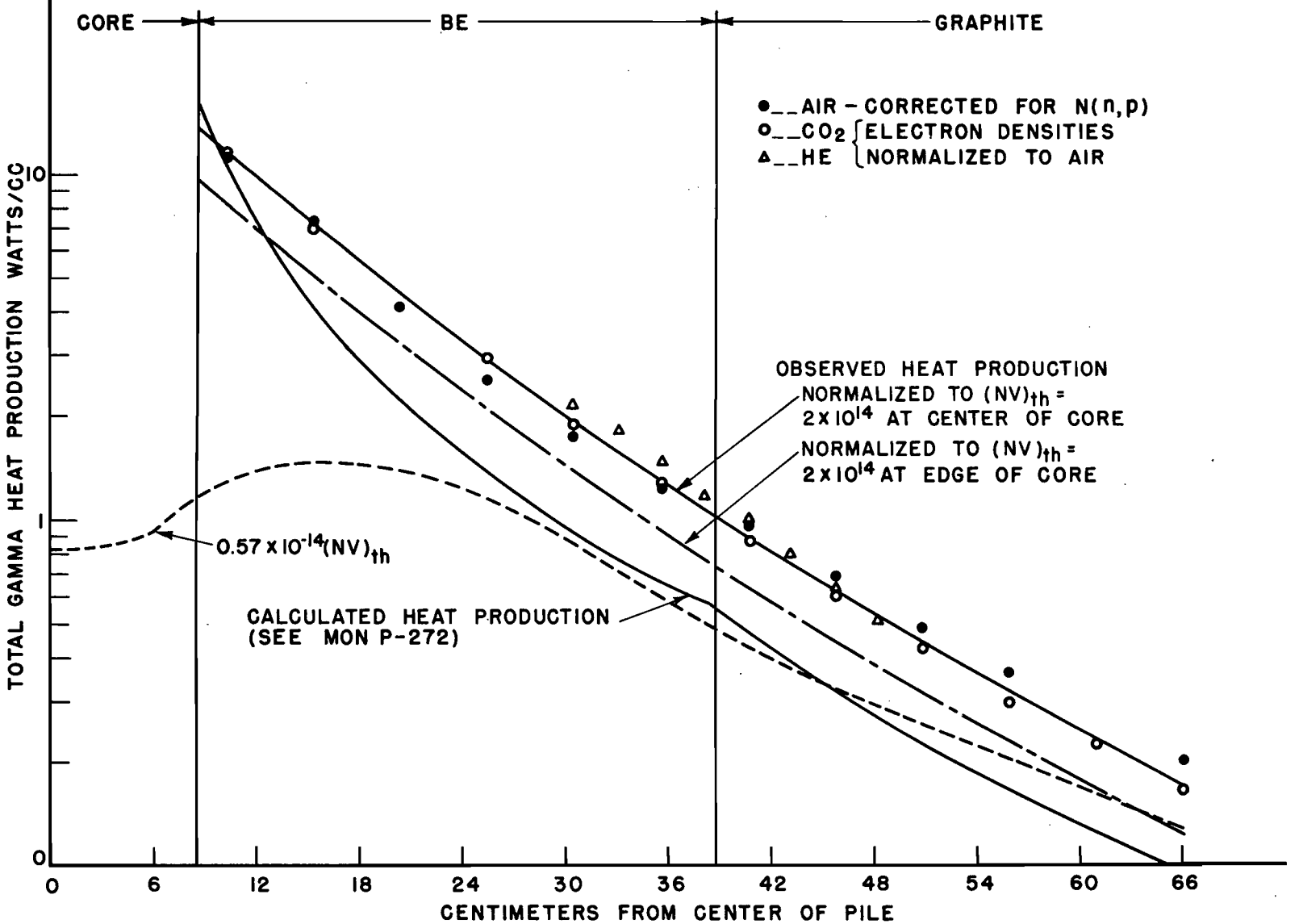
FIG. A3.J
GAMMA HEAT PRODUCTION IN REFLECTOR
OF 3 KG PILE (71 x 17 x 66 CM)



The thorium rods for the high-flux reactor are designed to be made up of several plates; they are stacked in a sandwich-like assembly, with spaces between plates for the passage of cooling water. Iron was used instead of thorium for reasons of availability and ease of fabrication, the amount of iron being adjusted to give the same total thermal cross-section as that provided by the thorium rods. The test rod then consisted of a rectangular sandwich of iron and lucite plates having the same dimensions as the Cd rods described above.

For use in the high-flux reactor it was planned for these rods to be attached at one end to fuel assemblies so that as a rod is withdrawn fuel is inserted in its place and vice versa. Therefore the measurement of interest was the effect of the rod versus fuel in the same position. The percentage changes in reactivity of the active portion of the core, when these rods were inserted in place of fuel, were as follows:

| ROD | $\Delta k/k$ (%) |
|---|---|
| Cd | 7.3 |
| Cd | 20.7 |
| Th | 4.2 |
| Th | 9.0 |

It was also observed that the reactivity change due to the removal of 180 g of fuel (nine tubes or a 3¼- by 3¼-in. section) from a central position amounted to about 3%.

## A3.6 REFERENCE REPORTS


(1) Martin, A. B. and Mann, M. M., *Critical Experiments for the High Flux Reactor*, ORNL-167, October 22, 1948.

(2) *Quarterly Report of the Physics Division for March, April and May, 1948*, ORNL-51, July 15, 1948.

(3) Mann, M. M. and Martin, A. B., *Further Critical Experiments on a Small Reactor of Enriched U-235/Al-$H_2O$ Moderator and Be Reflector*, ORNL-79, September 16, 1948.

# Appendix 4

## MOCK-UP

### INTRODUCTION

In 1947 when the design of the MTR's predecessor, the High Flux Pile, was nearing completion it was recognized that many features of the reactor would require thorough testing before the design could be considered final. It appeared logical, economically and experimentally, to assemble a full-scale model with which many studies could be made rather than to construct separate equipment for each experiment. It was planned to fabricate major parts (e.g., top plug, aluminum tank, grids, castings) with the intention of using them later, if proved satisfactory, in the actual reactor. The reflector, an item of major importance for hydraulic testing, was to be fabricated of aluminum because beryllium production was still in the development stage.

Two general groups of experiments were made. Studies of a mechanical nature included assembly and alignment techniques, stress-strain investigation of major structural parts, performance of the safety and regulating rod systems, and the development of tools for refueling and remote disassembly of all parts.

Hydraulic studies were made to investigate the adequacy of cooling in the reflector, safety rods, and fuel pieces.

After completion of this "mechanical" program in September, 1949, it appeared that much could be learned by converting the Mock-Up into equipment for low-power nuclear experiments such as critical mass determination, neutron- and gamma-flux measurements, and operation of the control system. These studies occupied the period from February, 1950 to September, 1950.

### A4.1 DESCRIPTION OF FACILITY

In an effort to keep the cost of the project at a minimum, the Mock-Up structure was designed for outdoor operation with the enclosed space limited to the control room. The structure, illustrated in Fig. A4.A, consists of four main steel columns to support the reactor tank and a 10-ton traveling crane. The working level at the top of the tank, 22 ft above grade level, is floored. A 12-ft deep "sub-pile room," 15 by 15 ft, is provided underneath the reactor tank.

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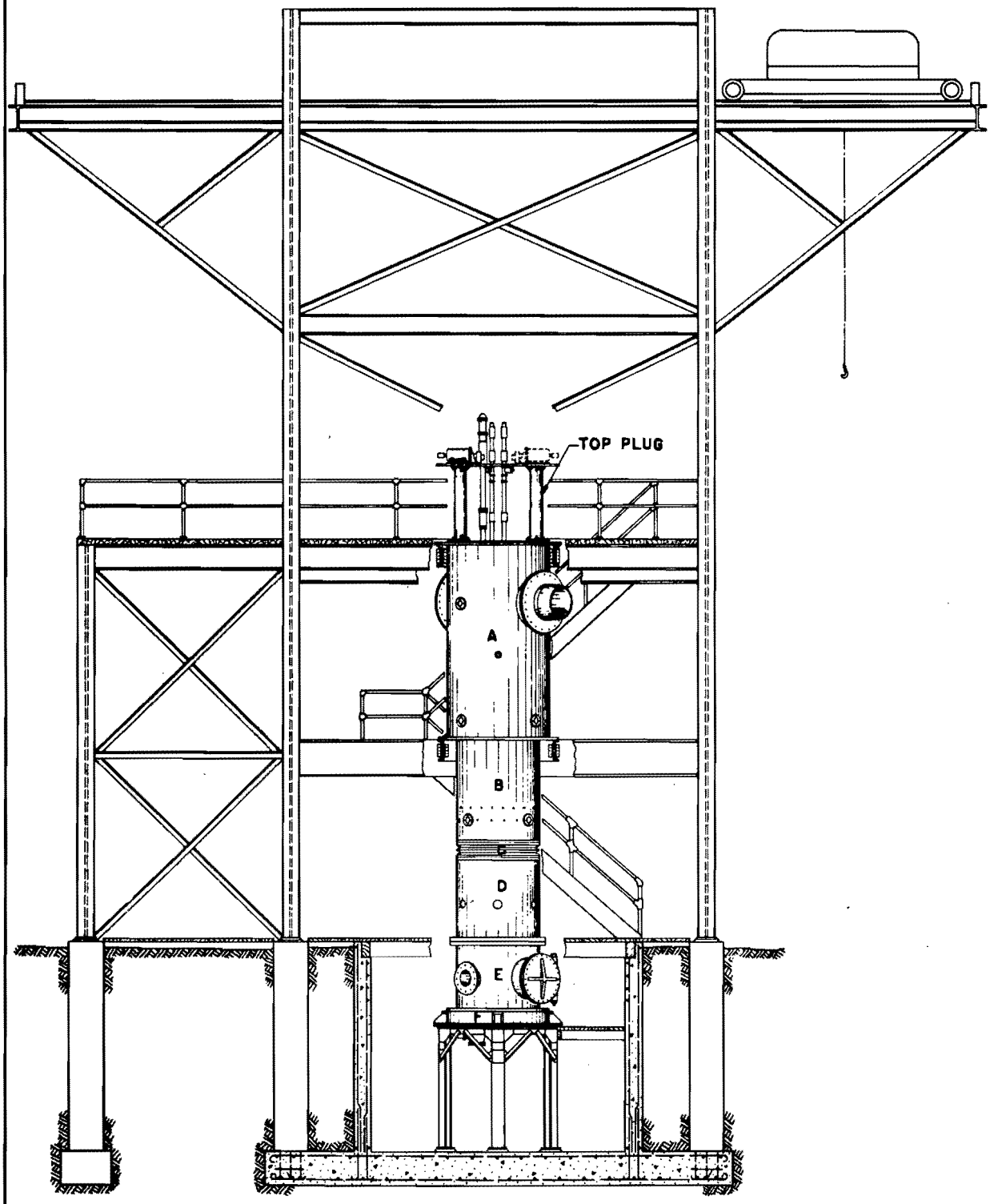


FIG. A4.A

DIAGRAM OF PILE TANK

The water-circulating system was greatly simplified as compared to the MTR. Its only purpose was to provide a maximum flow of 15,000 gpm, and it consisted of two 7500-gpm 100-psi-head pumps which fed the reactor tank through 20-in.-diameter pipe lines. The discharge from the base of the reactor flowed to a 6000-gal capacity surge tank and thence back to the pumps. Six small shell-and-tube heat exchangers were supplied to remove the energy input (about 350 kw) of the pumps. The entire circulating system with the exception of the aluminum section of the reactor tank was constructed of mild steel and coated with Amercoat, a plastic material for corrosion protection.

The reactor tank from top plug to bottom plug was identical to the actual reactor design[1] of that date except for the substitution of Amercoat-covered mild steel for stainless steel at tank sections A, B, E, and F. The top plug was equipped with two safety rods and one regulating-rod drive. All bearings, grids, etc. were supplied for the aluminum tank section. The reflector was made of aluminum, as already mentioned, and the "fuel" was fabricated as uranium-aluminum alloy.

A detailed description of all parts of the Mock-Up may be found in the *Mock-up Design Report*.[2] Information concerning the assembly and erection of the system is presented in *Procedure for Erection of the MTR Mock-Up*.[3]

After approximately six months of study on the dummy reactor, changes were made to allow operation as a nuclear machine intended to duplicate MTR conditions at very low power levels. A detailed account of the alterations made at that time appear in ORNL CF-50-5-140.[5] The alterations consisted essentially in providing the following necessities:

1. Fuel in the form of standard MTR assemblies.
2. A beryllium reflector built with small beryllium bricks to approximate MTR dimensions on two faces of the lattice with Class II reflector pieces ("L" pieces, Be in lattice) on the other two faces.
3. A graphite reflector consisting of 300,000 graphite pebbles and 50 tons of block graphite arranged in three quadrants of the reactor tank.
4. A 4-ft-thick shield of concrete block, unmortared, surrounding the graphite.
5. Instrumentation equivalent to two MTR safety channels and other necessary operating circuits.

These details are shown in Figs. A4.B and C.

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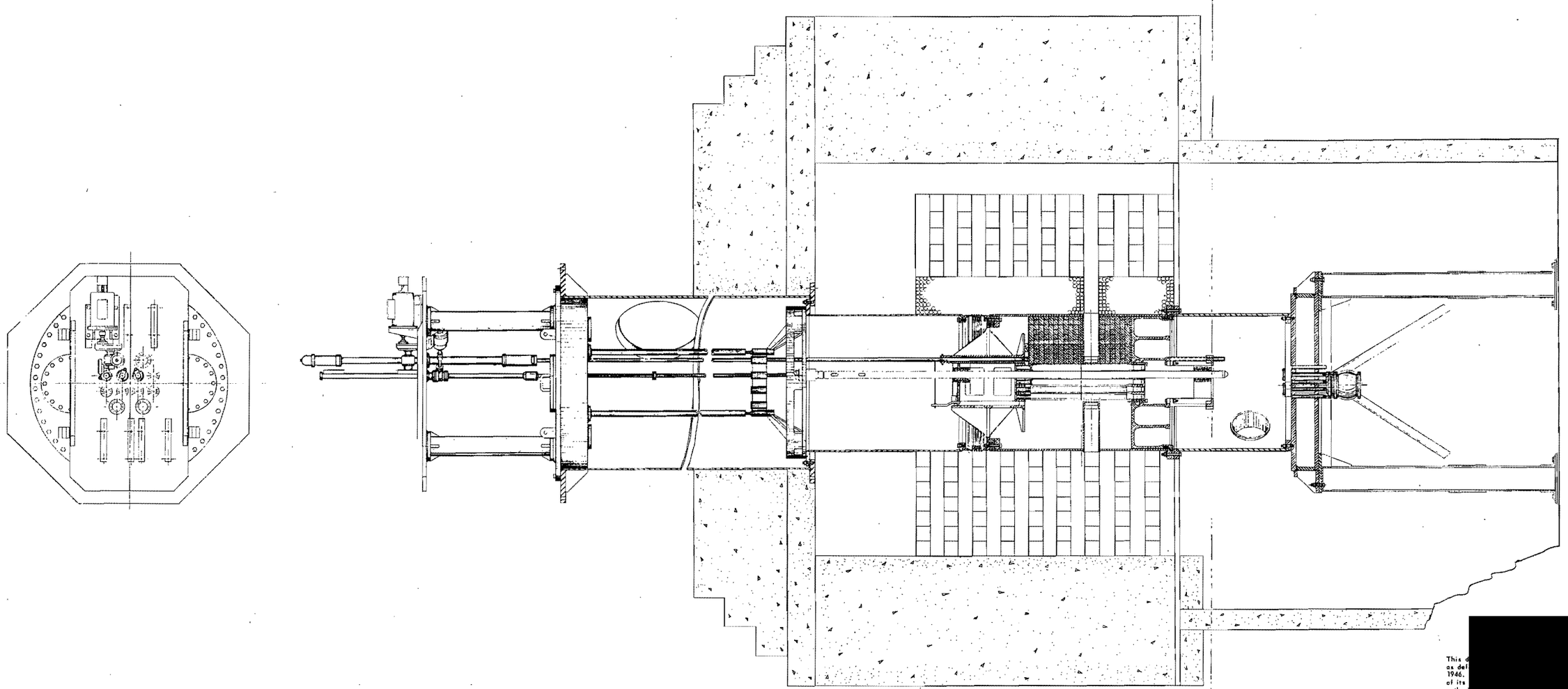

FIG. A.4.B

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TOLERANCE UNLESS OTHERWISE SPECIFIED FRACTIONS ± 1/64 DECIMALS ± .005
BREAK ALL SHARP EDGES UNLESS OTHERWISE SPECIFIED.
FINISH MARKS ASA STANDARD.
DO NOT SCALE THIS DRAWING.

| APPD. | DATE | OAK RIDGE NATIONAL LABORATORY |
|---|---|---|
| | | TECHNICAL DIV — P.O. BOX — OAK RIDGE, TENN |
| | | MTR MOCK-UP ASSEMBLY FOR LOW POWER EXPERIMENTS |
| | | DRAWN BY / DATE / SCALE |
| | | CHECKED BY / DATE / DRAWING NO. TD-1968 / REV. |

| REV. NO. | REVISION | APPD. | DATE |
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Dwg. No. 8916

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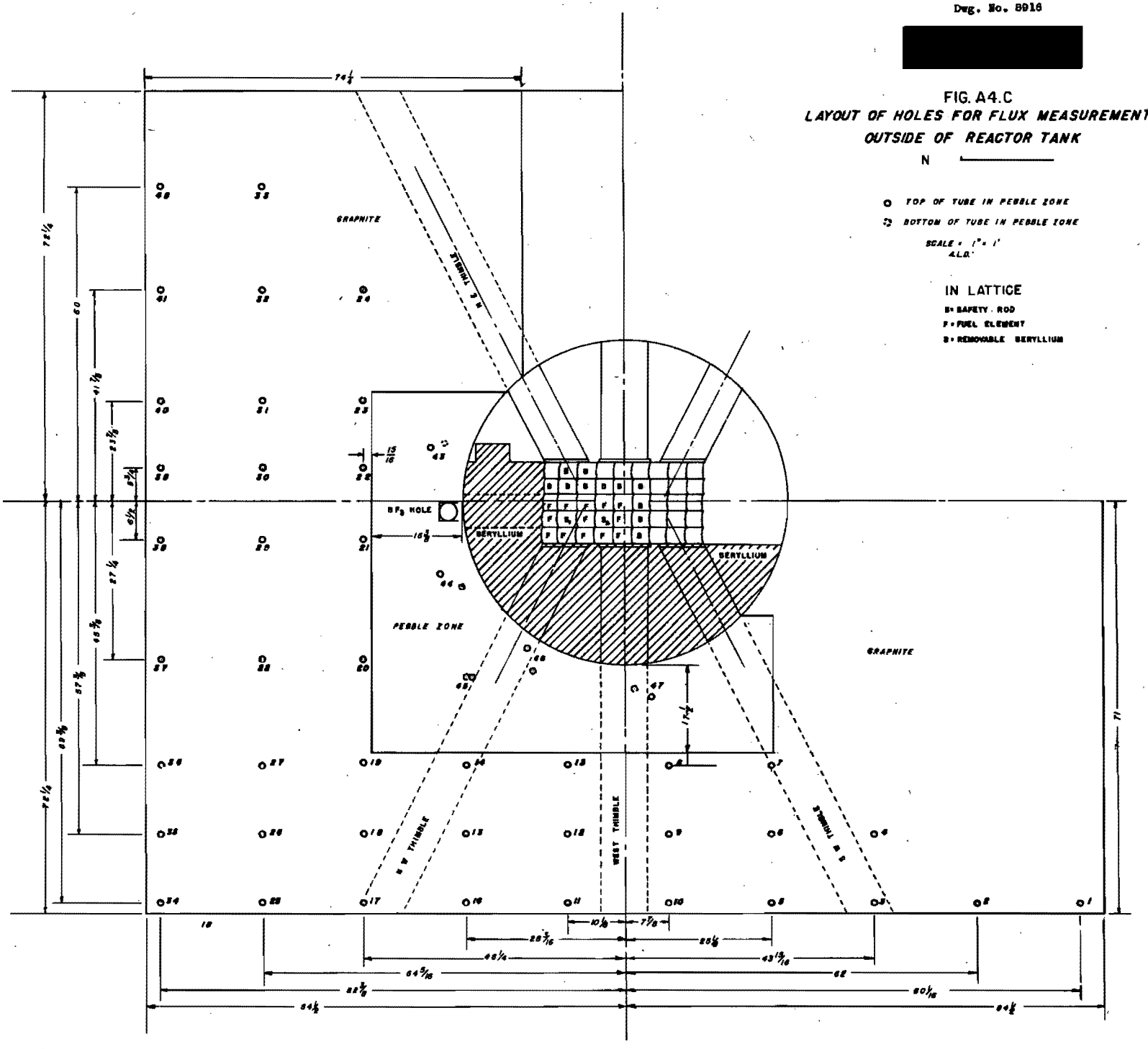
FIG. A4.C

*LAYOUT OF HOLES FOR FLUX MEASUREMENTS OUTSIDE OF REACTOR TANK*

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## A4.2 MECHANICAL EXPERIMENTS

The mechanical and hydraulic investigations were begun in May, 1949 and consisted of the following:

1. Stress-strain measurements.
2. Mechanical test of the control system.
3. Development of remote-handling tools.
4. Hydraulic measurements in lattice and reflector.

**A4.2.1 Stress-Strain Measurements.** The complexity of tank structure D with respect to stress analysis made it desirable to obtain measurements at questionable points in the Mock-Up. The required information was obtained from Baldwin-Southwork strain gauges attached directly to the various components of the tank. The results of these measurements are reported in ORNL-472.[4]

The electric resistance wire elements were attached according to the following plan:

| | NO. OF ELEMENTS |
|---|---|
| Tank section D, outside surface | 14 |
| Skirt plates | 4 |
| Upper assembly grid | 2 |
| Upper support casting | 5 |
| Lower support casting | 5 |

Roughly half of the strain gauges were single elements which were employed at locations where the direction of the stress could be easily predicted. Where the direction was unknown, a triangular arrangement of three elements was used so that both the magnitude and the direction of the stress could be determined.

Measurements were made early in the experimental program and before the full pressure drop across the lattice was applied. However, the magnitude of the stresses calculated from the strain measurements were low enough in all cases to provide assurance that no component was in danger of being overstressed even at much higher pressure drops.

Maximum stresses for the components during operation at about 30 psi $\Delta P$ were:

| LOCATION | STRESS (psi) |
|---|---|
| Tank section D, 6 in. below upper flange | 1920 (tension) |
| Skirt plate, center | 1850 (tension) |
| Upper assembly grid, center | 950 (compression) |
| Upper support casting, center bottom | 1300 (compression) |
| Lower support casting (side, center of grid opening) | 2020 (compression) |

The 356-T7A aluminum alloy with which the castings were made has a yield strength of 25,000 psi at 0.2% set. Tank section D was fabricated of 3SF aluminum with a tensile strength of 17,850 psi and a minimum yield strength of 6,000 psi.

**A4.2.2 Control System.** Except for the control system, which includes both shim-safety and regulating rods the reactor has practically no moving parts. Because this system is roughly the brain and nervous system of the reactor and should operate for its lifetime it was important that the control mechanics be tested very thoroughly.

For these reasons the Mock Up was provided with two complete shim-safety rod channels and one regulating rod channel. The control room contained the operating console which served as the study model for the MTR desk and a relay panel, both equipped with the necessary electrical equipment for testing system performance.

*Console.* The console for the Mock Up was little different from that planned for the MTR. A few switches were relocated and the functions of one or two were changed The wings to the right and left of the operator were moved inward to bring them within easier reach. Except for these minor changes, the design was considered adequate and was approved for the MTR.

*Shim-Safety Rods and Drive Mechanisms* [6,7] The shim-safety rods and drive mechanisms constituting the original Mock Up controls were considerably different from the elements currently designed for the MTR. A brief history of the evolution of the present MTR shim safety assemblies will be given for each part of the assembly.

(a) Shim-Safety Rods. Originally the rods consisted of five separate sections which were, from top to bottom: armature section, upper stainless steel section, cadmium (or thorium) section, fuel (or beryllium) section, and lower stainless steel section. These rods were expensive (about $3,000 each) to manufacture considering their estimated life (one month). In the course of testing the rods several possible refinements to the original design became apparent.

In the armature section the cadmium plating was not sufficient protection to prevent rusting of the 1010 steel. Nickel plating was substituted for better corrosion protection. The nickel plating, being magnetic, had the additional advantage of reducing the magnet-to-armature air gap, allowing a gain of about 25 lb in magnet holding force. The original sharp-edged corners of the armature made it difficult to position the upper assembly of grids over the rods, and as a result the corners were beveled to make this operation easier.

The upper section of the rod serves three purposes: (1) as bearing surface for the guide bearings, (2) as inlet port for cooling water, and (3) as a "spacer" to keep electromagnets outside the intense neutron-flux region. Being necessarily long, this stainless steel section was also heavy (about 50 lb). Mass at this point and in the absorber section below is undesirable because of the large force exerted on the comparatively weak uranium-aluminum section during deceleration of the rod. In fact, after repeated drops the aluminum was found to be deformed at its joints and the machine screws holding these joints were loosened. It was apparent that the upper sections of the rod should be made of lighter material, and an aluminum extrusion with a cadmium box insert was designed to replace the upper stainless steel and cadmium sections.

Although this weight reduction was desirable from the standpoint of damage to the fuel section and joint strength, it was considered necessary to keep the total weight of the rod above 100 lb for the situation requiring a scram with no water flow. Experiments showed that buoyancy and bearing-friction forces were great enough to keep a rod weighing less than 70 lb from falling if the 350-lb driving force due to water pressure were absent.[8] To eliminate this danger lead was added to the lower stainless steel section, and guide bearings were redesigned to minimize friction.

Velocity measurements on the original rods showed a peak speed of 16 ft/sec with full water flow (see Fig. A4.D). An acceleration of about 4g is indicated, and these data should hold for present MTR rods.

The Mock-Up was equipped with safety-rod shock absorbers with bellows-sealed plungers at the base to indicate the presence of the rod. The indicating function is unrelated to the primary purpose of the shock absorber, that of decelerating the falling rod, but is important from the standpoint of control. The plunger type indicator gave constant trouble by water leakage to the limit switches underneath and from galling. Hydraulic actuators were designed for the MTR but could not be tested in the Mock-Up because of limited time.

The operation of the dash pots as decelerators is illustrated by the curve of Fig. A4.E which shows a terminal rod velocity of 3 ft/sec. Maximum pressures were measured for various entrance velocities and found to be 200 psi for the 16-ft/sec velocity resulting from a 10-psi pressure drop across the reactor tank.

The bearings which serve to guide the upper and lower ends of the rods originally contained spring-loaded graphite pads as the bearing surfaces. The graphite scored easily and was found to offer a resistance of approximately 40 lb to the fall of the rods. Stainless steel rollers with graphite inserts have been designed to replace the graphite and have proved adequate during several months of usage.

(*b*) Electromagnets. The initial design of the electromagnets was based on the requirement that they operate under a water pressure of 50 psi, support 1000 lb load at 200 mamp current, and release this load in less than 30 msec (the minimum period of which the reactor is capable in a start-up accident). After test units had been constructed, it was found that the release times were about double those expected and that the watertight seals were inadequate.

The release time deficiency was traced to shorted turns in the coils. Subsequent coils were redesigned and manufactured by the Westinghouse Electric Corporation. In the course of the investigation, the coils were tested on core materials of Permandure and Hypersil alloys in addition to the 1010 steel chosen for the final design. With properly manufactured coils on the 1010 steel cores, release times of considerably less than 30 msec were measured.

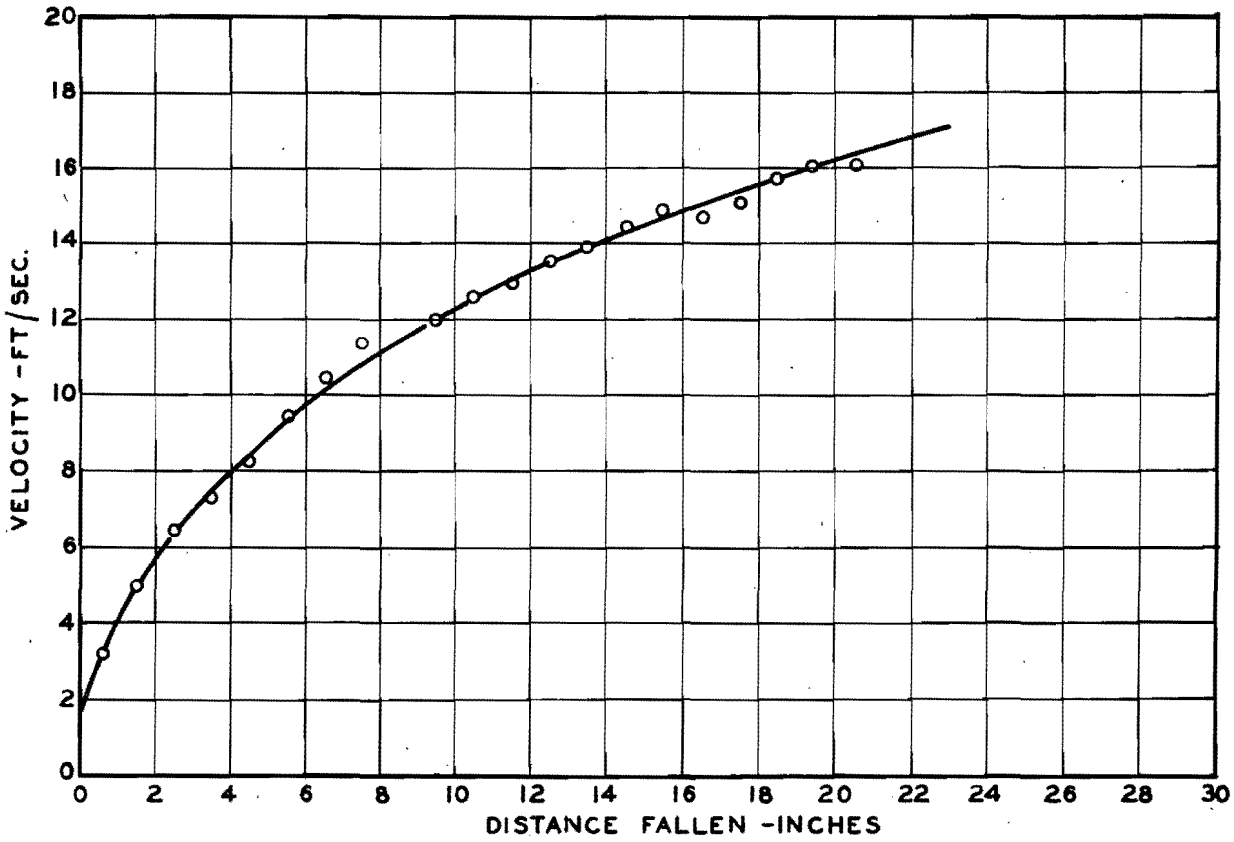



FIG. A.4.D
RATE OF FALL - SAFETY ROD
40 PSI ΔP

DWG. # 10805
NOT CLASSIFIED

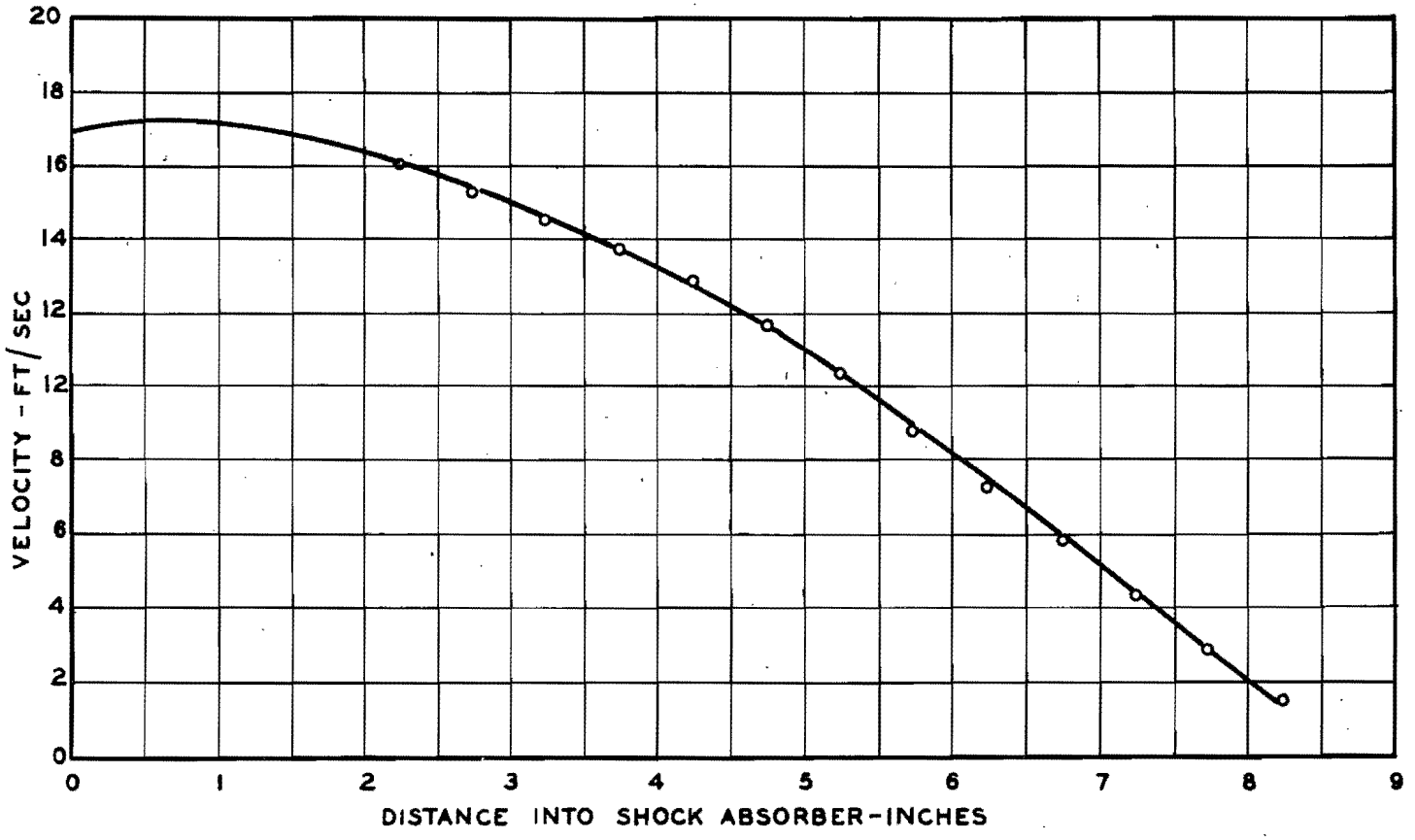



FIG. A. 4.E
DECELERATION OF SAFETY ROD

DWG. #10804
NOT CLASSIFIED

In fact if one minimizes the ratio of holding force to force actually held by varying the current, the armature-to-pole-face air gap, or the load supported, the release time can be reduced to 2 or 3 msec. The net result of much experimentation with these factors was to specify a current of 100 mamp and an air gap of about 0.010 in. which can be adjusted to give a holding force of 600 lb. For the case of full water flow, the total load on the magnets is about 500 lb and a release time of about 10 msec can be expected. With no water flowing and only the dead weight of the rod (about 100 lb) supported, the release time is 30 to 40 msec. The holding-force curve for a typical magnet is presented in Fig. A4.F. In Fig. A4.G are the records of two release-time measurements.

The failure of the gasketed water seals on the pole pieces of the initial design led to a complete enclosure of the core with a stainless steel can. This change necessitated removal of a microswitch and a bellows-sealed actuator which were contained inside the old can for the purpose of indicating physical contact with the shim-safety rod. The indicator currently in use takes advantage of the fact that the rods are at ground potential electrically and the demineralized water has a high resistance. By insulating the magnets from the drive shafts and applying a 6-volt potential, the current which flows when the magnet face contacts the rods is sufficient to trip a relay which indicates contact and energizes the magnets.

The magnets installed at the Mock-Up were fastened to the drive shafts by six ½-in. bolts. Calculations indicated that the magnets would be considerably above tolerance radiation after long-time operation at MTR levels. To simplify the replacement of faulty magnets a quick-release mechanism has been adopted for MTP magnets.

The requirement for an insulating and shielding liquid with which magnets and drive shafts could be filled is that the liquid not suffer radiation damage (i.e., loss of insulating qualities or severe degassing), that it be heavier than water (so that leakage water will not accumulate above the magnets), and that it not attack the magnet construction materials. The liquid now used is the only insulator of many tested which adequately fills these requirements.

The flexible connections now incorporated in the shim-safety rod design were previously a section of the drive shafts, immediately above the magnets.

DWG. #10806
NOT CLASSIFIED

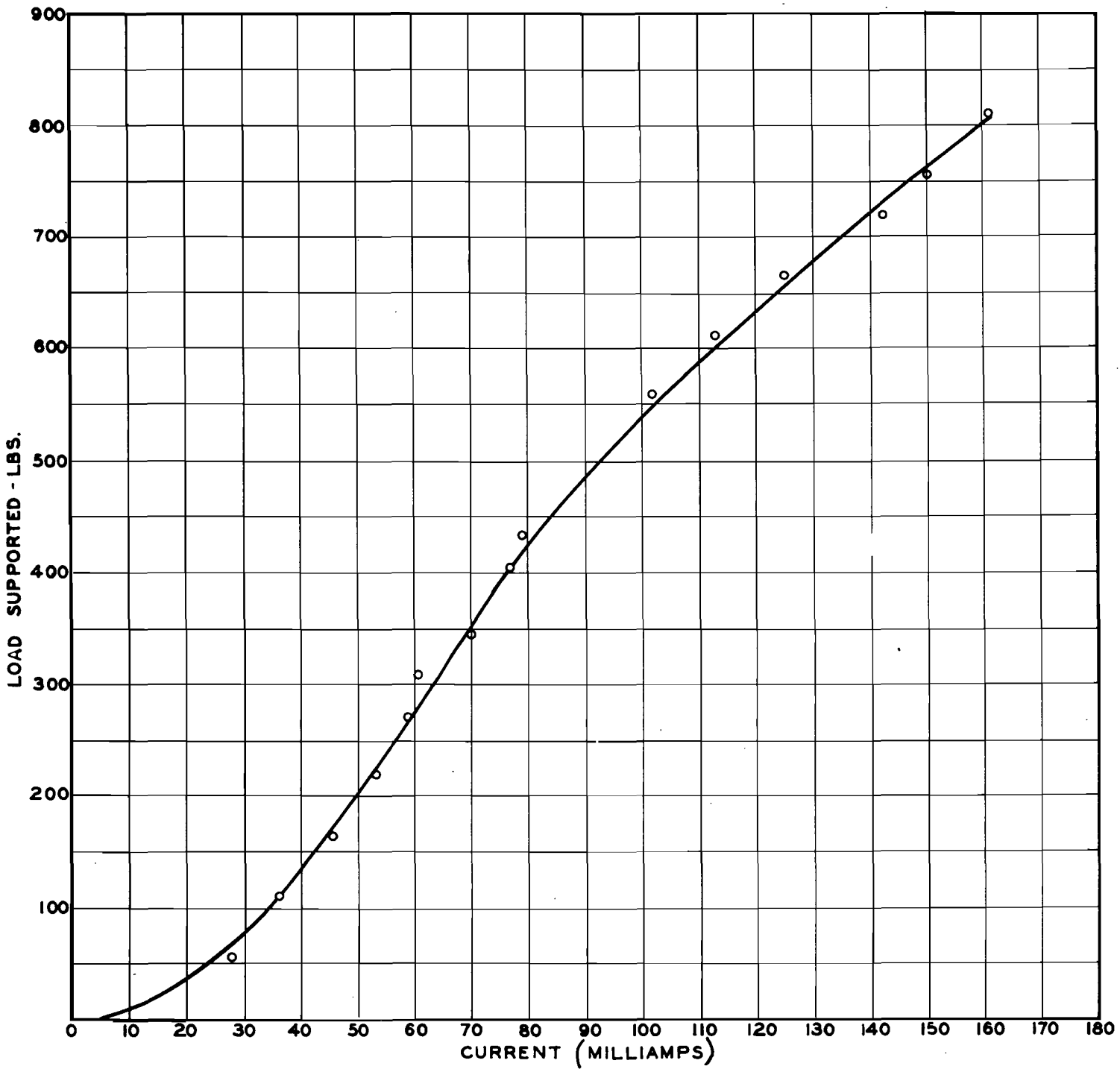



FIG. A. 4.F

WESTINGHOUSE BUILT MTR MAGNET #1
LOAD VS RELEASE CURRENT

0.002 MICA SHIM
0.008 STAINLESS STEEL SHIM

DWG. #10811
NOT CLASSIFIED

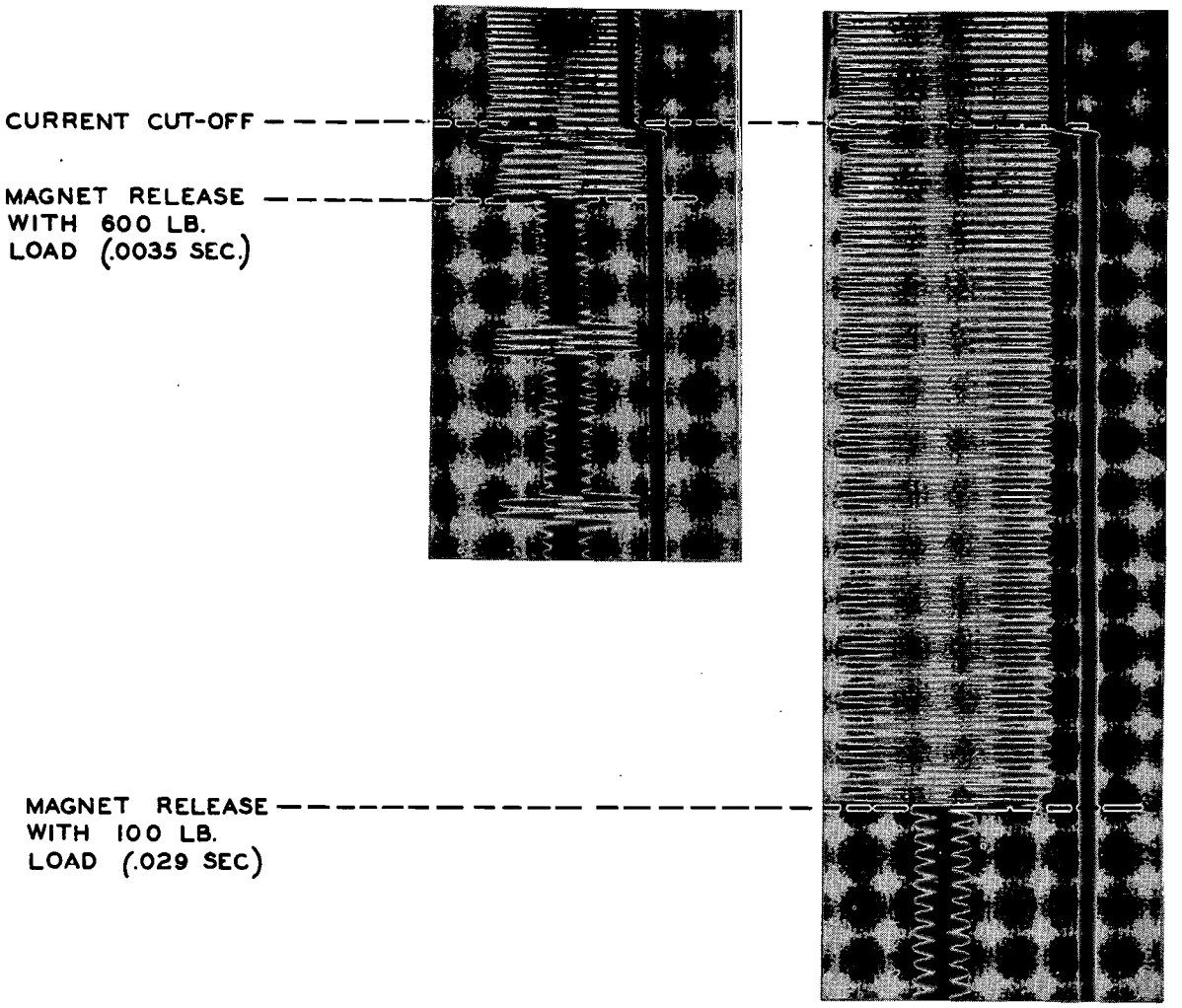



FIG. A.4.G

TYPICAL MAGNET RELEASE TESTS

*(2000 cps Time Trace)*

Their purpose is to correct misalignment and to allow approximately 3/8-in. overtravel of the drive motors after current cut-off. The original design allowed excessive movement of the magnets under full water flow and gave considerable difficulty through abrasion of the magnet leads. The simpler joint now at the top of the safety rod appears to be adequate.

(*c*) Drive Mechanisms. The worm gear--acme thread speed reduction units were studied with respect to measurement of backlash, overtravel, wear, actual speed reduction, and time of reversal. Total backlash from motor shaft to magnet drive shaft was less than 0.010 in. Overtravel was 0.10 in. upward and 0.17 in. downward. The actual speed of rod withdrawal was 5.0 in./min, and time of reversal was less than 0.1 sec. All measurements were made with shafts loaded to 600 lb. After approximately 400 hr of operation the total backlash had not increased, and it was assumed that no serious wear had resulted.

*Regulating Rod and Drive Mechanism.*[9] The MTR regulating-rod and drive mechanism assembly are very similar to the originals installed in the Mock-Up. The Mock-Up rod was thoroughly tested from standpoints of general operability, mechanical design, and reactor control.

Some difficulty was experienced with the sleeve-guide bearing located in the upper and lower support castings. Because of grit particles the rod and bearings galled severely, necessitating removal and refinishing. The bearings were redesigned as spring-loaded stainless steel rollers which eliminated the galling trouble. However, during the low-power nuclear experiments with full water flow it was learned that the bearings permitted lateral movement of the rods, which affected neutron detection instruments as would a minor instability of the reactor. The low-power experiments were completed before stronger springs could be tested.

Another difficulty with the original design was the method of attaching the rod to its drive shaft. The coupling was designed as a screw joint to be actuated by a 20-ft-long bolt after removal of the upper shock absorber. Rotation of the rod during the coupling operation was prevented by a horizontal pin through the rod which mated with a slot in the upper bearing. Awkward features were eliminated by a redesign including a new upper shock absorber which need not be removed to accomplish attachment or detachment of the rod; a ball-and-socket joint replaced the threaded joint, and the necessity for a pin and slot were thus eliminated.

The lower shock absorber of the drive mechanism exhibited an annoying quirk of not returning to its rest position after compression. This trouble was corrected by insertion of a stronger spring and by attachment of a small check valve at the oil-fill port.

Before the regulating rod was used with its servo mechanism as a control device, experiments were devised to measure its response time. The test consisted in unbalancing the electronic system with an applied voltage (instead of an ion chamber signal) and measuring the time required for rebalancing. The response time for the maximum unbalance was found to be about 0.1 sec and is considered adequate for control of known MTR instabilities.

Full-scale test of the servo mechanism as an automatic control device was accomplished during the low-power nuclear experiments described in Section A4.5.7. The system performed satisfactorily for several months at power levels from 100 watts to 175 kilowatts.

## A4.3 HYDRAULIC EXPERIMENTS

The high-power density (about 300 kw/liter) at which the MTR operates imposes rather stringent requirements for heat removal. Cooling-water velocities of 30 ft/sec and heat fluxes as high as 500,000 Btu/sq ft-hr are necessary for the realization of design performance.

The desire for more complete knowledge of the complex hydraulic characteristics of the MTR was an important reason for construction of the full-scale Mock-Up.

Early design calculations indicated that the following conditions could be expected in the reflector and fuel lattice:

| LOCATION | VELOCITY (ft/sec) | PRESSURE DROP (psi) |
|---|---|---|
| (1) 1/8-in.-diameter holes in reflector pieces | 10 | 6 |
| (2) 0.118-in. slot between fuel elements | 25 | 17 |
| (3) Fuel elements with 0.118-in. gaps | 30 | 42 (including end boxes) |
| (4) Shim-safety rods (Th-U) | 30 | 42 |
| (5) Across upper assembly grid | | 12 |
| (6) Across lower assembly grid | | 13 |
| (7) Across lower support casting | | 36 |

Water requirements for 42 psi pressure drop were:

| ELEMENT | FLOW OF WATER (gpm) |
|---|---|
| 23 fuel pieces (slab loading) | 10,500 |
| 8 shim-safety rods | 3,200 |
| Reflector | 1,000 |
| Lattice interstices and Be in lattice | 1,500 |
| Total water requirement | 16,200 |

The Mock-Up problem assignment was to determine whether the calculated pressure drops and flow rates actually existed.

The experimental work was divided into two groups: (1) flow in the reflector, and (2) flow inside the fuel lattice.

**A4.3.1 Flow in Reflector Outside the Fuel Lattice.** The reactor is designed in a manner such that the beryllium reflector occupies the space (15 by 30 in.) between the rectangular lattice area at the center and the wall of the 54-in.-diameter aluminum tank. At the time the Mock-Up was built the reflector consisted of approximately two hundred separate pieces, each roughly 3 by 3 by 40 in. high. In general, each separate piece was provided with several 1/8-in.-diameter cooling holes extending through the piece from top to bottom. A few pieces of odd sizes and shapes, such as those above and below the experimental thimbles, had larger or different cooling passages. The general aim in the design of the reflector cooling passages was to provide sufficient water to remove approximately 6,000 kw of heat produced by $\gamma$-ray and neutron energy from the fuel lattice.

The problem of measuring the various velocities and pressures was approached by selecting a typical quadrant of the reflector for detailed study. A few pieces were chosen in other quadrants as check points and others where special situations arose. Figures A4.H and I show these selections. One cooling channel in each reflector piece was provided with two static pressure taps and calibrated for flow versus pressure drop before installation.

After positioning a particular piece in the Mock-Up reflector, small stainless steel tubing leads from the pressure points were attached to Cannon

[redacted]

Drawing #7641

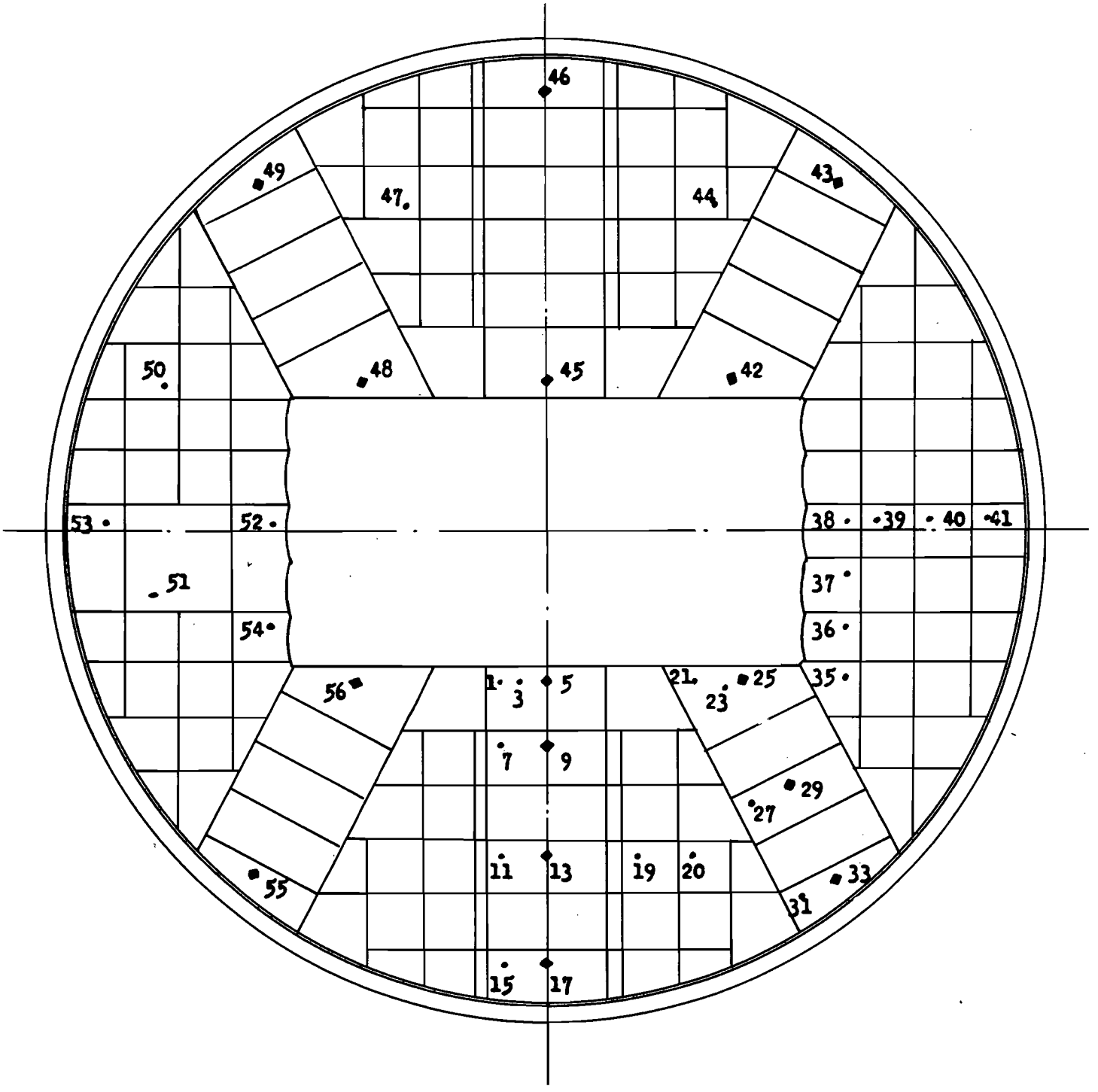



Fig. A 4. H

LOCATION OF PRESSURE TAPS ABOVE THIMBLES

(Reference: Dwg TD-500)

[redacted]

Drawing 7642

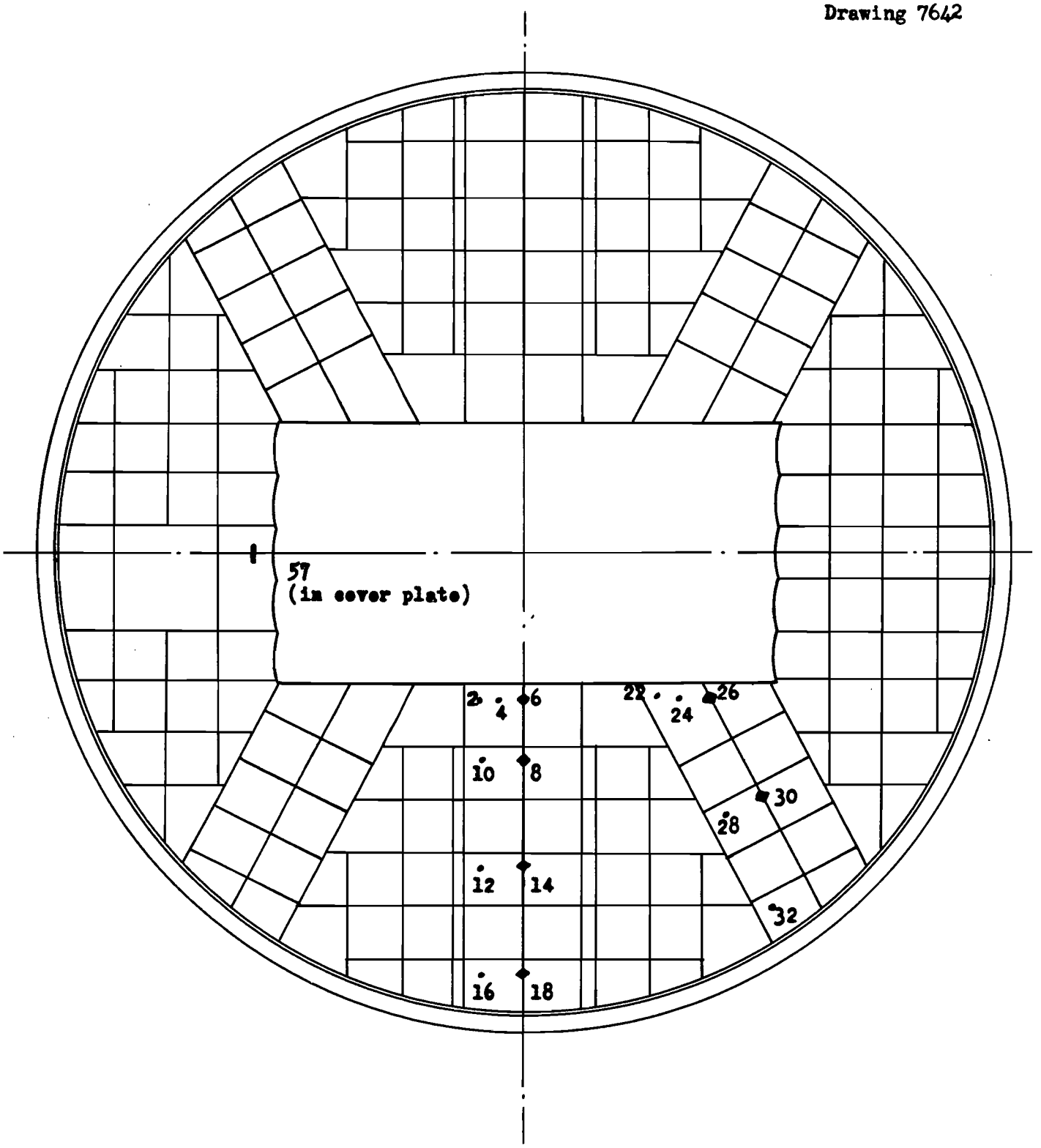



Fig. A 4. I
LOCATION OF PRESSURE TAPS BELOW THIMBLES

(Reference: Dwg TD-502)



[redacted]

type connectors in the reactor tank wall and then run directly to a manometer board in the control room. Approximately 120 lines were connected from inside the reactor tank to manifolds and thence to mercury and oil-filled Merriam manometers.

Nine experimental determinations of velocities were made in the 56 reflector pieces selected for study at pressure drops ranging from 17.5 to 33.7 psi. After two runs it was seen that flow in the reflector was considerably in excess of that anticipated. An unsuccessful attempt was made to reduce the flow by plugging weep holes in the lower support casting and filling some open area at the "A" piece end box sockets. Mean velocities in various types of channels for a typical test at 33.6 psi, after reducing the open area in the lower support casting, are shown in Table A4.A.

**TABLE A4.A**

**Water Velocities in Various Reflector Channels**

| CHANNEL | VELOCITY (ft/sec) |
|---|---|
| "A" pieces (1/8-in. holes) | 18.7 |
| 5/16-in. holes above thimbles | 9.0 |
| 5/16-in. holes below thimbles | 8.9 |
| 1/8-in. holes above thimbles | 10.9 |
| 1/8-in. holes below thimbles | 3.8 |
| 1/8-in. holes above ½-in. horizontal tubes | 15.2 |

Using these velocities as a basis, the total flow rate through the reflector was computed as follows:

A section was taken above the thimbles and just below the top surface of the reflector. Counting the numbers of various types of holes and using measured dimensions, flow rates in gallons per minute for each foot per second measured velocity were derived. These are shown in Table A4.B.

## TABLE A4.B

### Flow Rates for Various Reflector Holes

| TYPE OF CHANNEL | NO. OF HOLES | CROSS-SECTION (sq in.) | FLOW RATE (gpm) AT 1 ft/sec |
|---|---|---|---|
| 1/8-in. holes above thimbles | 320 | 0.0129 | 12.9 |
| 5/16-in. holes above thimbles | 86 | 0.101 | 27.1 |
| Full length 1/8-in. holes | 744 | 0.0129 | 30.0 |
| 1/8-in. holes in "A" pieces | 208 | 0.0129 | 8.4 |
| 1/8-in. holes above ½-in. horizontal tubes | 44 | 0.0129 | 1.8 |
| 1/8-in. holes above 1-in. horizontal tubes | 64 | 0.0129 | 2.6 |

Flow through gaps between reflector components was estimated as a function of gap thickness. Assuming that the velocities vary as the 2/3 power of the equivalent diameters, and that the equivalent diameter of a wide thin slot is approximately twice the thickness $T$,

$$V_2 = V_1 \left[\frac{D_2}{D_1}\right]^{2/3} = V_1 \left[\frac{2T_2}{D_1}\right]^{2/3}$$

The 0.127-in. hole designated as pressure tap 41 (Fig. A4.D) is suitable as a standard. Then

$$V_2 = V_1 \left[\frac{2}{0.127}\right]^{2/3} (T_2)^{2/3} = 6.30\ V_1 (T_2)^{2/3}$$

Values of flow rates in gallons per minute at 1 ft/sec measured velocity at pressure tap 41 are thus derived for the gaps between reflector pieces. These values are shown in Table A4.C.

## TABLE A4.C

### Flow Rates at Pressure Tap 41

| CLEARANCE, $T_2$ | NO. OF GAPS | $(T_2)^{2/3}$ | CROSS-SECTION (sq in.) | FLOW RATE (gpm) AT 1 ft/sec |
|---|---|---|---|---|
| 0.017 | 228 | 0.065 | 0.051 | 14.9 |
| 0.0435 | 6 | 0.122 | 0.130 | 1.9 |
| 0.060 | 20 | 0.153 | 0.180 | 10.9 |
| 0.080 | 12 | 0.185 | 0.240 | 10.5 |
| 0.1065 | 2 | 0.225 | 0.320 | 2.8 |
| 0.0625 (at tank wall) | 1 | 0.158 | 10.6 | 33.0 |
| Total | | | | 74.0 |

Components of the reflector flow rate are computed in Table A4.D, using the factors derived above and the measured velocities in Table A4.A.

## TABLE A4.D

### Flow Rates in Reflector

| | FLOW RATE (gpm) AT 1 ft/sec | TOTAL FLOW (gpm) | | | |
|---|---|---|---|---|---|
| | | TEST 6 $\Delta P$ 33.5 psi | TEST 7 $\Delta P$ 33.5 psi | TEST 8 $\Delta P$ 33.7 psi | TEST 9 $\Delta P$ 33.6 psi |
| Components of flow rate | | | | | |
| 1/8-in. holes above thimbles | 12.9 | 138 | 140 | 138 | 141 |
| 5/16-in. holes above thimbles | 27.1 | 244 | 264 | 255 | 244 |
| Full length 1/8-in. holes | 30.0 | 362 | 362 | 362 | 372 |
| 1/8-in. holes in "A" pieces | 8.4 | 154 | 152 | 152 | 149 |
| 1/8-in. holes above ½-in. horizontal tubes | 1.8 | 25 | 28 | 30 | 27 |
| 1/8-in. holes above 1-in. horizontal tubes | 2.6 | 33 | 34 | 30 | 30 |
| Subtotal | | 956 | 980 | 967 | 963 |
| Clearance channels | 74.0 | 898 | 898 | 898 | 920 |
| Total reflector flow | | 1854 | 1878 | 1865 | 1883 |

During these experiments measurements were also made of total flow and velocities through fuel elements (see Section A4.3.2, Lattice Measurements). A typical flow balance at 33.5 psi pressure drop was as follows:

| | |
|---|---|
| Total flow, by calibrated orifice (gpm) | 14,900 |
| Component flow, by calibrated pressure taps (gpm) | |
| 1. 19 fuel elements (27.5 ft/sec velocity) | 8,300 |
| 2. Shim rods and substitutes | 2,890 |
| 3. "Beryllium" lattice pieces | 70 |
| 4. Lattice interstices | 1,577 |
| 5. Open 1-in. rabbit tubes | 220 |
| Total flow, by components except reflector (gpm) | 13,057 |
| Flow through reflector, by difference (gpm) | 1,843 |

This excellent agreement of two independent measures indicates accuracy better than the ±1% which more probably existed.

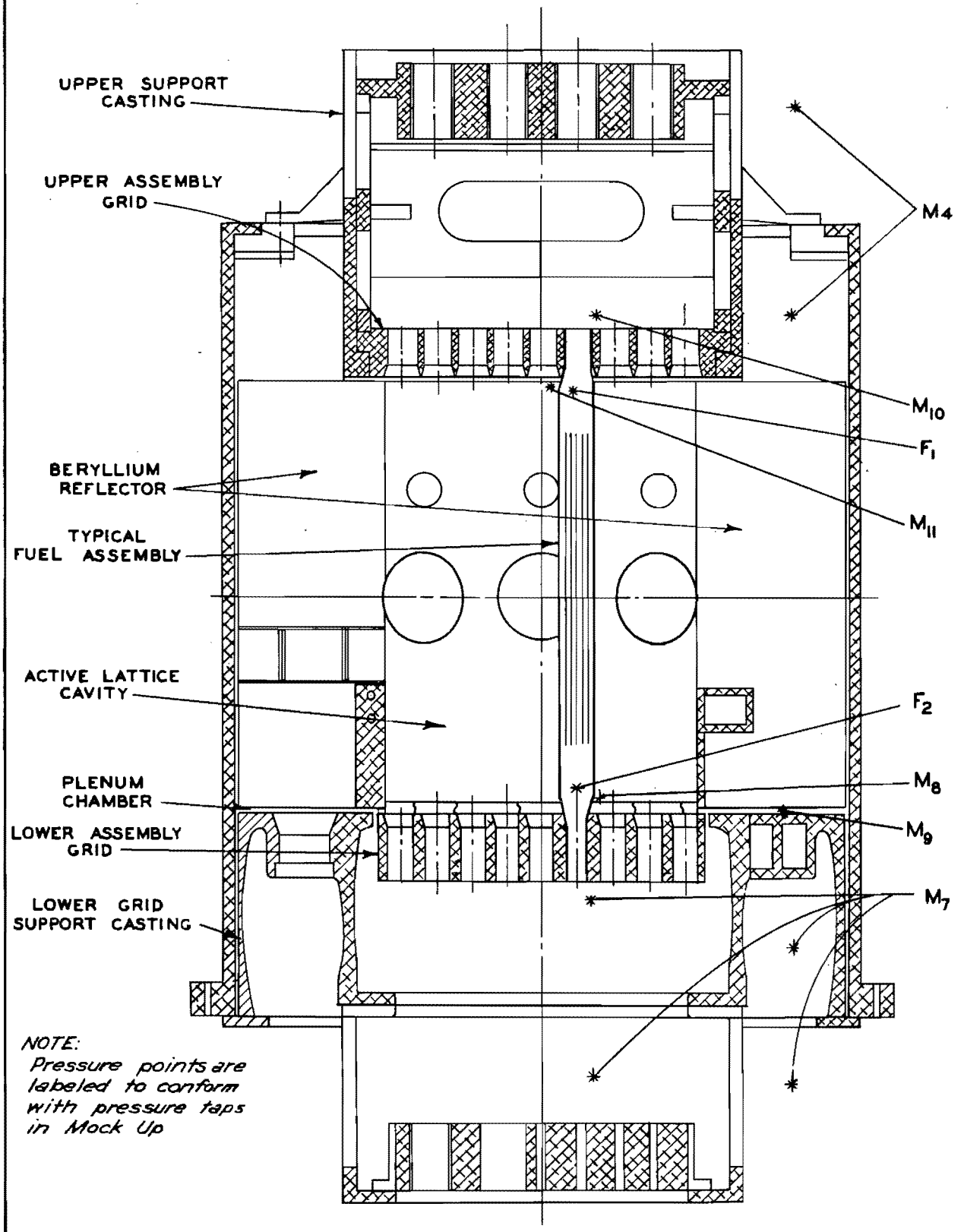
On the basis of these data and simultaneous lattice measurements (Section A4.3.2), which indicated that a pressure drop of approximately 40 psi would be required to realize a 30-ft/sec velocity through the fuel, reflector requirements at 40 psi were estimated as approximately 2000 gpm. This figure was roughly double the amount of water originally allowed for the reflector area. For this and for other perhaps more important reasons, a redesign of the reflector was undertaken.

**A4.3.2 Flow and Pressure Drops Inside Fuel Lattice.** Velocity measurements were taken inside fuel elements and shim-safety rods and in the interstices between fuel elements. Pressure drops were studied wherever significant reductions in static pressure were thought to occur. The points selected for measurement are shown in Fig. A4.J. An analysis of these data is presented in ORNL CF-50-6-102.[10]

*Flow Through Fuel Elements.* A group of six fuel elements was selected for calibration. Two static pressure taps were installed in each of these assemblies and used to measure pressure drop as a function of velocity through the fuel plates. Calibrations were obtained in a separate experiment before the fuel pieces were installed in the reactor lattice. Figure A4.K is a plot of the calibration of these assemblies. Also shown are data from a previous

[redacted]

Drawing # 9101




*NOTE:*
*Pressure points are labeled to conform with pressure taps in Mock Up*

FIG. A4.J
CROSS SECTION OF MTR

DWG. #10810
~~SECRET~~

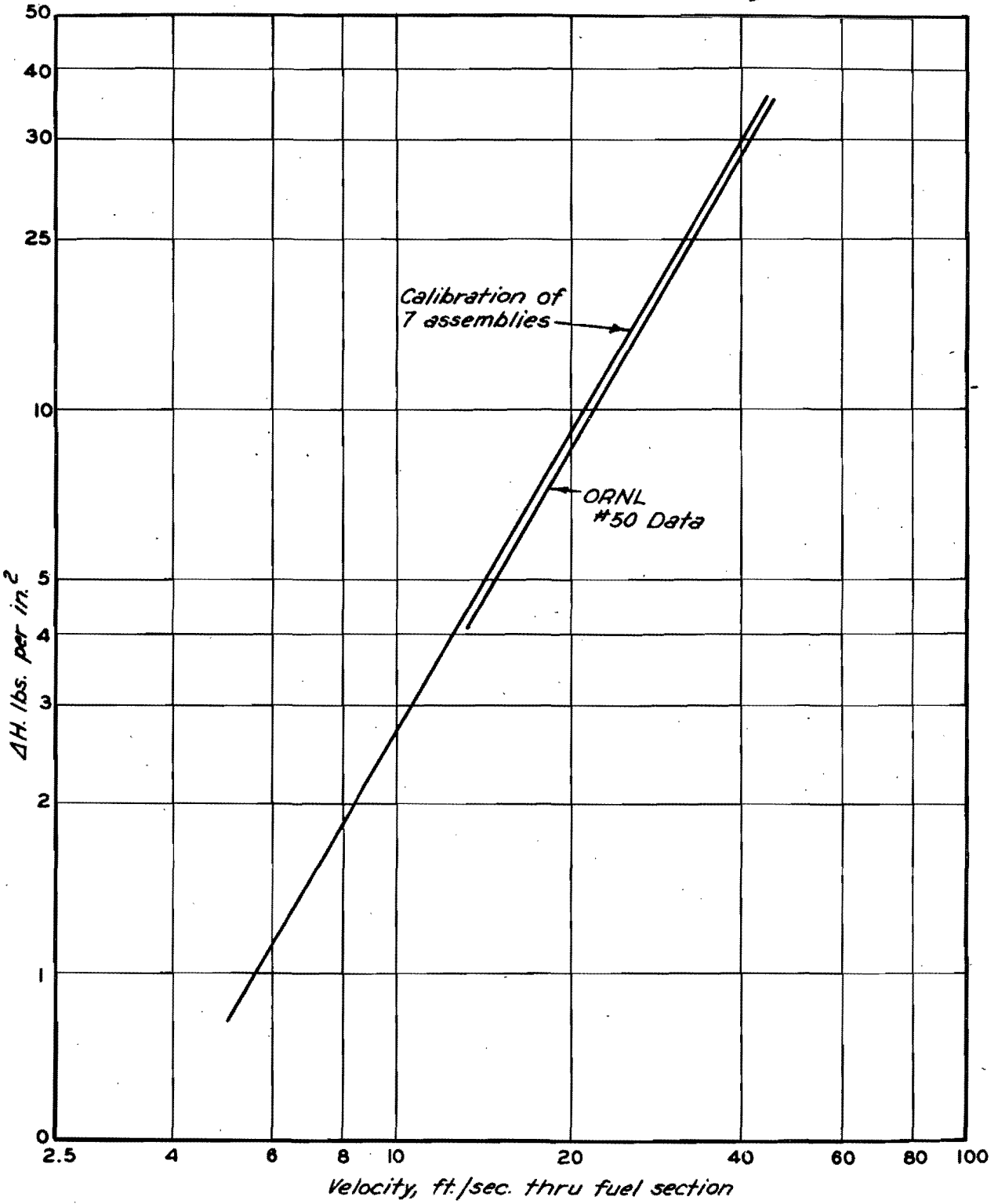



FIG. A.4.K
VELOCITY - PRESSURE DROP CURVE FOR FUEL ASSEMBLIES

calibration of a single assembly reported in ORNL-50. The pressure drops were measured over the length of the fuel plates and the agreement of the two independent calibrations is good.

The overall pressure drop, which includes losses in both upper and lower end boxes (reported in ORNL-50), indicated that the desired velocity of 30 ft/sec past the plates would be realized at a total pressure drop of 30 to 35 psi. This value was taken as the basis for design of the hydraulic system for the MTR. However, full-scale measurements in the Mock-Up indicated that an overall pressure drop of about 40 psi was required to attain the 30-ft/sec velocity.

This diagreement of 5 to 10 psi was the subject of an investigation which resulted in an explanation of the difference in the two cases. It was found that the upper end box of the fuel piece used in the experiments described in ORNL-50 had an inside diameter 1/8 in. longer than the end boxes of more recent design used in the Mock-Up. A calculation of the two cases confirmed a loss of about 4 psi for the smaller Mock-Up end boxes. An additional loss of 3 psi was measured across the upper support casting, through which the water must pass before it reaches the fuel. Thus the total design pressure drop for the reactor tank was increased to 40 psi, which may be broken down as follows:

| RESISTANCE | PRESSURE DROP (psi) |
|---|---|
| Upper support and grid spacer castings | 3 |
| Upper end box | 13 |
| Fuel plates | 18 |
| Lower end box | 6 |
| Total pressure loss in tank section D | 40 |

*Flow Through Interstices Between Fuel Elements.* In order for cooling water to flow between adjacent fuel elements, it must move past the upper assembly grid, down past the outer fuel plates, and discharge through the lower assembly grid. It was desired to maintain a velocity of 25 ft/sec in the spaces between fuel elements, but the Mock-Up measurements indicated

velocities of roughly 15 ft/sec. It was found that the pressure drop across the upper assembly grid was about 10 psi and across the lower assembly grid was 18 psi. Subsequently the open area in the lower grid was increased and its pressure drop reduced to 10 psi to give the correct velocity through the interstices.

*Flow Through Safety Rods.* The shim-safety rods also require a cooling velocity of 30 ft/sec through the fuel section. Measurements in a previous experiment[11] indicated a pressure drop of 40 psi for this velocity. Calibration of the Mock-Up rods agreed with this figure. The curve of velocity versus pressure drop is presented in Fig. A4.L.

On the basis of the Mock-Up measurements, the pressure drops and flow rates through tank section D are summarized in Table A4.E.

**TABLE A4.E**

**Pressure Drops and Flow Rates Through Tank Section D**

| LOCATION | PRESSURE DROP (psi) | FLOW (gpm) | |
|---|---|---|---|
| | | COMPONENTS PARALLEL | COMPONENTS NONPARALLEL |
| Upper support casting | 3 | | 17,670 |
| Upper assembly grid | 10 | | 2,500 |
| Fuel assemblies (23 required) | 37 | 10,925 | |
| Lattice beryllium pieces (14 required) | 37 | 35 | |
| Gaps in lattice | 17 | 2,500 | |
| Shim rods, cadmium (4 required) | 40 | 1,725 | |
| Shim rods, beryllium (4 required) | 40 | 10 | |
| Reflector (Mock-Up design) | 9 | 2,000 | |
| Lower support casting | 31 | | 2,000 |
| Lower assembly grid | 10 | | 2,500 |
| By-pass | | 475 | |
| Total flow required (with 4 cadmium shim rods) | | 17,670 | |

DWG. #10809
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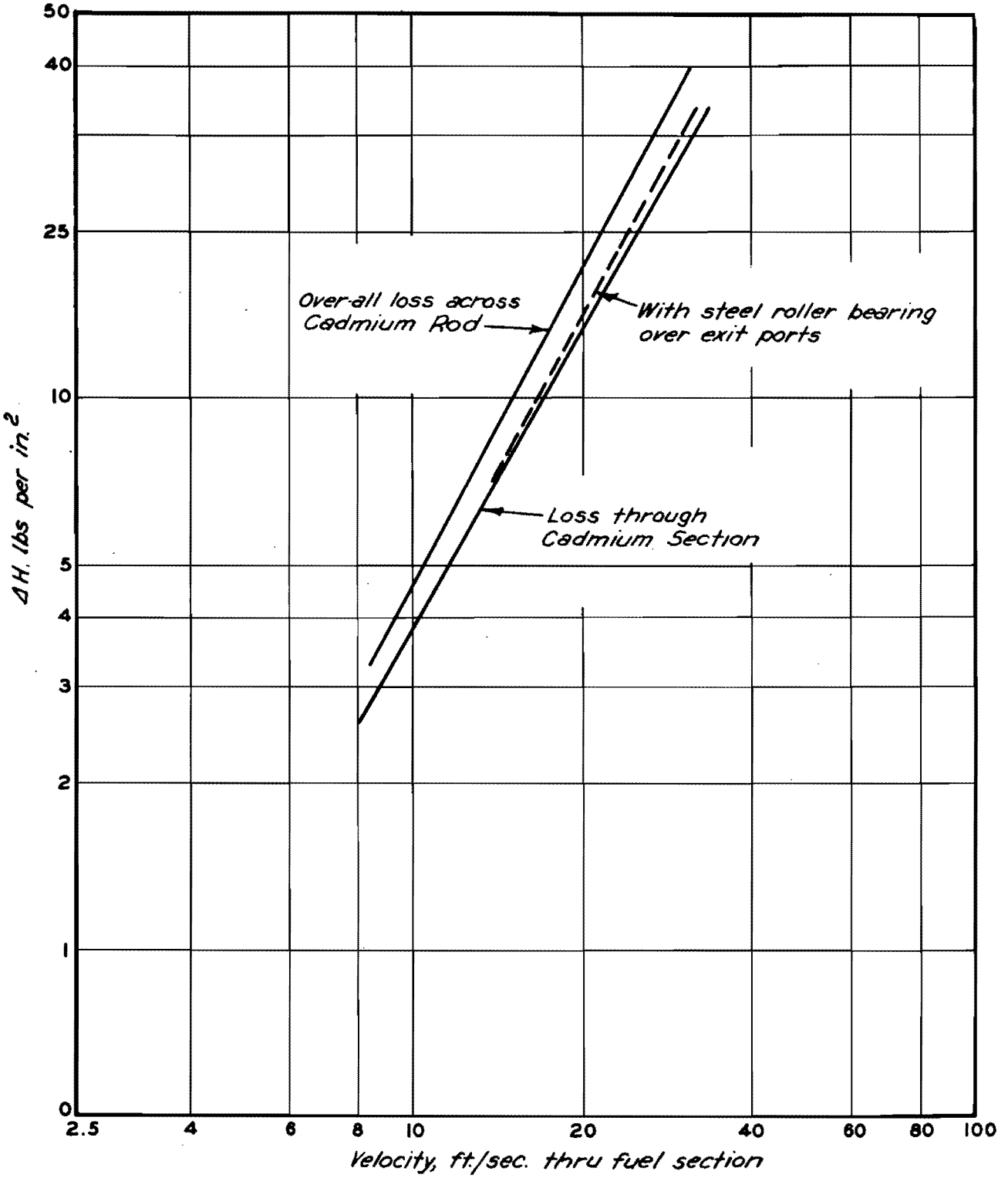



FIG. A.4.L

CALIBRATION OF CADMIUM SHIM SAFETY ROD

NEW TYPE - 0.060" FUEL PLATES

## A4.4 HANDLING TOOLS

Normal and frequent operations such as fuel replacement and adjustment of reactor and reflector geometry for experiments require special tools and appurtenances for remote handling. It will be recalled, furthermore, that one of the basic design requirements of the MTR is that any part of the reactor proper be replaceable. Because many parts will be intensely radioactive after extended operation, remote handling is required. Working models for most of the operations noted have been tested in the Mock-Up; Figs. A4.M and N are shown as typical of this work.

The results of Mock-Up experience have been incorporated in the MTR handling tools mentioned in Section 2.9.

## A4.5 NUCLEAR EXPERIMENTS

Problems associated with the nuclear characteristics of the operating reactor were studied for a period of seven months (starting in February, 1950) using the assembly illustrated in Figs. A4.B and C. As is often the case when a heavy program is undertaken with limited time available, it was impossible to refine the experiments as desired. However, from the MTR design standpoint little could have been gained by greater precision because MTR experimental conditions could not be exactly duplicated. The principal topics of study were the following:

1. Determination of critical mass for a slab type loading.
2. Calibration of regulating rod.
3. Measurements of temperature coefficient.
4. Measurements of reactivity changes caused by water flow, air bubbles, and open beam holes.
5. Performance of the control and safety systems.
6. Measurements of neutron and gamma fluxes inside and above the fuel, in the graphite reflector outside the tank, and in the thermal shield.

**A4.5.1 Critical Mass Determination.** Several extraordinary safety precautions were taken for the initial fuel loading. For instance, a standard safety rod was rebuilt with a beryllium section in place of the fuel section

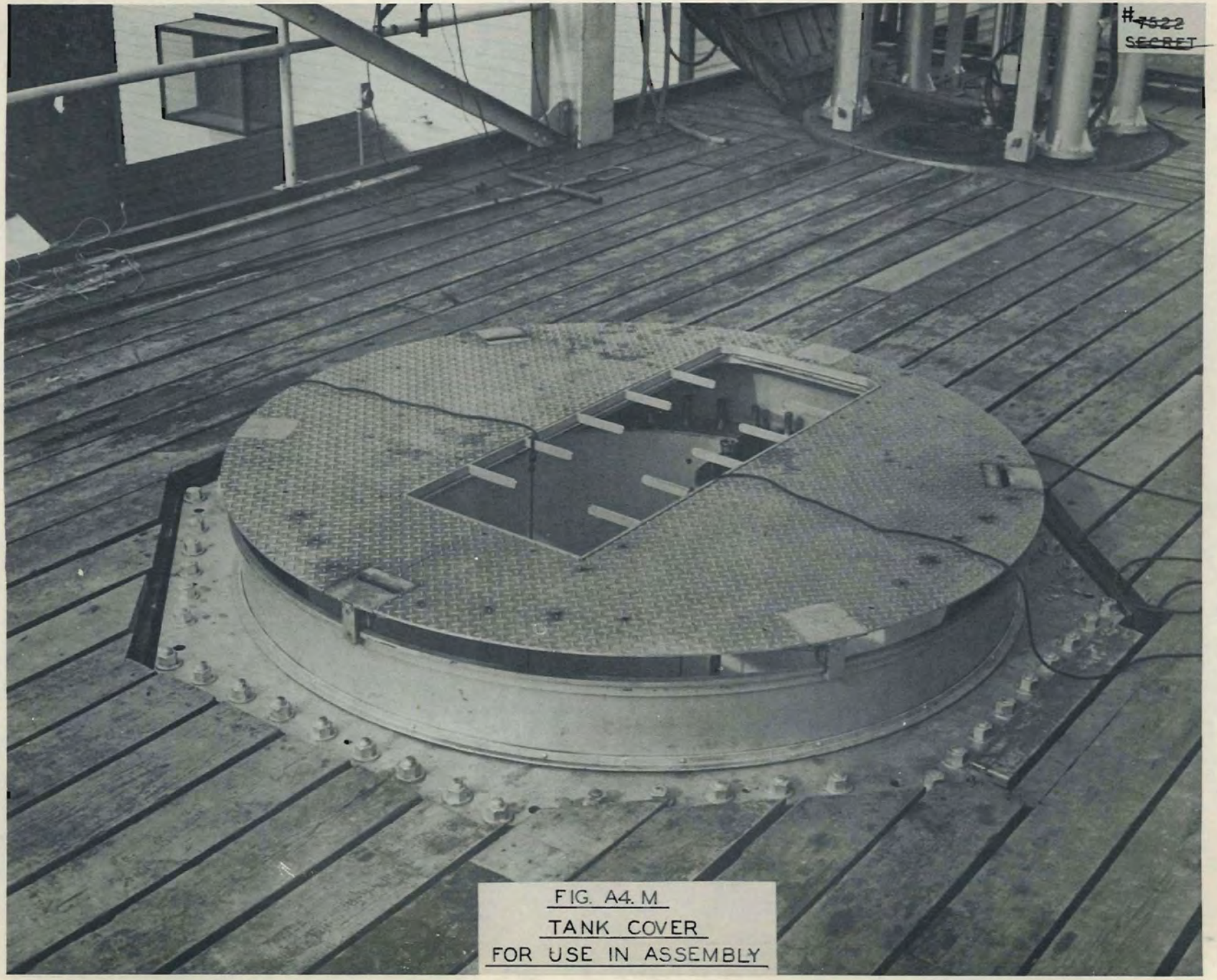



FIG. A4. M
TANK COVER
FOR USE IN ASSEMBLY

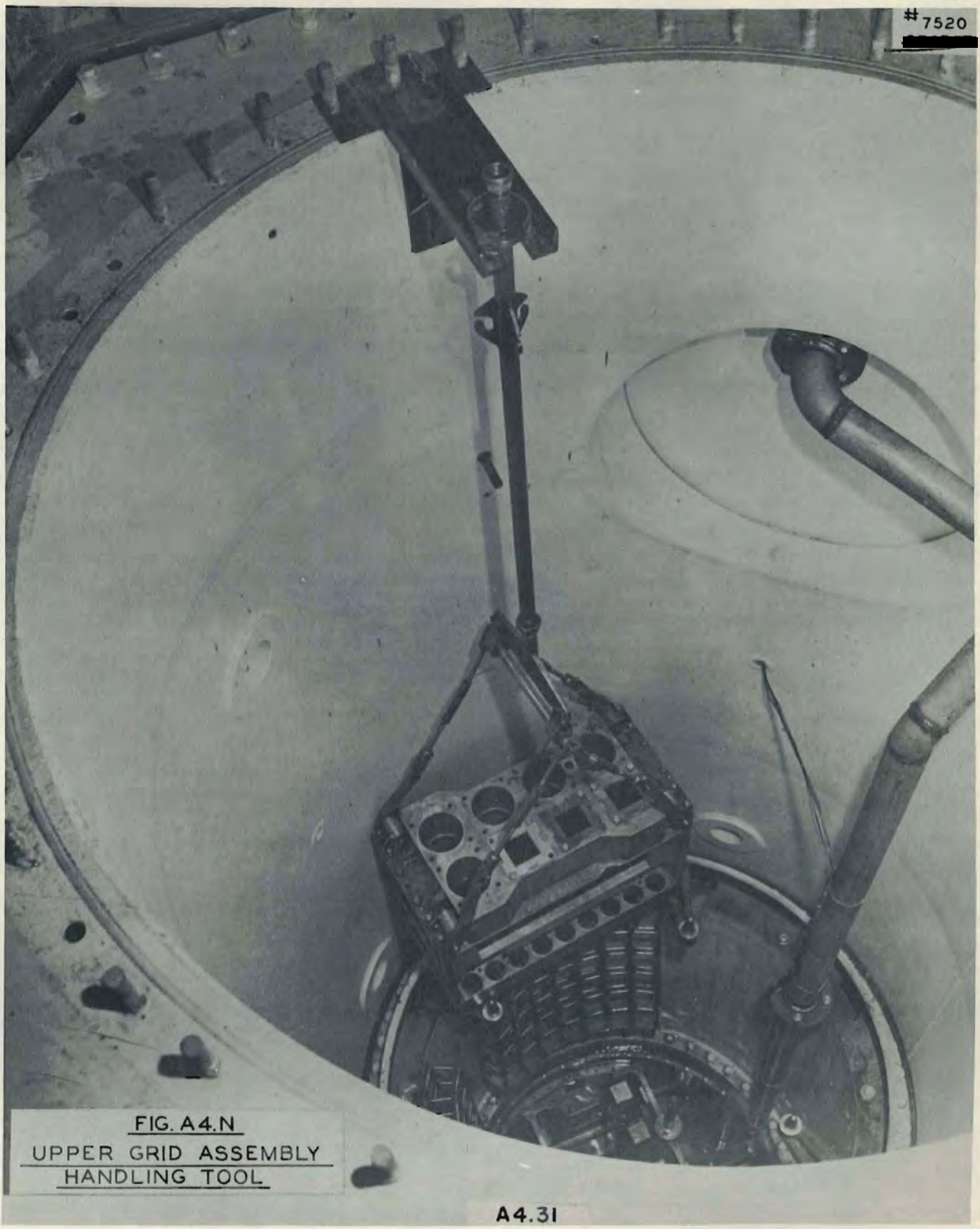



FIG. A4.N
UPPER GRID ASSEMBLY
HANDLING TOOL

for use as a "cocked" safety during loading. The cadmium section of this rod was kept suspended above the lattice by an electromagnet while each fuel piece was inserted.

Another precaution was the installation of separate neutron-level and period indicators at the loading platform which was 30 ft above the control room. Also, a boron fluoride neutron detector was used in addition to the slow fission chamber and three other neutron chambers. The relative locations of source, fuel, and detectors is shown in Fig. A4.O.

The critical mass determination was begun after recording backgrounds of all the neutron detectors with the addition of four fuel pieces containing a total of 560 g of $U^{235}$. The control rods were then withdrawn completely and the new counting rates were measured with pulse counters. The usual plot of reciprocal counting rate against mass of $U^{235}$ was made upon addition of each additional fuel element (140 g of $U^{235}$). As the critical point was approached, smaller quantities of uranium in the form of partial fuel elements were added. Criticality was attained with approximately 2000 g of $U^{235}$.

**A4.5.2 Regulating-rod Calibration.** The regulating rod is a 1½-in.-diameter aluminum rod with a 20-in.-long cadmium tube inserted at its center; it is located in the permanent beryllium at the edge of the lattice. The total reactivity of the rod, expressed as $\Delta k/k$, is intended to be less than the delayed neutron fraction to prevent sudden complete removal from making the reactor "prompt-critical." For safety, therefore, it was necessary to be certain of the total reactivity change possible from the regulating rod. Furthermore, for control purposes it was desirable to determine a curve of sensitivity as a function of rod position.

The method of calibration was as follows: Beginning with the rod completely inserted and the reactor level (infinite period), the rod was withdrawn a short distance manually and the resultant period was measured as the slope of the logarithmic plot of neutron level. After complete withdrawal of the rod, similar points were obtained by stepwise insertion. The reactivity, $(\Delta k/k)_{eff}$, was obtained from the usual in-hour-relation.[12] Curves of sensitivity and total reactivity are presented in Figs. A4.P and Q.

Considering the accuracy of the determination (about ±5%), it is seen that the original rod was possibly worth as much as 0.7% in $(\Delta k/k)_{eff}$.

DWG. #10808 [redacted]

TANK FILLED WITH WATER

B- REMOVABLE BERYLLIUM
F- FUEL ELEMENT - 140 gms. U235
S- SAFETY ROD

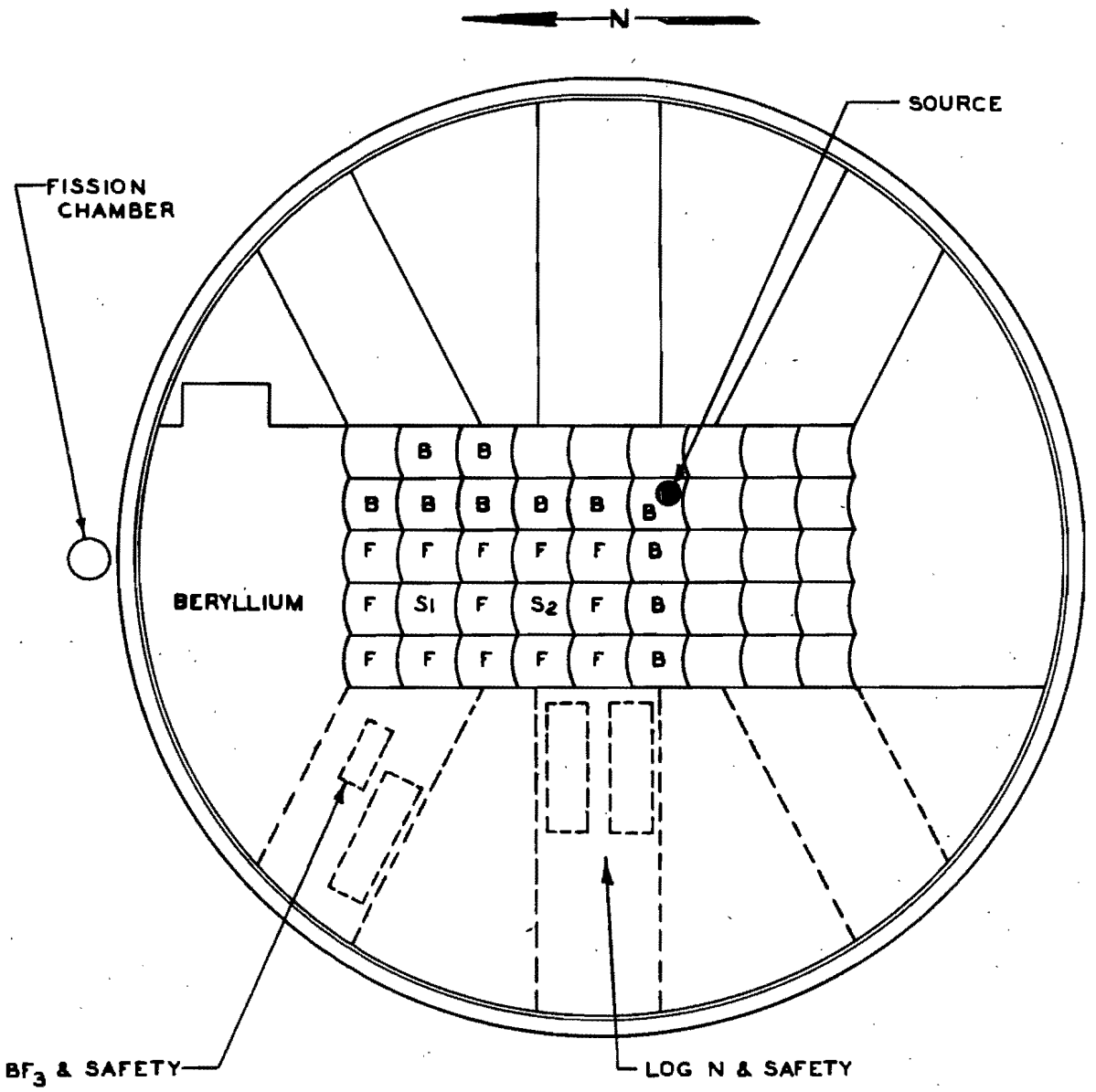



FIG. A.4.0

REACTOR TANK SHOWING ARRANGEMENT OF LATTICE AND BERYLLIUM REFLECTOR

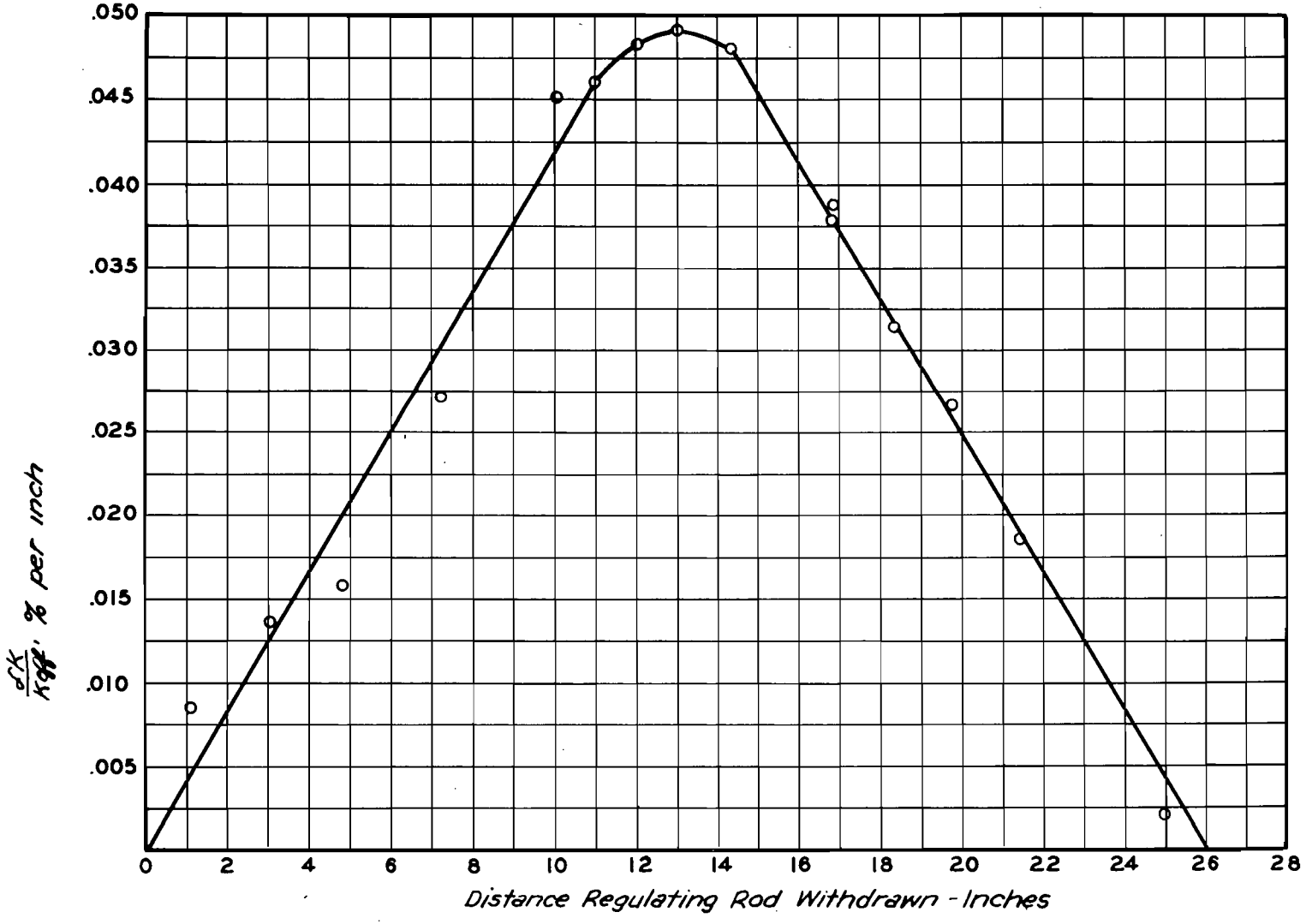



FIG. A.4.P

CALIBRATION OF REGULATING ROD

DWG. #10812

DWG. # 10813 [redacted]

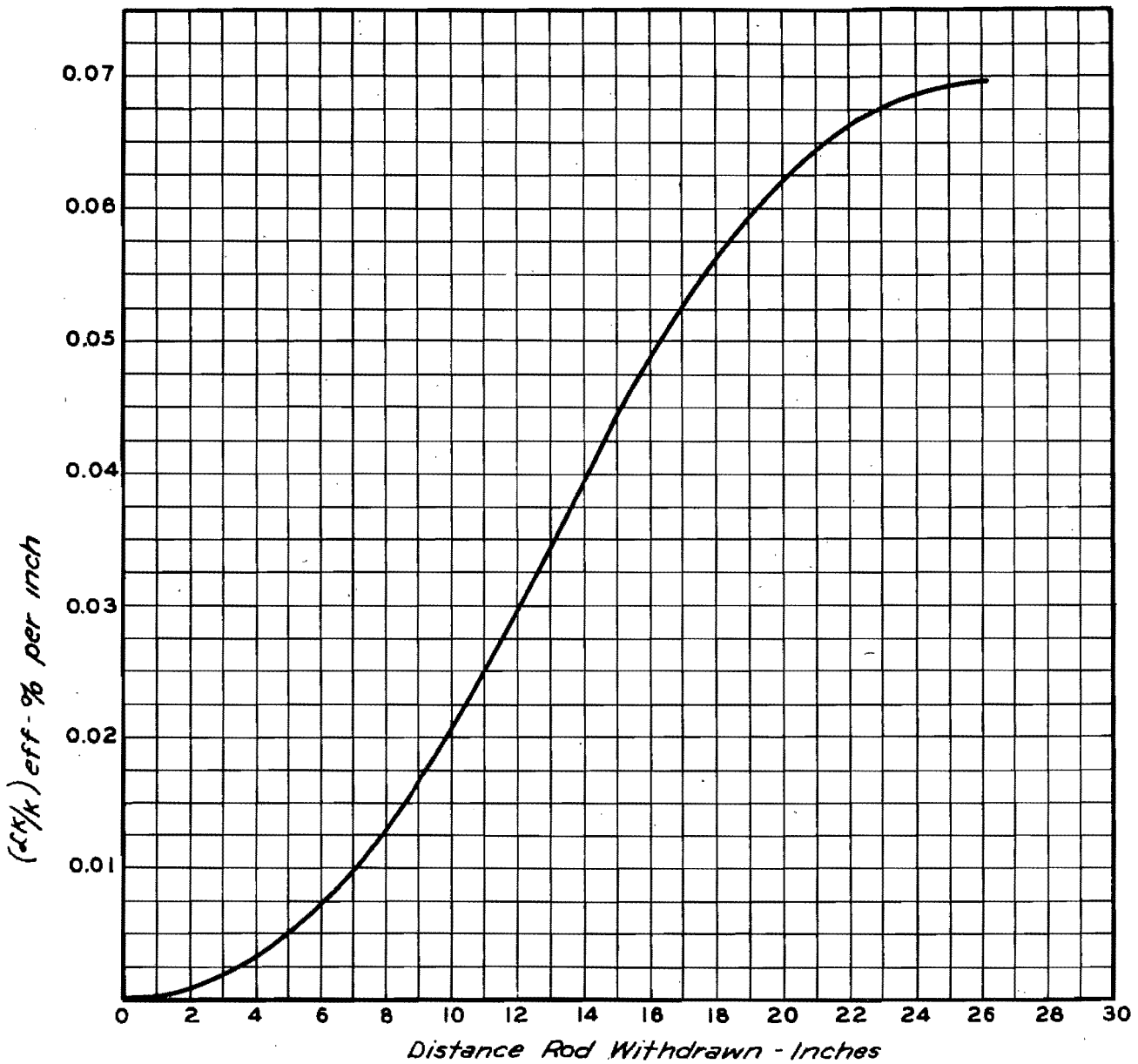



FIG. A.4.Q

SENSITIVITY OF REGULATING ROD

As a result, the total movement of the cadmium section of the rod was reduced by 4 in. with a resultant reduction in equivalent reactivity to about 0.65%. It is expected that the regulating rod in the MTR will contain less than 0.65% in $(\Delta k/k)_{eff}$ because the MTR core will be larger than that in the Mock-Up.

**A4.5.3 Temperature Coefficient.** The measurement of the Mock-Up temperature coefficient was made in the following way: The reactor was made critical at the ambient bulk water temperature (usually 20 to 25°C), and positions of the shim and regulating rods were carefully recorded. After the reactor was shut down, the water pumps were operated until the pump input energy (about 350 kw) had increased the temperature of the circulating water a sufficient amount (5 or 10°C). At the new temperature, the pumps were stopped and the reactor was again brought to some low power level with the shim rods set exactly at their previous locations, corrected for rod expansion at the increased temperature, and the new position of the regulating rod was recorded. From the rod calibration curve the change in reactivity for that particular temperature increment was read. This change in $(\Delta k/k)_{eff}$ was checked later by measuring the reactor period resulting from the same movement of the regulating rod.

Measurements made in this manner are listed in Table A4.F. The average of 0.013 is in fairly good agreement with the calculated value of 0.017% per degree centigrade for the Mock-Up core with a $k$ of about 1.58.

**TABLE A4.F**

**Experimental Values for Temperature Coefficient**

| TEMPERATURE (°C) | PERCENT $(\Delta k/k)_{eff}$ PER °C |
|---|---|
| 36.1 | -0.013 |
| 37 | -0.012 |
| 43 | -0.012 |
| 45 | -0.012 |
| 46.5 | -0.014 |
| 48 | -0.012 |
| 53 | -0.014 |
| 54 | -0.013 |
| | -0.013 (avg.) |

**A4.5.4 Effect of an Empty Experimental Beam Hole.** The effect on reactivity of removing a beryllium plug from the 6-in.-diameter northwest experimental thimble (see Fig. A4.C) extending to the lattice face was determined by comparison with changes in the position of the regulating rod. For the Mock-Up core the effect was a 0.6% reduction in $(\Delta k/k)_{eff}$. Filling the empty hole with a paraffin plug resulted in a $(\Delta k/k)_{eff}$ increase of 0.3%. Although the effect of an empty hole measured here is considerably greater than that observed in the critical experiments, the difference in core size and asymmetry of reflector are probably responsible, and it is expected that the MTR will exhibit effects more similar to those of the critical experiments.

**A4.5.5 Effect of Water Flow.** It has been speculated that high-velocity water flow in the fuel assemblies might result in sufficient movement to be detectable as small fluctuations in reactivity, and an experiment was conducted to investigate this possibility. The upper fuel-positioning grid was removed so that fuel elements would be unconfined at their upper ends. A water velocity of 30 ft/sec was maintained through the cooling clearances. The ion chamber which normally fed the servo control mechanism was used as the detector of reactivity changes. Several measurements were made both with and without water flow, but no difference was detectable. It was therefore concluded that water flow had no significant effect on the stability of the reactor. Furthermore, since many small air bubbles were visible in the circulating water, it was concluded that, as expected, uniformly distributed air bubbles would not cause reactivity fluctuations.

**A4.5.6 Performance of Control and Safety Circuits.** The two instruments in which most interest has been evidenced are the period indicator and the servo mechanism.

The period meter is a direct indicator of rate of increase or decrease of reactor level. Its scale is calibrated in seconds so that the period, i.e., the time required for the power level to change by a factor of 2.7, can be read directly. This instrument proved extremely valuable in start-ups and was also useful for making changes in power level when the reactor was being controlled manually.

The servo mechanism performed exceptionally well. With this device any desired power level in the operating range could be maintained for hours with

only minor attention from the operator. The response of the circuit to large disturbances was measured as approximately 0.1 sec by applying an electrical pulse equivalent to a $(\Delta k/k)_{eff}$ step of about 0.6%. During operation, steps as high as 0.3% $(\Delta k/k)_{eff}$ were added in approximately 1 sec by injecting compressed air at the center of the fuel lattice. The servo mechanism easily followed these disturbances without losing control of the reactor.

Another feature of the MTR control arrangement is the safety circuits which protect against higher-than-normal power levels. Unsafe levels cause a reduction in current to the electromagnets and subsequent release of the control rods. The time required for release of the rods had been measured as approximately 15 msec before installation in the Mock-Up. During operation the time interval between the "scram" signal and shutdown of the reactor was observed to be approximately 30 msec.

Although a number of minor changes in the control instrumentation were deemed desirable before construction of the final MTR instruments, it can be said that their performance in the Mock-Up was very satisfactory.

**A4.5.7 Measurement of Neutron Fluxes.** The data presented below have been analyzed and compared with previous calculations.[13,14] Neutron flux distributions were measured by determining the saturated activity ($A_s$) induced in bare and cadmium-covered metal foils. The measurements can be divided into three series of experiments, based on the experimental conditions and the information sought in each series.

In the first group of experiments the northwest thimble was left open. Fluxes were measured in the fuel lattice, the northwest quadrant of the graphite reflector, and at other miscellaneous points in the structure.

In the second series the northwest thimble hole was closed, as described later, and additional foil exposures were made at many of the locations mentioned above. Neutron attenuations in the water above the lattice up to the elevation of the shim-rod electromagnets were measured also.

The third series included attenuation in concrete blocks and in paraffin stacked behind a portion of the steel thermal shield outside the graphite stack.

Gold foils (1 cm square, 0.005 in. thick) were used for exposures in the graphite. The foils were spaced 6 in. apart on 1/16-in.-thick aluminum strips

for exposures in the fuel lattice and in the 1-in.-diameter holes which extended vertically through the graphite reflector. Measurements of very low fluxes (e.g., in the water above the fuel) were made with large indium foils (25 cm square, 0.005 in. thick) which were held on lucite strips during exposure. Correction for the activity induced by resonance and fast neutrons was made by exposing foils covered with cadmium. The cadmium covers were 0.020 in. thick and were coated with lacquer to prevent contamination of the foils.

Mica-window Geiger tubes were used to count the 1-cm foils, and the glass tubes were used with the large indium foils. The small foils were placed on a flat aluminum plate beneath the mica-window tubes; the large foils were enclosed in a brass cylinder and slipped over the glass tubes. Only the side of the foil which had faced the reactor was counted, and this side was always placed nearest the counting tube.

The foils and counters were calibrated by irradiating foils in positions of known flux in a standard graphite pile.[16]

In almost all instances exposure time and start-up time were comparable; consequently a start-up correction has been applied to the foil data.[16] The power levels for the many foil activations varied from 10 watts to 2 kilowatts.

*First Series.* Experimental conditions for the first series were:

1. Lattice arrangements as shown in Fig. A4.C.
2. No thermal shield.
3. Experimental holes filled as follows:
   - (*a*) Southwest thimble — filled solidly with graphite from edge of the lattice to outer edge of graphite.
   - (*b*) Center west thimble — filled with an 18-in.-long by 6-in.-diameter cylinder of beryllium beginning 1 in. from the edge of the lattice; the differential chamber and one safety chamber occupied the otherwise empty space behind the beryllium plug.
   - (*c*) Northwest thimble — empty except for the $U^{235}$ fission chamber and one safety chamber.
   - (*d*) Southeast and center east thimbles — filled with paraffin.
   - (*e*) Northeast thimble — filled with graphite backed by a concrete plug.

4. Regulating rod removed.

5. Shim rod No. 1 completely withdrawn, shim rod No. 2 withdrawn 80%.

6. Graphite reflector arranged as indicated in Fig. A4.C with 1-in.-diameter holes extending from top to bottom in the numbered positions.

7. Reactor tank filled with demineralized, deaerated water.

All measurements in this series were normalized by comparing gold foils activated in graphite hole 44 during each run.

Figures A4.R and S show the data for the mid-plane and top plane of the active lattice. Typical curves showing vertical distribution of neutron fluxes in several holes in the graphite reflector are presented in Figs. A4.T, U, and V. Table A4.G lists the thermal-neutron flux values at various mid-plane positions in the graphite reflector with the cadmium ratio at the points where it was measured.

**TABLE A4.G**

**Neutron Flux in Mid-plane**

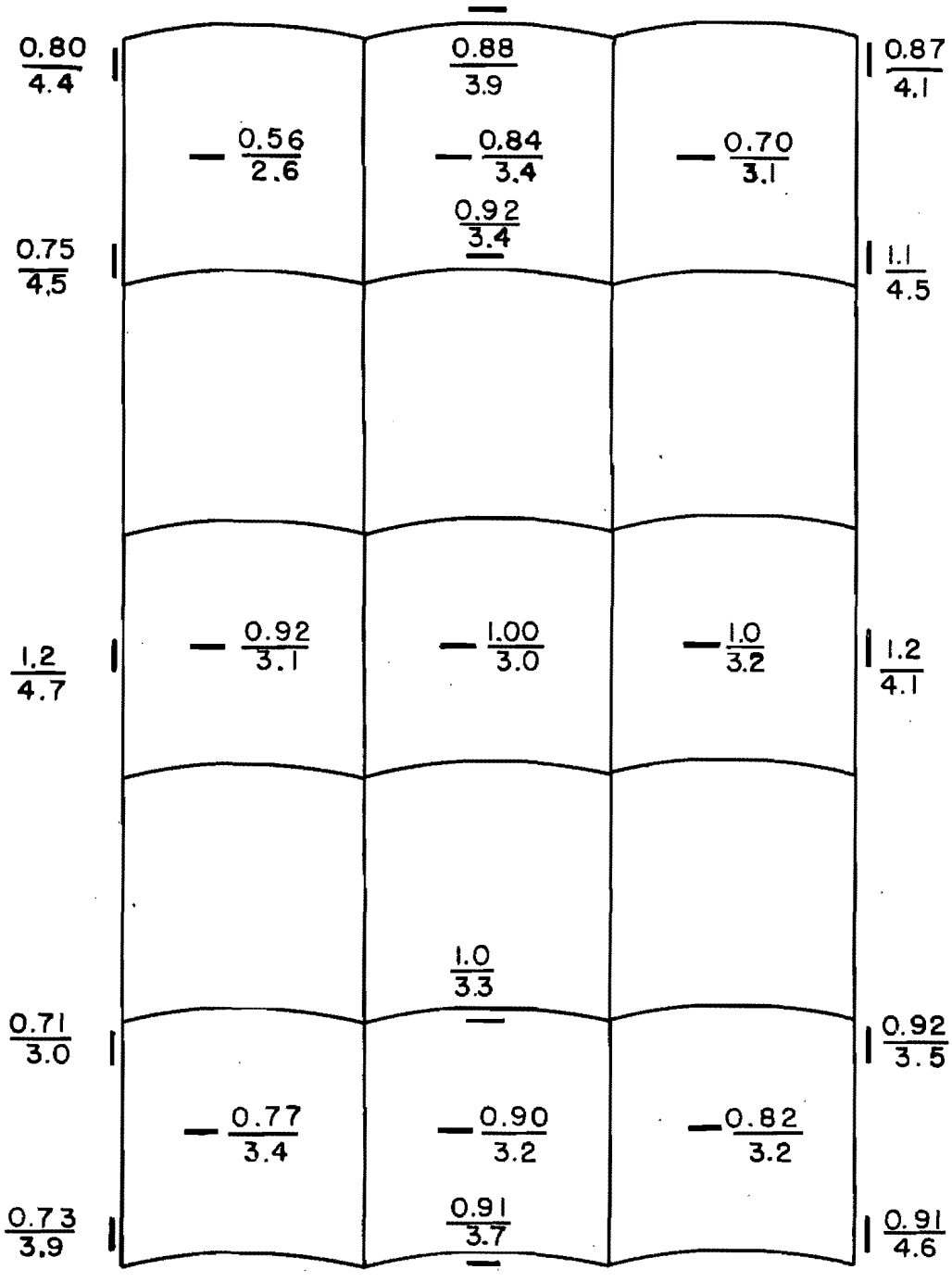
First series: Northwest Thimble Open

| HOLE NO. | nv (thermal)* | CADMIUM RATIO** |
|---|---|---|
| 44 | 123 $\times 10^{-3}$ | |
| 45 | 81 | 19 |
| 46 | 107 | |
| 13 | 27 | |
| 19 | 16 | 100 |
| 15 | 13 | 44 |
| 16 | 5.0 | 140 |
| 34 | 0.3 | 230 |
| 38 | 1.83 | |
| 11*** | 0.92 | |
| 16 | 1.1 | |
| 34 | 0.24 | |
| 37 | 0.65 | |

*Normalized to nv (thermal) of 1.0 neutron/cm$^2$-sec at center of lattice.
**Corrected for absorption of resonance neutrons in cadmium.
***Outer surface of graphite behind hole.

FIG. A4.R

Thermal Neutron Flux in Midplane of Active Lattice

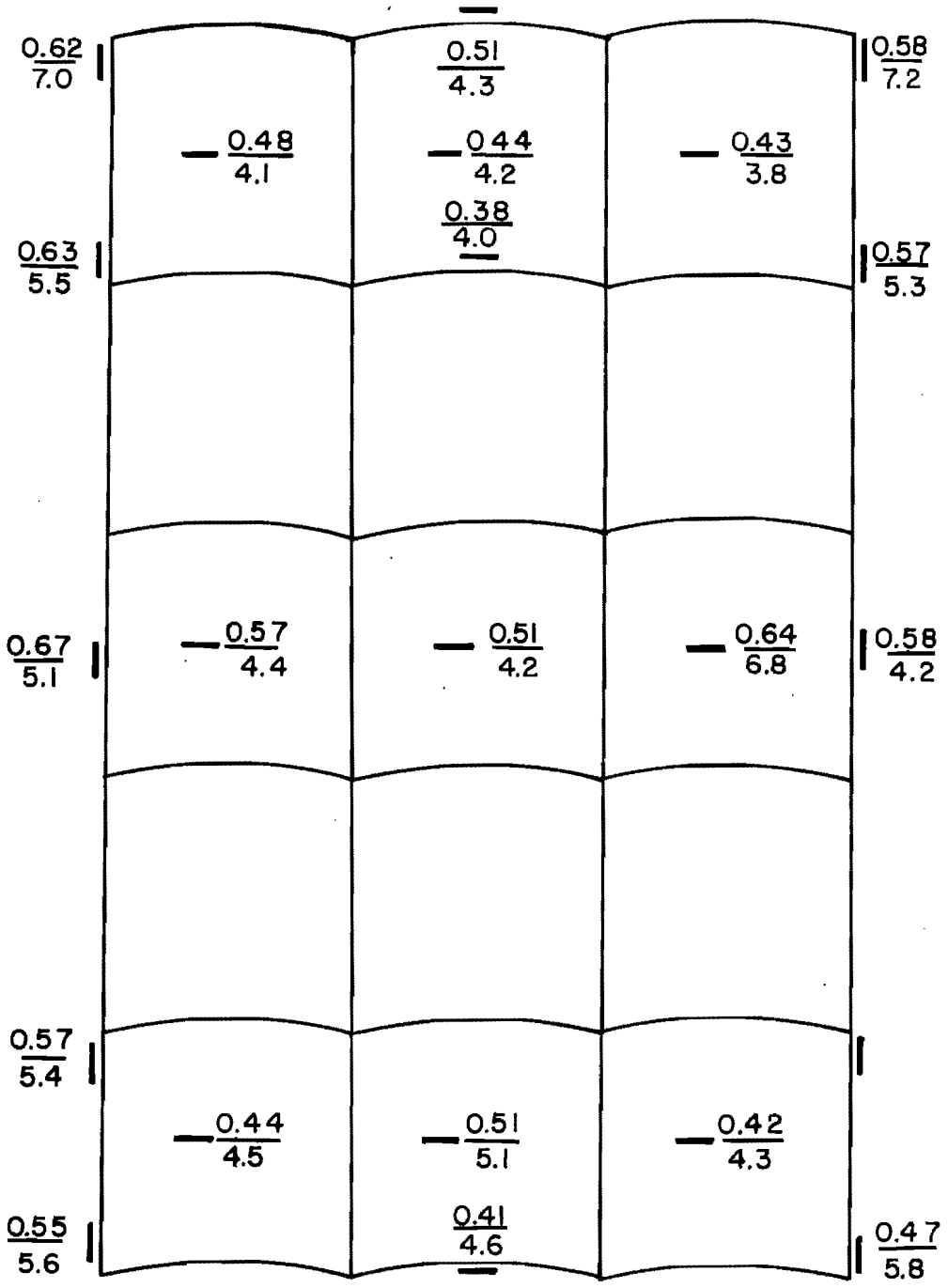



Numerator is $NV_{th}$ Relative to $NV_{th}$ at Center of Lattice

Denominator is Cadmium Ratio

Dwg. 9792
[redacted]

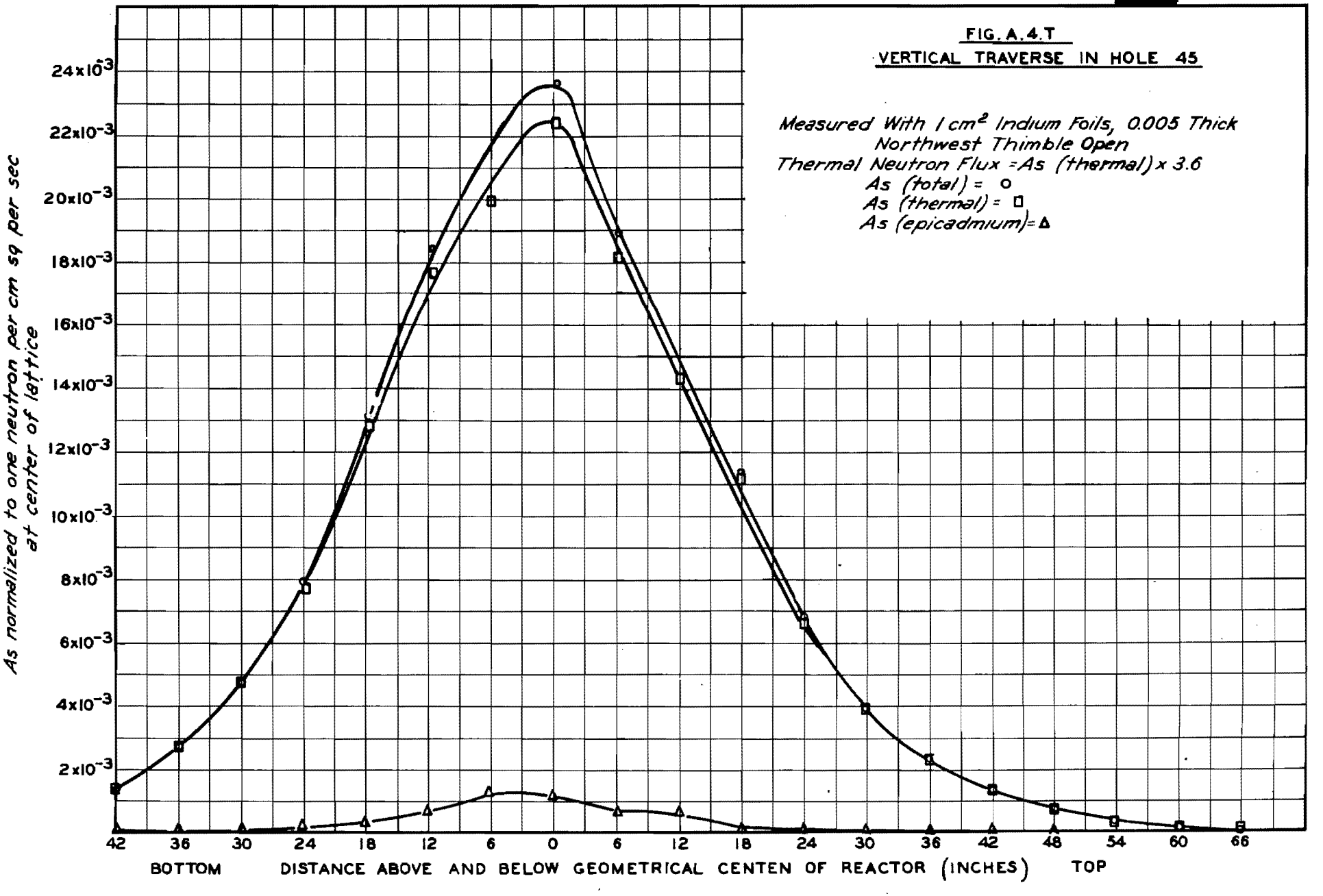
FIG. A4.S
Thermal Neutron Flux in Top Plane of Active Lattice



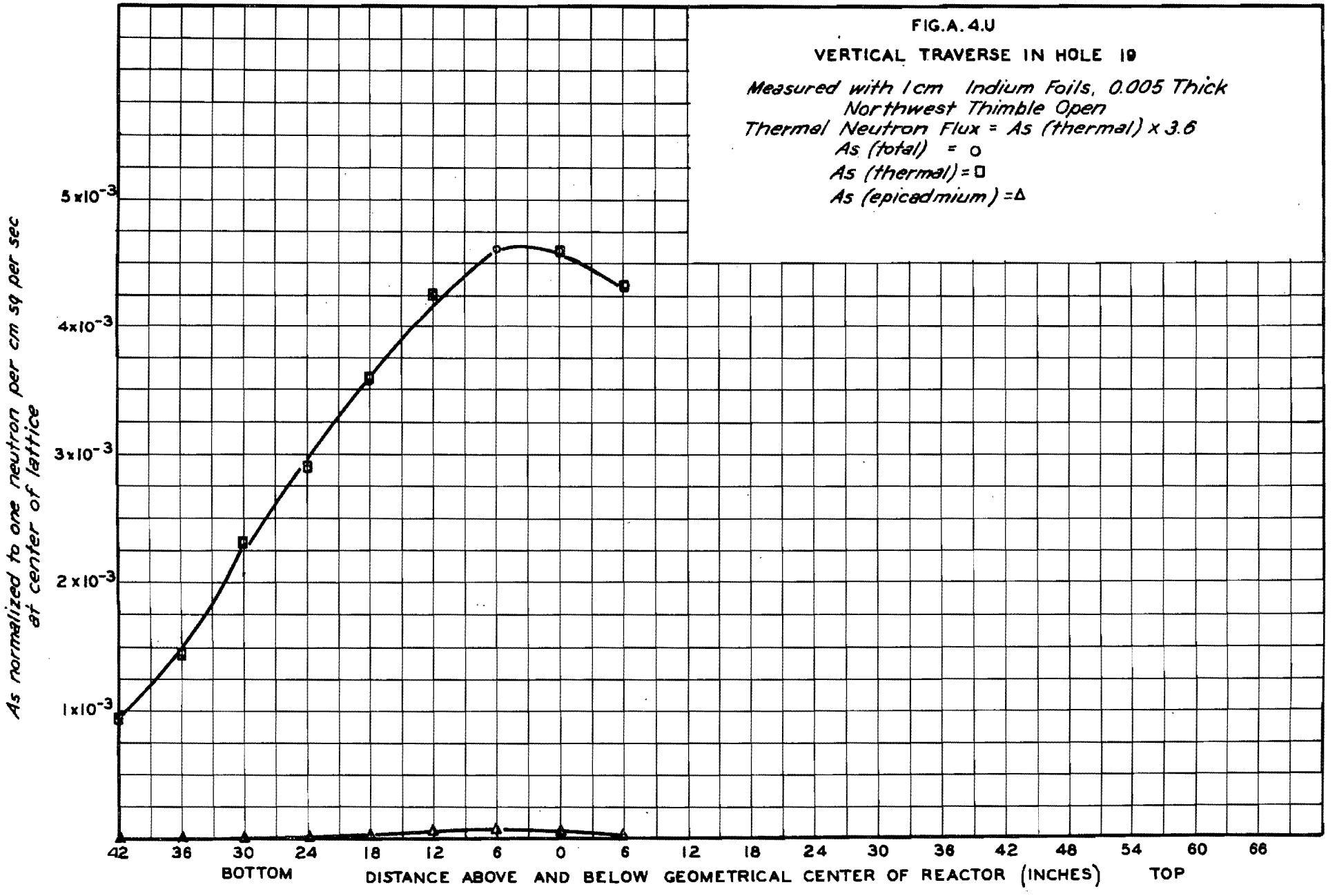

Numerator is $NV_{th}$ Relative to $NV_{th}$ at Center of Lattice

Denominator is Cadmium Ratio

DWG. #10816 [redacted]T



DWG. #10814 [redacted]



DWG. # 10815

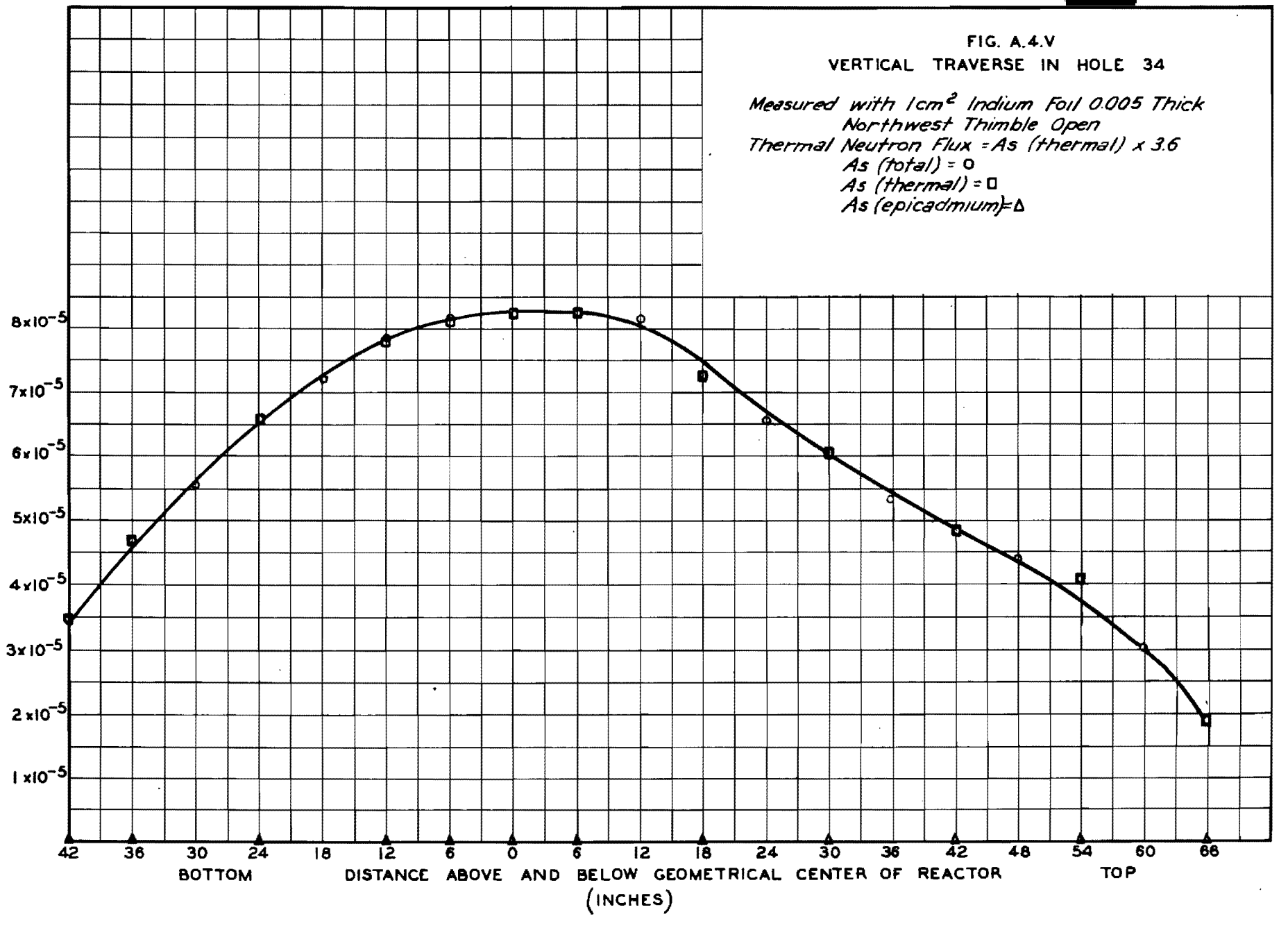


FIG. A.4.V
VERTICAL TRAVERSE IN HOLE 34
Measured with 1cm² Indium Foil 0.005 Thick
Northwest Thimble Open
Thermal Neutron Flux = As (thermal) x 3.6
As (total) = O
As (thermal) = □
As (epicadmium) = Δ
As normalized to one neutron per cm sq per sec at center of lattice
8x10⁻⁵
7x10⁻⁵
6x10⁻⁵
5x10⁻⁵
4x10⁻⁵
3x10⁻⁵
2x10⁻⁵
1x10⁻⁵
42 36 30 24 18 12 6 0 6 12 18 24 30 36 42 48 54 60 66
BOTTOM
DISTANCE ABOVE AND BELOW GEOMETRICAL CENTER OF REACTOR
TOP
(INCHES)

*Second Series.* The significant changes from the experimental conditions in the first series were:

1. The beryllium cylinder in the center west thimble was moved to the northwest thimble; the space behind the beryllium was filled with graphite.
2. The regulating rod was put into operation.
3. Both shim rods withdrawn to the same height (approximately 26 in.).
4. Center west thimble empty except for ion chamber.
5. Southwest thimble empty except for ion chamber.
6. A partial thermal shield 4 ft by 4 ft by 4 in. thick was placed temporarily behind hole 17 at the edge of the graphite for the measurements reported in Table A4.I.

Table A4.H lists the flux at mid-plane in most of the holes in the northwest quadrant of the graphite reflector and the cadmium ratio in the positions where it was measured. Table A4.I shows the measurements taken on the top and bottom of the graphite reflector. Traverses on the west face of the graphite behind hole 17 with and without the partial steel thermal shield section in place are presented in Table A4.J.

Figure A4.W is a plot of the neutron flux along the vertical centerline of the active lattice, starting at the bottom of the lattice and continuing up to the shim rod electromagnets. The flux at the magnets is seen to be too low to give good counting statistics. To check this measurement the neutron attenuation curve from the X-10 reactor lid tank was normalized and plotted with the curve from the Mock-Up, considering the top of the Mock-Up lattice to be the source plate.

*Third Series.* This series includes two sets of measurements taken behind a 4-ft-square section of the steel thermal shield.

For the first measurements barytes concrete blocks were stacked behind the steel thermal shield and covered with a boron-tygon sheet and paraffin. Other experimental conditions were as described for the Second Series except that the center west thimble was filled with paraffin backed with graphite. Fluxes were measured in the concrete blocks using small indium foils ( 1 cm square) which gave activities too low for accurate counting.

DWG. # 10803
NOT CLASSIFIED

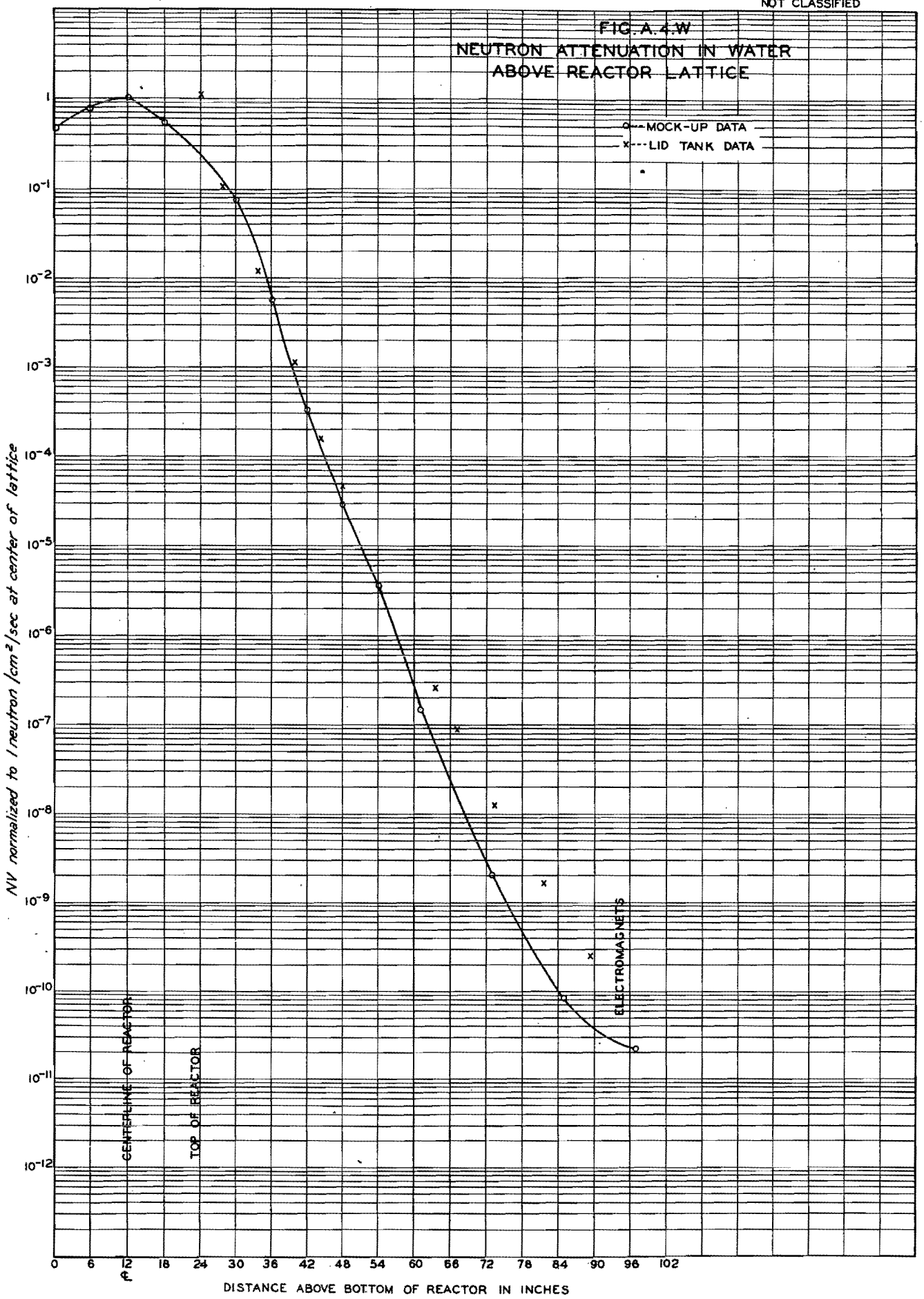


**TABLE A4.H**

**Neutron Flux in Mid-plane**

Second Series: Northwest Thimble Closed; Center Thimble Open

| HOLE NO. | nv (thermal)* | CADMIUM RATIO** |
|---|---|---|
| 44 | 126 X $10^{-3}$ | 40 |
| 45 | 61 | 80 |
| 46 | 82 | 60 |
| 13 | 31 | |
| 19 | 12 | 600 |
| 20 | 28 | 160 |
| 21 | 54 | |
| 12 | 12 | 900 |
| 15 | 8.5 | |
| 18 | 4.9 | 1600 |
| 26 | 2.0 | 2100 |
| 27 | 4.4 | 2200 |
| 28 | 8.5 | 1300 |
| 29 | 12 | |
| 11 | 3.1 | |
| 16 | 2.0 | 800 |
| 17 | 1.2 | 1800 |
| 25 | 0.54 | 1600 |
| 34 | 0.18 | 1050 |
| 35 | 0.49 | |
| 36 | 0.87 | |
| 37 | 1.6 | |
| 38 | 2.0 | 1300 |
| 11*** | 1.1 | |
| 16*** | 0.55 | |
| 17*** | 0.36 | |
| 25*** | 0.22 | |
| 34*** | 0.19 | |
| 35*** | 0.29 | |
| 36*** | 0.46 | |
| 37*** | 0.65 | |
| 38*** | 0.85 | |

*Normalized to nv (thermal) of 0.1 neutron/$cm^2$-sec at center of lattice.

**Corrected for absorption of resonance neutrons in cadmium.

***Outer surface of graphite behind hole.

## TABLE A4.I

### Horizontal Traverse

| DISTANCE FROM TANK WALL (in.) | nv (thermal)* |
|---|---|
| **On Bottom Plane of Graphite on North-South Centerline of Reactor** | |
| 6 | $1.20 \times 10^{-3}$ |
| 18 | 0.56 |
| 24 | 0.45 |
| 30 | 0.32 |
| **On Bottom Plane of Graphite from Tank Wall Toward Hole No. 17** | |
| 6 | $0.47 \times 10^{-3}$ |
| 18 | 0.69 |
| 24 | 0.73 |
| 30 | 0.44 |
| 47 | 0.10 |
| 59 | 0.050 |
| 62 | 0.045 |
| **On Top Plane of Graphite from Tank Wall Toward Hole No. 17** | |
| 6 | $0.14 \times 10^{-3}$ |
| 18 | 0.22 |
| 24 | 0.21 |
| 30 | 0.24 |
| 47 | 0.071 |
| 59 | 0.049 |
| 62 | 0.047 |

*Normalized to 1 neutron/$cm^2$-sec at the center of the lattice.

TABLE A4.J

Traverses Behind Hole No. 17 on Surface of Graphite

| POSITION | DISTANCE BETWEEN ADJACENT FOILS (in.) | nv (thermal)* NEGLECTING EPI-CADMIUM FLUX | |
|---|---|---|---|
| | | WITHOUT STEEL | WITH 4-in. STEEL |
| Vertical | | | |
| Top | 10 | $0.34 \times 10^{-3}$ | $0.35 \times 10^{-3}$ |
| | 12 | 0.32 | 0.38 |
| Centerline | 12 | 0.32 | 0.42 |
| | 10 | 0.30 | 0.33 |
| Bottom | | 0.27 | 0.23 |
| Horizontal | | | |
| North | 10 | 0.19 | 0.14 |
| | 12 | 0.24 | 0.24 |
| Centerline | 12 | 0.32 | 0.41 |
| | 10 | 0.43 | 0.53 |
| South | | 0.52 | 0.73 |

*Normalized to 1 neutron/$cm^2$-sec at the center of the lattice.

After this unsuccessful attempt to measure fluxes in the concrete, the barytes blocks were removed and the space behind the steel was filled with a block of paraffin, 4 ft by 4 ft by 20 in. thick (Fig. A4.X), containing horizontal and vertical slots for the exposure of large indium foils (25 sq cm). The paraffin cube was surrounded by sheets of boron-tygon plastic and had additional paraffin (left after the concrete experiment) on the north face and underneath.

Figure A4.Y is a centerline traverse perpendicular to the steel thermal shield through 18 in. of paraffin. It shows that leakage from the concrete biological shield behind the paraffin distorts the attenuation curve as far as the middle of the paraffin.

In Figs. A4.Z and AA horizontal and vertical traverses taken behind various thicknesses of paraffin show the centerline traverse to be relatively unaffected by leakage from the top and sides of the block, making Fig. A4.Y useful through the first half of the curve. Many of the cadmium-covered foils exposed in the paraffin showed essentially no activity above background and are not plotted.

**A4.5.8 Heat Production in Graphite, Thermal Shield, and Concrete.** One of the most important MTR design problems has been the determination of the cooling requirements of the graphite, thermal shield, and concrete portions of the reactor structure. It was felt therefore that measurements of the heat generation in these media in the Mock-Up were of particular importance because once the reactor has been constructed major changes in the structure will be impossible.

Measurements of this kind have been made for the beryllium reflector in the critical experiments.[18] For this purpose, small thimble-type ion chambers were used; this method also was employed in the present experiments.

The chambers used were of three sizes, 10, 50, and 95 $cm^3$. Figure A4.BB shows a typical design. A six-channel vacuum-tube electrometer circuit[19] was used with the chambers, making it possible to minimize the time required for the necessarily large number of measurements.

The chambers were calibrated by two independent methods: (1) against radium and cobalt sources (in air), and (2) against a sensitive liquid-nitrogen calorimeter by comparison of heat generated in a graphite block inserted in the ORNL graphite reactor with that calculated from ion chamber measurements in a similar block.[20]

DWG. #10820
NOT CLASSIFIED

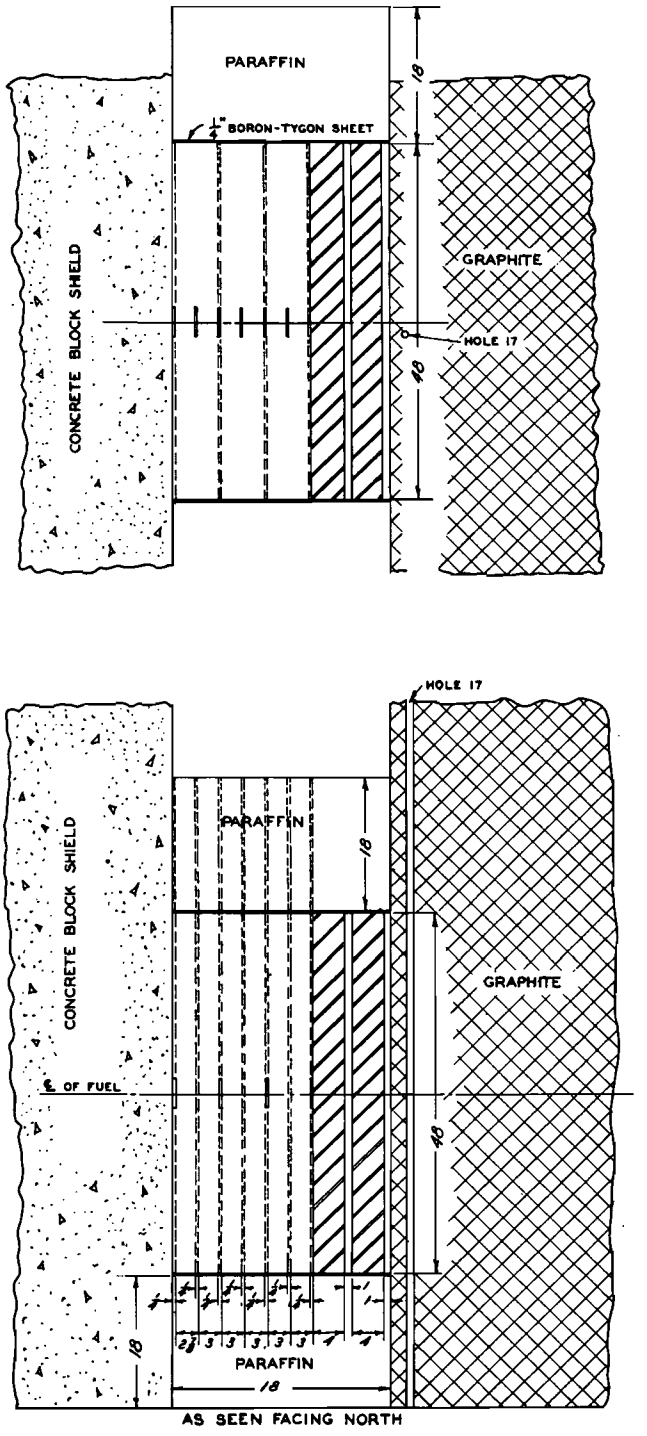



FIG. A.4.X
ARRANGEMENT OF PARAFFIN AND STEEL THERMAL SHIELD SHOWING FOIL EXPOSURE SLOTS

DWG [redacted]8

FIG. A4.Y

NEUTRON ATTENUATION IN PARAFFIN BEHIND THE THERMAL SHIELD

Points are As vs Thickness of Paraffin behind 8 in. of steel

Foils are Indium, 25 cm sq, 0.005 in. thick

Reactor Power ~ 2 KW

NV(th) = As(th) X 0.13

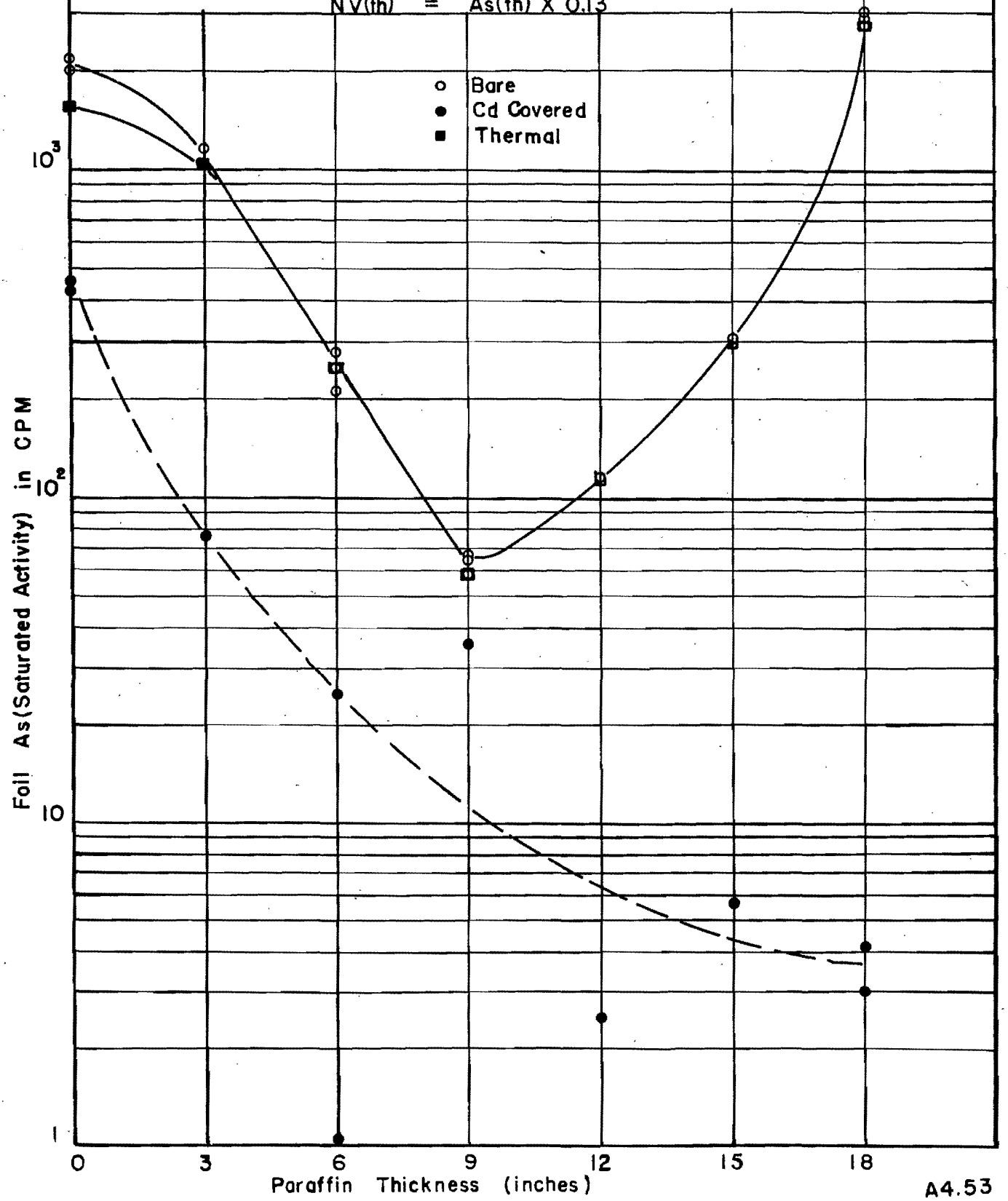


FIG. A4.Z

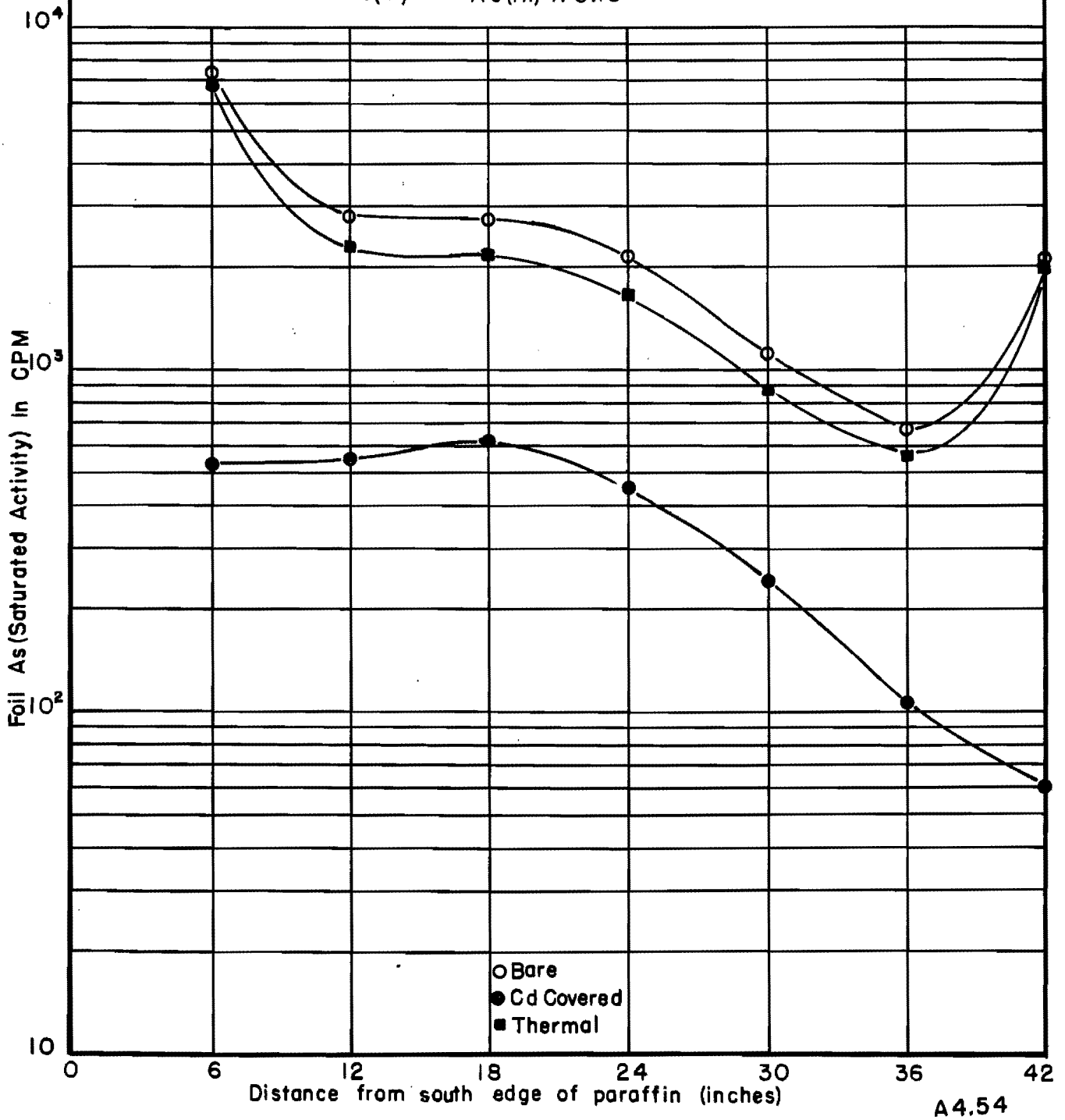
HORIZONTAL TRAVERSE BETWEEN PARAFFIN AND STEEL

Points are saturated activity vs distance from the south end of a block of paraffin placed behind eight inches of steel.

Foils are Indium, 25 cm sq, 0.005 in. thick,

Reactor Power ~ 2 KW

NV(th) = As(th) X 0.13



Dwg. 9790

FIG. A4.AA

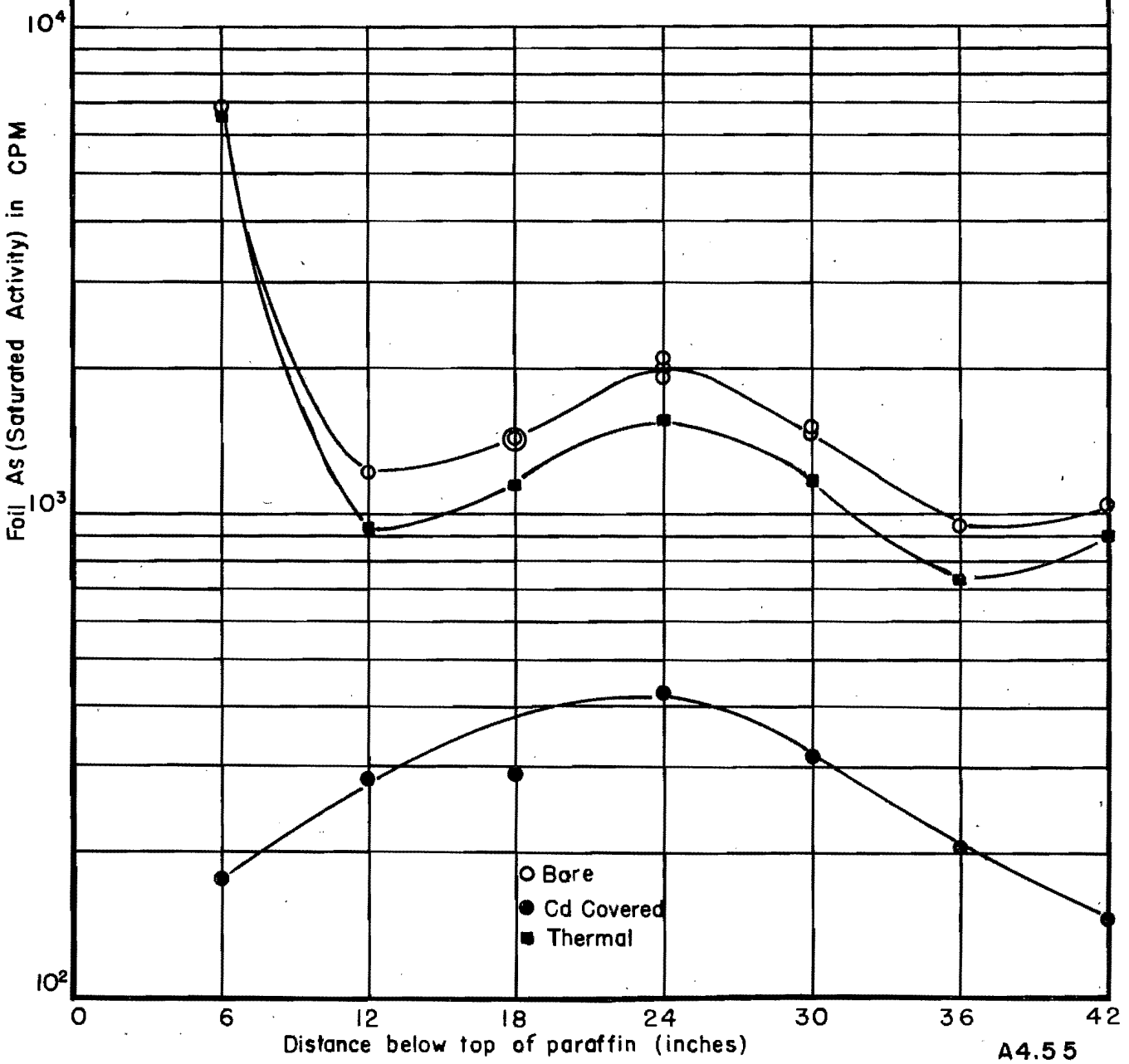
VERTICAL TRAVERSE BETWEEN PARAFFIN AND STEEL

Points are saturated activity vs distance below the top of a block of paraffin placed behind eight inches of steel.

Foils are Indium, 25 cm sq, 0.005 in. thick.

Reactor Power ~ 2 KW

NV(th) = As(th) X 0.13



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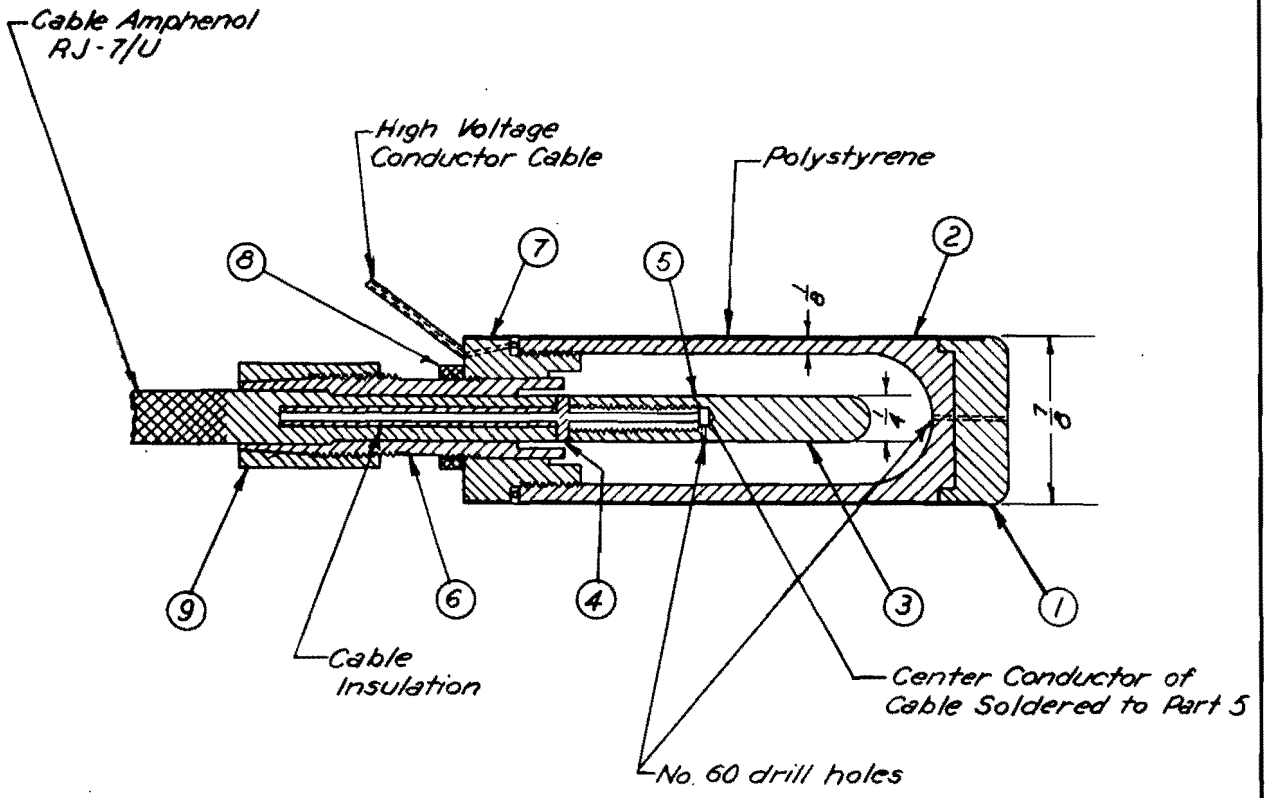



1. End Cover
2. Shell
3. Electrode
4. Connector
5. Sleeve
6. Cable Clamp
7. Chamber Cap
8. Nut
9. Nut

FIG. A.4.BB

$10cm^3$ GRAPHITE IONIZATION CHAMBER USED IN MTR MOCK-UP HEAT PRODUCTION MEASUREMENTS

Calibrations were made with air, $CO_2$, argon, and helium, but $CO_2$ was used for the measurements, making it unnecessary to correct for ionization by protons from the nitrogen $(n,p)$ reaction. Constant flow of gas through the chambers was supplied through the coaxial cables and connectors.

The method of measurement consisted simply of insertion of ion chambers in the various holes provided in the graphite for this purpose (see Fig. A4.C) and observation of the ion current with the reactor operating at a constant power level. For the purpose of normalization, ion chambers in two holes (8 and 13) were checked carefully against neutron (foil) measurements in the core, and thereafter all measurements were referred to chambers in these holes.

In computation of the heat production, the ion chamber measurements have been referred to a neutron flux of 1 neutron/$cm^2$-sec at the center of the core.

Two major groups of measurements were made at various points in the graphite reflector. In the first experiments the arrangement of the experimental holes and tank thimbles was that of the first series of neutron-flux measurements. The second series of ion chamber measurements corresponds to the second series of neutron experiments. Tables A4.K and L and Figs. A4.CC, DD, EE, and FF contain the results. These data have been analyzed in ORNL CF-50-7-86.

Measurements were also made in sections of barytes concrete which were placed behind the simulated thermal shield. For this purpose a special chamber with a polystyrene wall of 0.010 in. thickness was constructed. This modification produced, in effect, an air-wall chamber, and it is believed therefore that the measurements in concrete are probably correct within about 25%. Typical results are shown in Fig. A4.GG.

**A4.5.9 Attenuation of $\gamma$-Rays Inside Reactor Tank.** The top plug of the MTR tank is designed so that it can be filled with lead to a maximum thickness of 10 in., although the exact thickness required for shielding at 60,000 kw was thought to be less than 10 in. An estimation of this thickness has been obtained from measurements of $\gamma$-ray attenuation in the Mock-Up tank with the active lattice operating at approximately 270 watts.

Attenuation measurements were made with sensitive watertight ion chambers along two vertical lines above the active lattice. Data are plotted in Fig. A4.HH and indicate a factor of $e$ reduction in 29.5 cm. The discontinuity

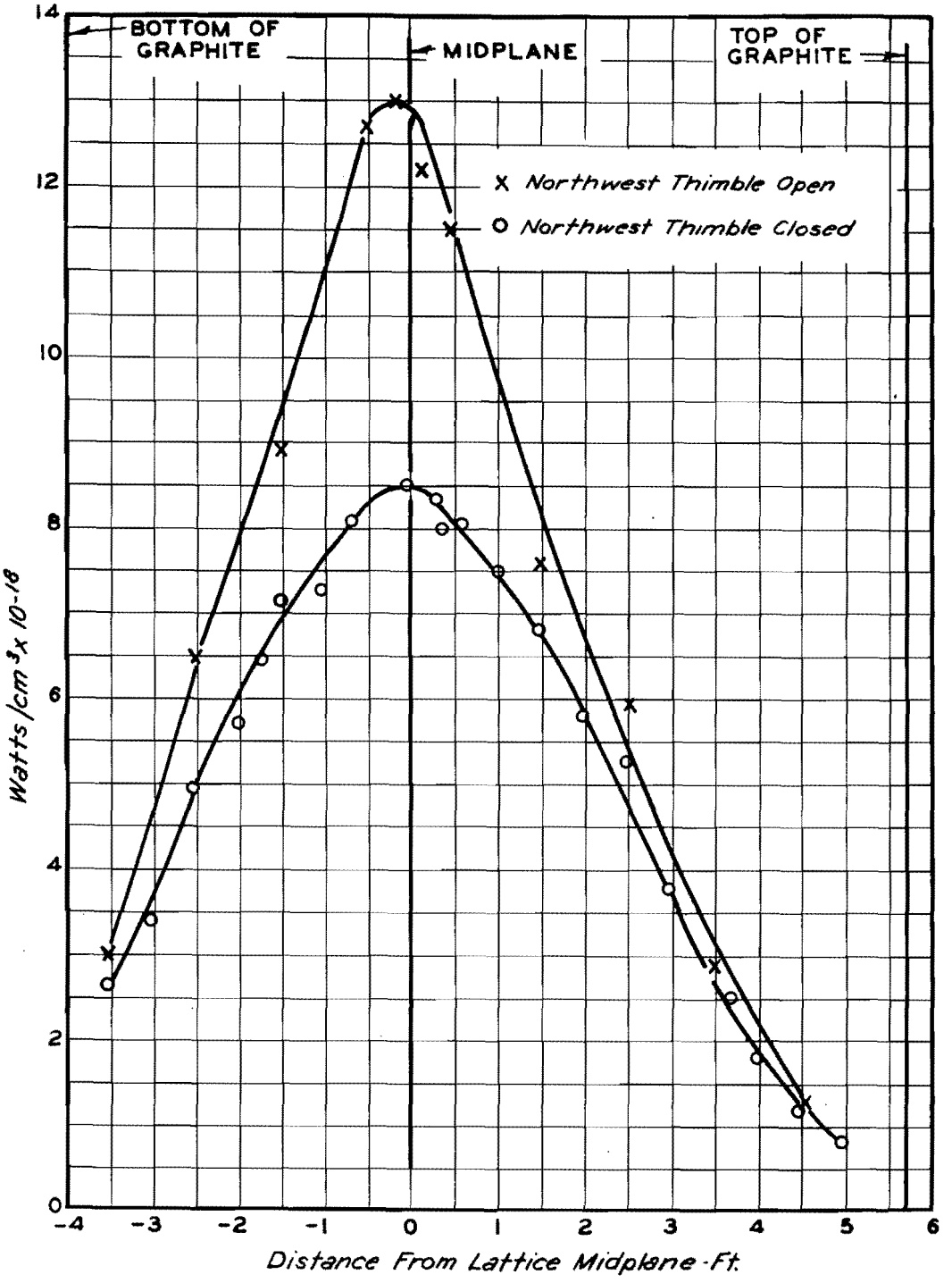



FIG. A.4.CC

HEAT PRODUCTION IN HOLE 19

*Values Normalized To $NV_{th}$ = 1.0n/cm² sec at Core Center*

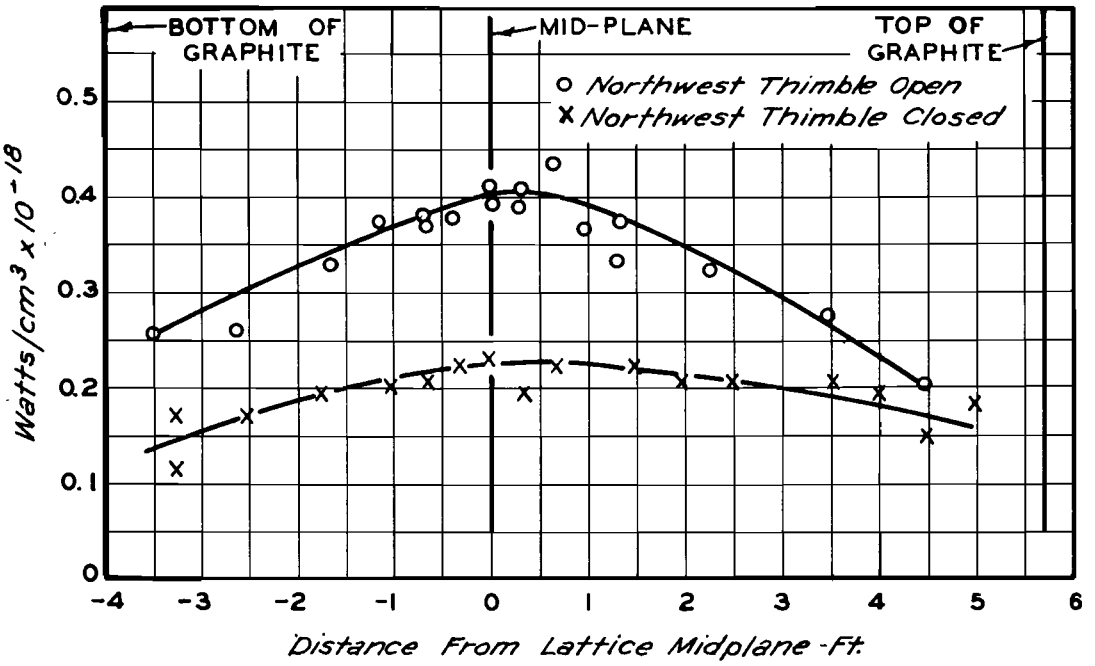



FIG. A.4.DD
HEAT PRODUCTION IN HOLE 34

*Values Normalized To $NV_{th} = 1.0 n/cm^2$ sec At Core Center*

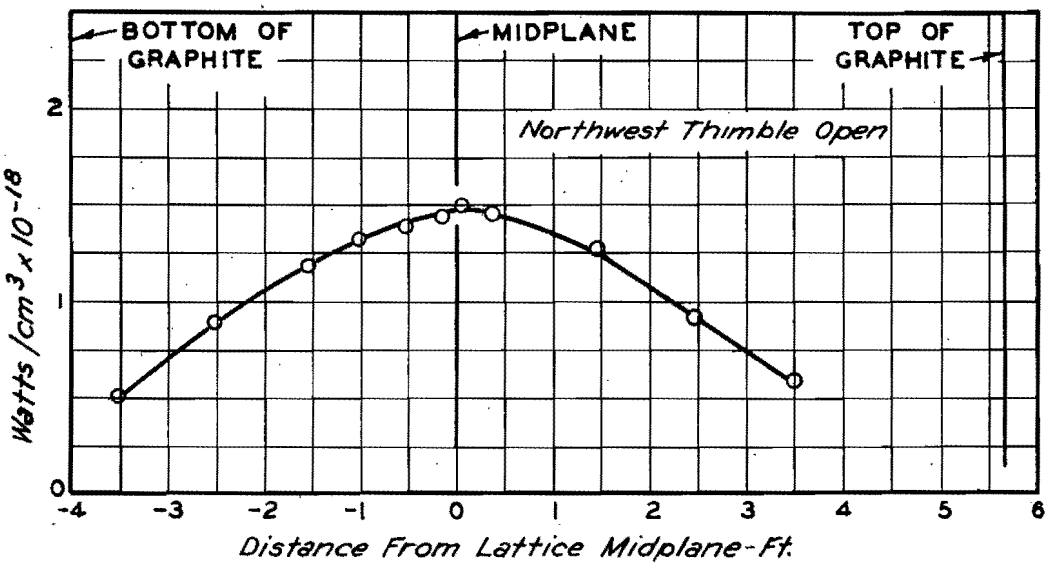



FIG. A.4.EE

HEAT PRODUCTION IN HOLE 26

*Values Normalized To $NV_{th}$ = 1.0 n cm²/sec At Core Center*

DWG. # 10767
~~SECRET~~

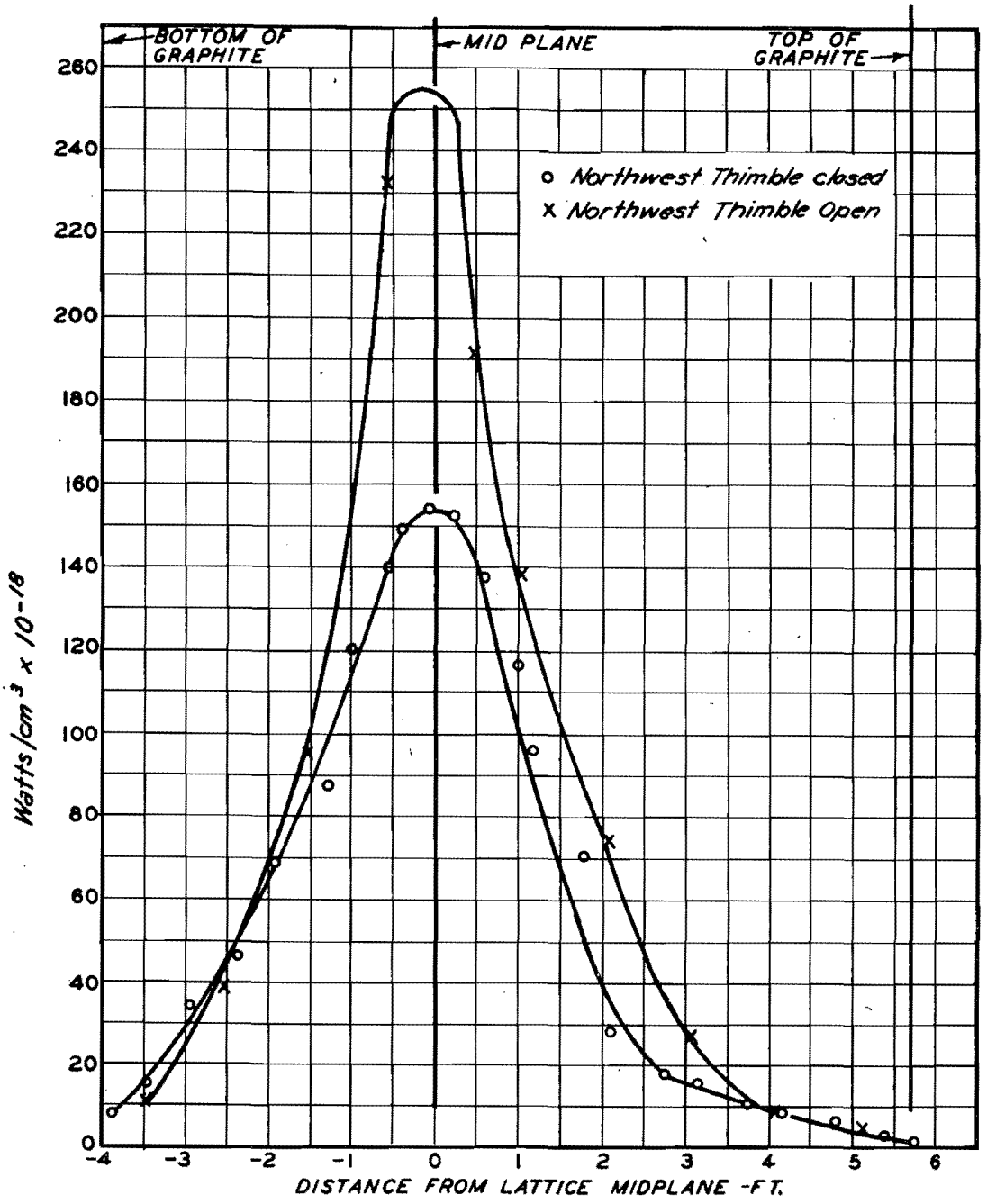


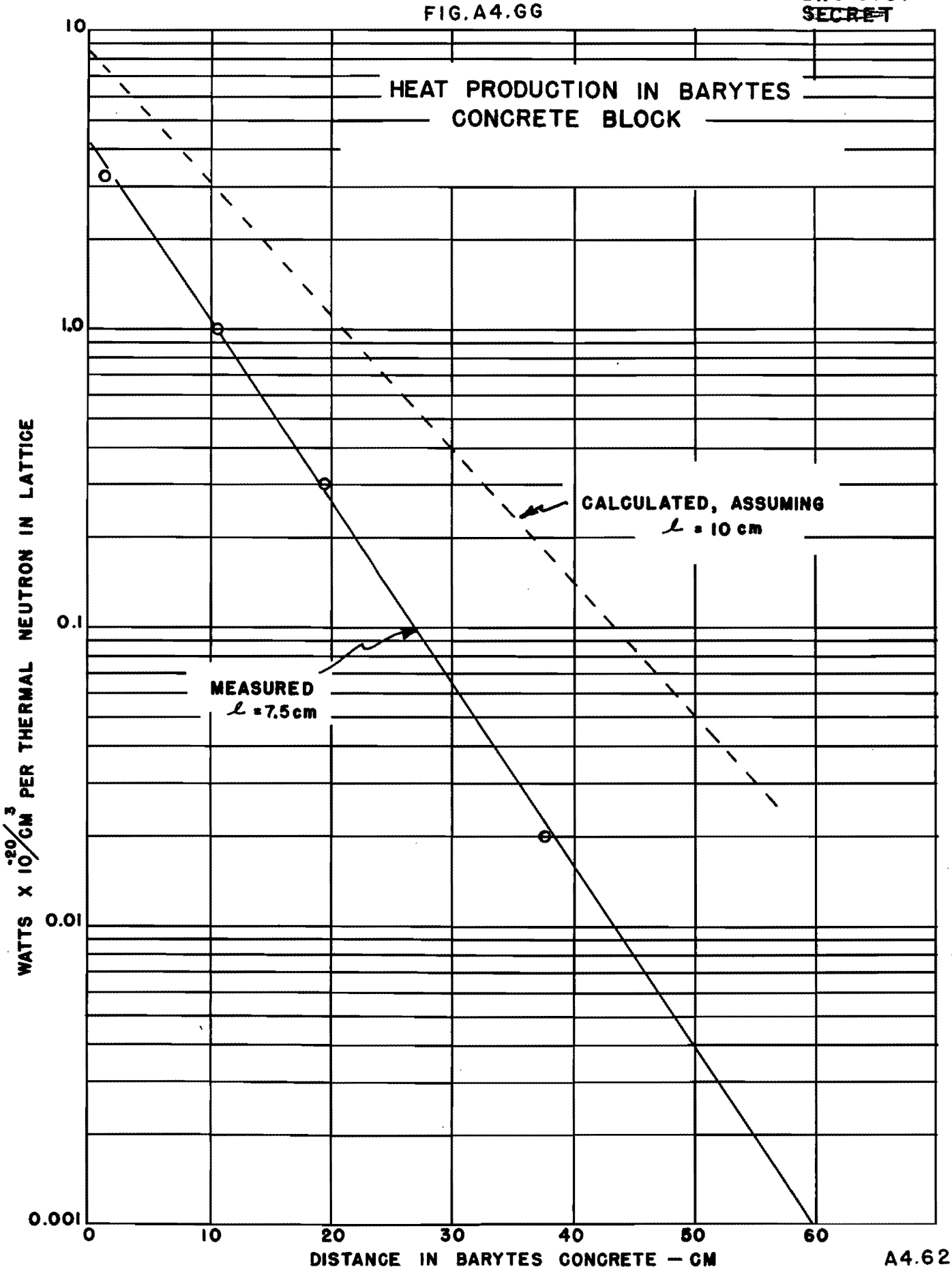

FIG. A.4.FF

HEAT PRODUCTION IN HOLE 45

*Values Normalized to $NV_{th}$ = 1.0 /cm² sec At Core Center*

DWG 9787
~~SECRET~~

FIG. A4.GG




A4.62

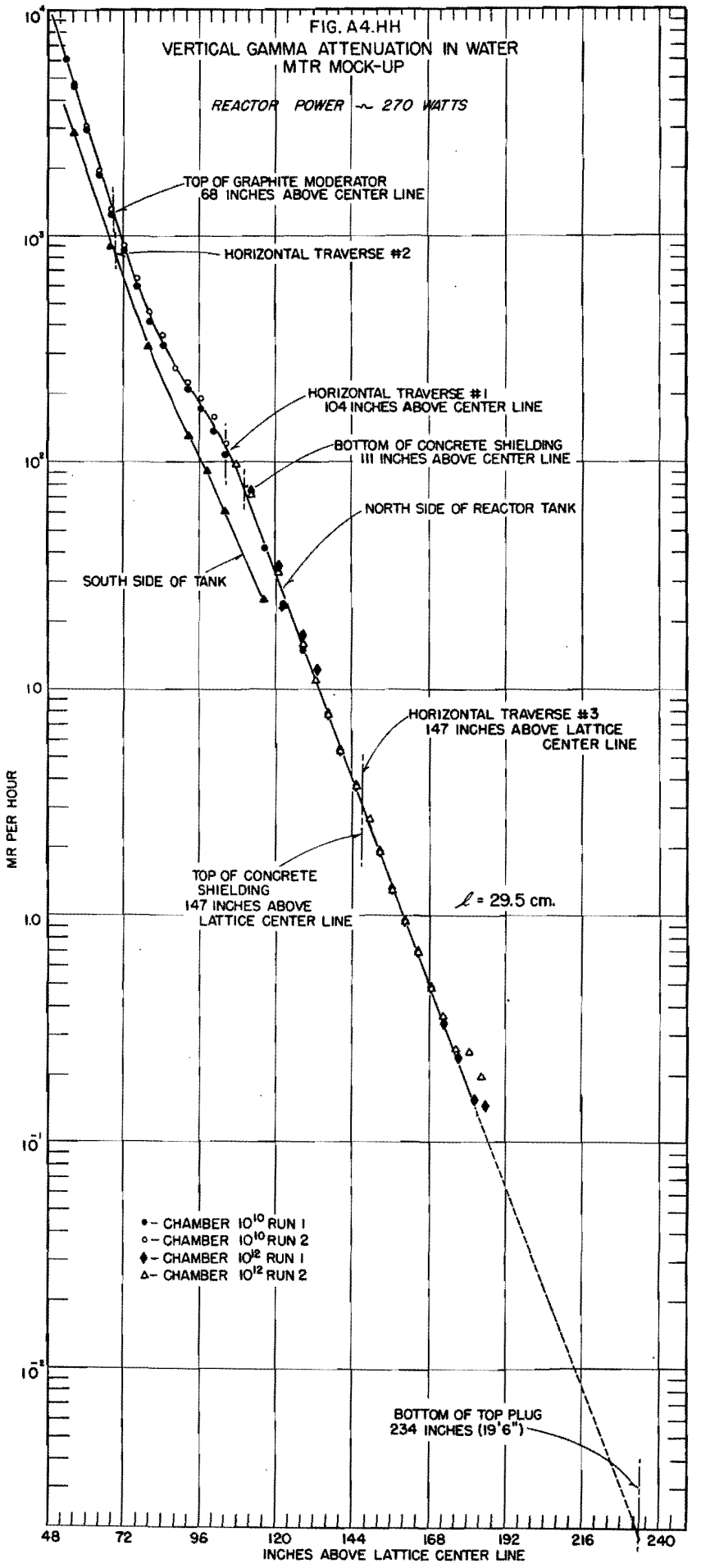


FIG. A4.HH
VERTICAL GAMMA ATTENUATION IN WATER
MTR MOCK-UP
REACTOR POWER ~ 270 WATTS
TOP OF GRAPHITE MODERATOR
68 INCHES ABOVE CENTER LINE
HORIZONTAL TRAVERSE #2
HORIZONTAL TRAVERSE #1
104 INCHES ABOVE CENTER LINE
BOTTOM OF CONCRETE SHIELDING
111 INCHES ABOVE CENTER LINE
NORTH SIDE OF REACTOR TANK
SOUTH SIDE OF TANK
HORIZONTAL TRAVERSE #3
147 INCHES ABOVE LATTICE
CENTER LINE
TOP OF CONCRETE
SHIELDING
147 INCHES ABOVE
LATTICE CENTER LINE
ℓ = 29.5 cm.
MR PER HOUR
10⁴
10³
10²
10
1.0
10⁻¹
10⁻²
• - CHAMBER 10¹⁰ RUN 1
○ - CHAMBER 10¹⁰ RUN 2
♦ - CHAMBER 10¹² RUN 1
△ - CHAMBER 10¹² RUN 2
BOTTOM OF TOP PLUG
234 INCHES (19'6")
48
72
96
120
144
168
192
216
240
INCHES ABOVE LATTICE CENTER LINE

## TABLE A4.K

### Heat Production in Mid-plane, Graphite Reflector

| HOLE NO. | HEAT PRODUCTION (watts/cm$^3$/unit neutron flux in core) | |
|---|---|---|
| | NORTHWEST THIMBLE CLOSED, CENTER THIMBLE OPEN | NORTHWEST THIMBLE OPEN, CENTER THIMBLE CLOSED |
| 11 | $2.4 \times 10^{-18}$ | $2.8 \times 10^{-18}$ |
| 12 | 8.2 | 7.5 |
| 13 | | 36.3 |
| 15 | 6.1* | |
| 16 | 2.3* | 3.8 |
| 17 | 1.9* (0.7) | 4.2 |
| 18 | 2.95* (3.35) | 6.0 |
| 19 | (8.9)* (8.5) | 11.6 |
| 20 | 32 | 29 |
| 21 | 74.5 | 70.5 |
| 22 | 68.0 | 71 |
| 25 | 0.85* (0.4) | 0.75 |
| 26 | 1.07 | 1.3 |
| 27 | 2.3 | |
| 28 | 5.0 | |
| 29 | 7.55 | |
| 30 | | 53 |
| 34 | 0.23 | 0.36 |
| 35 | 0.31 | |
| 36 | 0.52 | |
| 37 | 0.96 | |
| 38 | 1.1 | |
| 44 | 445 | |
| 45 | 155 | 185 |
| 46 | 380 | |

*Steel section, simulating thermal shield, behind graphite; parentheses indicate no steel behind graphite.

## TABLE A4.L

### Measured Heat Production in Top and Bottom Planes of Graphite Reflector

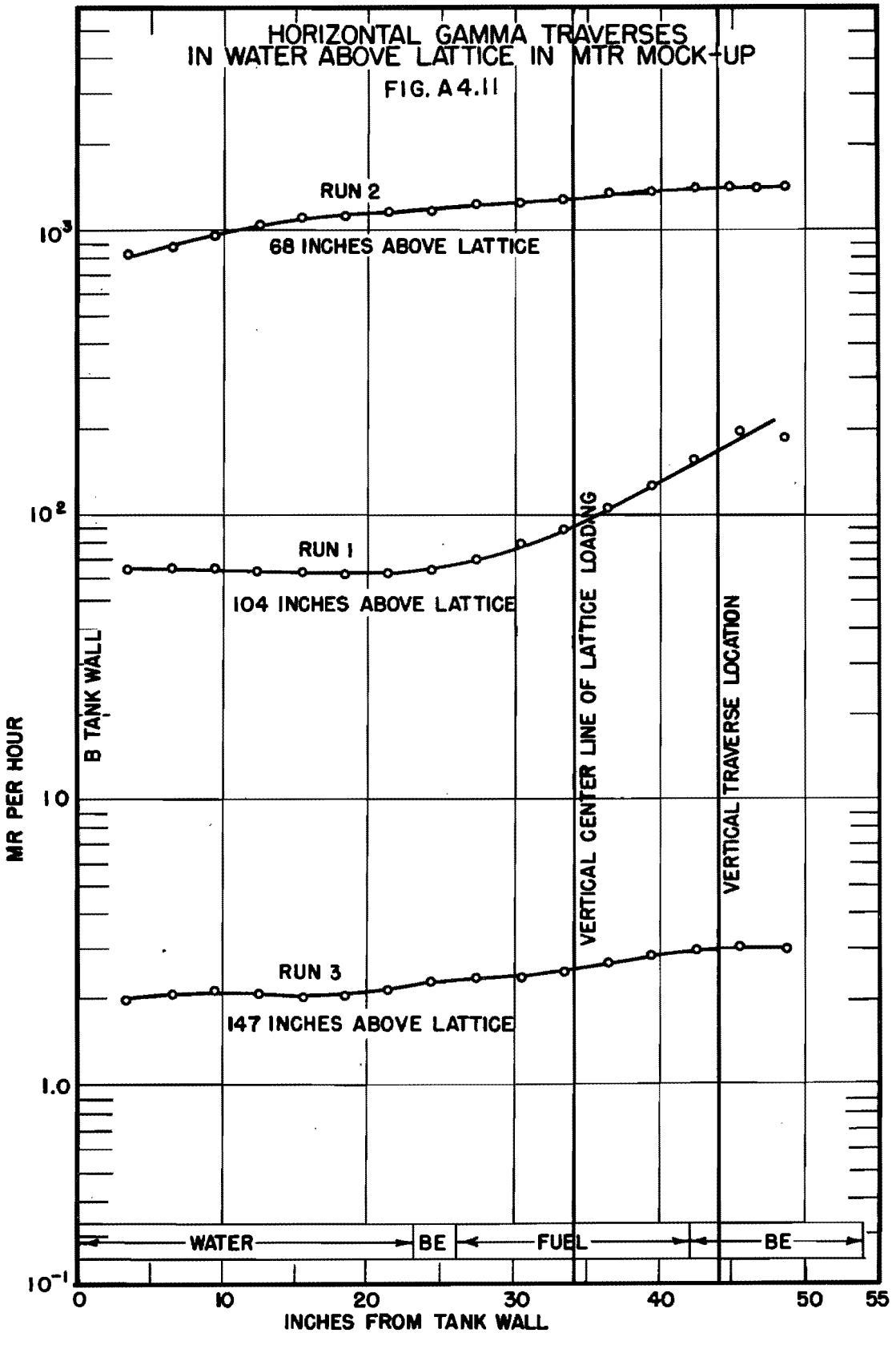
| HOLE NO. | HEAT PRODUCTION (watts/cm$^3$/unit neutron flux in core) | |
|---|---|---|
| | TOP PLANE 6 in. BELOW TOP GRAPHITE | BOTTOM PLANE 6 in. ABOVE BOTTOM GRAPHITE |
| 11 | $0.5 \times 10^{-18}$ | $0.6 \times 10^{-18}$ |
| 12 | 1.0 | 1.5 |
| 13 | | |
| 15 | 0.7 | 1.5 |
| 16 | 0.5 | 1.0 |
| 17 | 0.4 | 0.4 |
| 18 | 1.2 | 1.0 |
| 19 | 0.8 | 3.0 |
| 20 | 6.0 | 7.0 |
| 21 | 1.4 | 8.65 |
| 22 | | |
| 25 | 0.2 | 0.2 |
| 26 | 0.2 | 0.5 |
| 27 | 0.3 | 0.8 |
| 28 | 0.5 | 1.14 |
| 29 | 0.4 | 1.4 |
| 30 | | |
| 34 | 0.2 | 0.3 |
| 35 | 0.3 | 0.3 |
| 36 | 0.3 | 0.3 |
| 37 | 0.2 | 0.3 |
| 38 | 0.2 | 0.3 |
| 44 | 1.9 | 15 |
| 45 | 2.0 | 10.5 |
| 46 | 1.6 | 9.5 |

of the curves, beginning at approximately 70 in. above the fuel, was studied further with three horizontal traverses, plotted in Fig. A4.II. Traverse No. 1 clearly indicated an additional $\gamma$-ray source on the north side of the tank. This source is explained by the construction of the graphite reflector and the concrete shield as shown in Fig. A4.JJ. The graphite and the air space above permit a high thermal flux to exist up to the 111-in. elevation. The capture of these neutrons by the steel tank sections provides the additional $\gamma$-ray source which disrupts the attenuation curves of Fig. A4.HH.

With these measurements it was possible to extrapolate the attenuation lines to the top plug (elevation 234 in.). Normalization of this point to 60,000 kw indicated approximately 500 mr/hr, requiring a 6-in. thickness of lead shot in the top plug for adequate protection.

Complete details of this work are reported in ORNL CF-50-8-85.[21]

[redacted] DWG. 9402



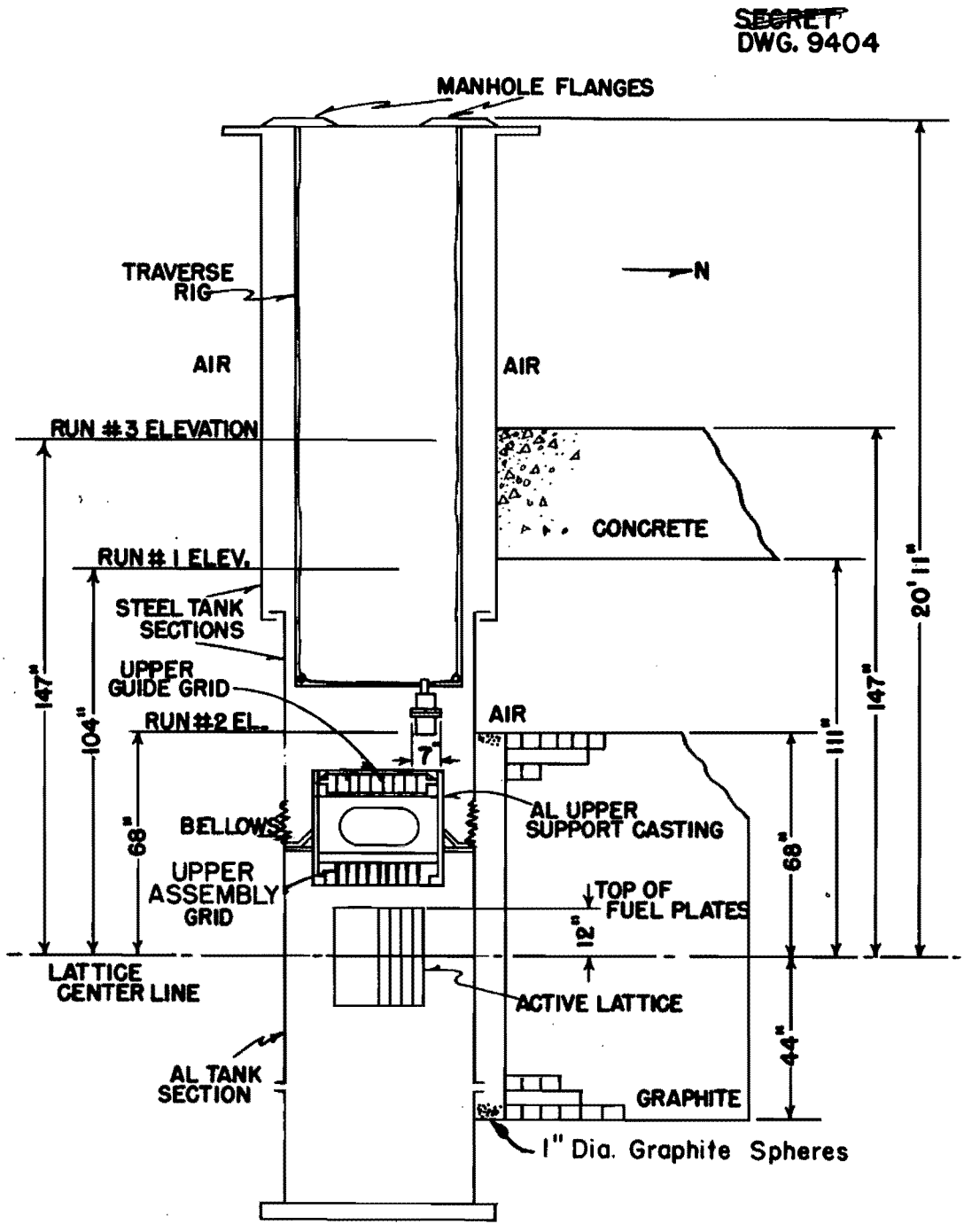



FIG. A4.JJ

CHAMBER ELEVATIONS FOR HORIZONTAL GAMMA TRAVERSES AND RELATIVE LOCATIONS OF GRAPHITE, AIR AND CONCRETE IN MTR MOCK-UP

## A4.6 REFERENCE REPORTS


(1) Huffman, J. R., *et al.*, *Feasibility Report on Clinton High Flux Pile*, MonT-433, December 1, 1947.

(2) Gall, W. R., and Mallon, D. J., *Mock-Up Design Report*, ORNL CF-49-10-185, October 27, 1949.

(3) Beall, S. E., *Procedure for Erection of MTR Mock-Up*, ORNL-692, May 18, 1950.

(4) Savage, H. C., *MTR Strain Measurements*, ORNL-472, November 20, 1949.

(5) Beall, S. E., *Interim Report of Neutron and Gamma Experiments in the MTR Mock-Up*, ORNL CF-50-5-140, May 11, 1950.

(6) Nicoll, D., *Design of Shim-Safety Rods*, ORNL CF-48-2-139, February 4, 1948.

(7) Nicoll, D., *Supplementary Report to Design of Shim-Safety Rods*, ORNL CF-49-1-65, December 31, 1948.

(8) Beall, S. E., *Shim-Safety Rod Data*, ORNL CF-49-8-188, August 17, 1949.

(9) Nicoll, D., *Design of Regulating Rods for High Flux Reactor*, ORNL CF-49-4-123, March 14, 1949.

(10) Kasper, S., *Water Flow in Materials Testing Reactor*, ORNL CF-50-6-102, June 15, 1950.

(11) Winters, C. E., *Technical Division, Section II, Report for Period Ending September 15, 1948*, ORNL CF-48-9-278, September 17, 1948.

(12) *Calibration of Control and Shutoff Rods in Gleep by Relaxation Time Measurements*, AERE-NR-130, December 5, 1947.

(13) Briggs, R. B., *A Comparison of Design Neutron Fluxes and Heat Production in the Graphite Reflector and Thermal Shield of the Materials Testing Reactor with Values Measured in the MTR Mock-Up*, ORNL CF-50-7-86, July 20, 1950.

(14) Lane, J. A., *Shielding Requirements, Top and Bottom Plugs*, ORNL CF-50-7-7, July 5, 1950.

(15) Arnette, T., and Jones, H., *Thermal Neutron Flux Distribution in a Standard Graphite Pile*, CP-2804, April 30, 1945.

(16) Weinberg, A. M., *Physics Division Quarterly Report for Period Ending February, 1948*, CNL-35, March 26, 1948.

(17) Kunstadter, J. W., "A Correction to Be Applied to the Activity of Neutron-Activity Cadmium Covered Indium Foils," *Phys. Rev.*, **78**, 484 (1950).

(18) Mann, M. M., and Martin, A. B., *Critical Experiments for High Flux Reactor*, ORNL-167, October 22, 1948.

(19) Glass, F. M., *Low Drift Electrometer*, ORNL-486, December 20, 1949.

(20) Jenks, G., Breazeale, W. M., and Hairston, J. J., *Calorimeteric Calibration of a Graphite-Walled, Cavity-Type Ionization Chamber*, ORNL-923, January 23, 1951.

(21) Clifford, C. E., and Mauney, T. H., *Gamma Attenuation Measurements Through Water in the MTR Mock-Up*, ORNL CF-50-8-85, November 3, 1950.

# Appendix 5

# BERYLLIUM

## A5.1 CHEMICAL AND MECHANICAL TESTS

When it became evident in 1947 that the beryllium supply was to be adequate, immediate tests were started to determine its corrosion resistance and mechanical properties. The first beryllium* to become available was extruded material, and most of the initial corrosion testing was done on this. The corrosion-test procedures and preliminary results are fully described in ORNL-298.[1] This and later reports emphasize that the corrosion of beryllium in simulated cooling water can be controlled to a large extent by:

1. Controlling the amounts of impurities present in the metal through improved processing techniques. Corrosion-test data indicate that these impurities are a very important factor in determining the corrosion resistance of this metal.
2. Controlling the quality of the demineralized water used as a cooling medium. A water of high specific resistance is desired as well as one relatively free from such detrimental ions as chlorides, sulfates, copper, and iron.

In general, however, data presented in ORNL-298 and in ORNL-733[2] indicate that the corrosion resistance of either extruded or hot-pressed beryllium is such that no serious corrosion problems are to be expected. As a result of the data in ORNL-733, the decision was made to use hot-pressed beryllium for the MTR rather than extruded material because:

1. The hot-pressed method allowed the manufacture of large odd-shaped pieces and hence reduced the stacking problem in the MTR.
2. Hot-pressed beryllium appeared then to have a somewhat better corrosion resistance. It is now known that there is very little difference between the two in this respect.

*The following types of beryllium are mentioned in various reports:

1. Extruded beryllium.
2. Q.M.: Normal high-purity beryllium used to produce powder for hot pressing into required shapes.
3. Q.R.M.: Recast from previous pressings of Q.M.
4. Q.M.V.: High-purity beryllium cast by a vacuum method into billets which are then pulverized into powder for hot pressing. This method reduces the oxygen content and therefore reduces the beryllium oxide. All the MTR beryllium is Q.M.V. metal.

A final report on the corrosion tests on beryllium is now in composition and will be issued sometime after March, 1951 as an ORNL report.

The mechanical tests on beryllium involved both machining tests and calculation and experimental work on thermal stresses. An example of the machining tests is given in ORNL CF-50-4-110,[3] which is a report on deep-hole drilling in beryllium. As of Jan. 1, 1951 the machining of the MTR beryllium had been 95% completed by the Y-12 Beryllium Shop with no major difficulties encountered.

Some concern was felt that the temperature differentials in the beryllium would be sufficient to set up stresses large enough to crack the metal blocks. Calculations were made for blocks containing various sizes of cooling holes. The results for the 3- by 3-in. beryllium bars with 3/16-in. axial cooling holes (pieces labeled "A" in Fig. 2.2.A) are given in ORNL CF-50-6-9.[4] These calculations showed that for MTR operation at 30 megawatts the maximum thermal stresses occurring are 12,000 psi at the surface of the bar and 4700 psi at the surface of the axial hole, well within the tensile strength of beryllium.

To check these calculations, experiments[5] were performed in which thermal stresses were set up in beryllium bars using the same basis of calculation as above. It was found experimentally that the minimum stress at failure for the hot-pressed beryllium was 28,200 psi, well above the value obtained in MTR operation.

## A5.2 MANUFACTURE AND SPECIFICATIONS OF MTR BERYLLIUM

Since a full report on the method of manufacture and chemical analysis of each piece of MTR beryllium is being prepared (Gregg, J. L., *Beryllium for the MTR,* to be issued as an ORNL report), only a brief summary and an example of the chemical analyses will be given here.

All the beryllium for the MTR is being made by hot pressing of the powder obtained from pulverizing vacuum-cast billets. The raw metal is refined and then is melted and cast in vacuum. The castings are cut on a lathe, rolled, and pulverized to produce particles that will pass through a 200-mesh screen. This fine powder is put into dies and placed in a vacuum furnace at approximately 1050°C and 100 psi pressure for 20 to 30 hr. When these pressings are removed from the furnace, they are cut into rough shapes and shipped to Y-12 for final machining.

The specifications set up for the rough sections of beryllium are given in detail in ORNL CF-50-3-114.[6] These include chemical, corrosion, and physical requirements and test procedures. Since the chemical specifications may be useful in future checking, they are given below.

> The chemical composition of the beryllium shall be such that the total danger summation (see Section A5.3) does not exceed 15 for all impurities or elements other than boron, and the boron content shall be below that which increases the total danger summation by 15. In addition, the chemical composition shall be within the following limits:

| | |
|---|---|
| Free beryllium metal assay | 98.0% min. |
| Total carbon reported as $Be_2C$ | 0.25% max. |
| Aluminum | 1500 ppm max. |
| Cobalt | 2 ppm max. |
| Copper | 500 ppm max. |
| Iron | 2000 ppm max. |
| Lithium | 3 ppm max. |
| Magnesium | 3000 ppm max. |
| Manganese | 250 ppm max. |
| Nickel | 200 ppm max. |
| Silicon | 700 ppm max. |
| Silver | 5 ppm max. |
| Zinc | 150 ppm max. |

An example of how well the metal is meeting these specifications is given in Table A5.A, which is an excerpt from the Beryllium Control Record for one pressing of the MTR metal from which were obtained five "A" pieces (see Fig. 2.4.A).

## TABLE A5.A

### SINTERED PRODUCT

| DATE OF PRESSING | DIMENSIONS OF BLOCK (in.) | WEIGHT OF BLOCK (lb) | FURNACE | COMPACTING PRESSURE (psi) | SINTERING | | | | RADIOGRAPH NO. | DENSITY |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | TEMPERATURE (°C) | TIME (hr) | ATMOSPHERE | PRESSURE ($\mu$ Hg) | | |
| 7-14-50 | 41 x 21 x 3.6 | 211.6 | Horizontal | 125 | 1050 | 30% | Vacuum | 100 | | 1.855 |

### ANALYSIS

| | (percent) | | | (parts per million) | | | | | | | | | | | | | | | TDS Excl. B | TDS Incl. B |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Assay | BeO | $Be_2C$ | Ag | Al | B | Ca | Cd | Co | Cr | Cu | Fe | Li | Mg | Mn | Ni | Si | Zn | | |
| Brush | 99.45 | 1.03 | 0.20 | -1 | 700 | 0.4 | -10 | -0.2 | -1 | 73 | 80 | 900 | -0.3 | 50 | 88 | 85 | 380 | -30 | 8.21 | 10.59 |
| N.B. | | | | ~1 | 290 | 0.9 | -20 | ND | ~1 | 180 | 27 | 980 | -1 | ~15 | 100 | ~100 | 1500 | ~20 | 9.33 | 14.69 |

Remarks: Si over limit based on New Brunswick analysis.

### AUTOCLAVE

| PIECE NO. | WEIGHT (lb) | PRESSURE (psi) | TEMPERATURE (°C) | TIME (hr) | AUTOCLAVE NO. | PLACE | GRADING | COMMENTS | DISPOSITION |
|---|---|---|---|---|---|---|---|---|---|
| A-14 | 26.03 | 550 ± 30 | 250 ± 5 | 96 | 1 | Cleveland | Excellent | No visible attack noticed | All pieces shipped on 7-31-50 to Oak Ridge on MTC No. 1346B BG/L No. AT-85193; total weight 155.98 lb |
| A-15 | 26.08 | 550 ± 30 | 250 ± 5 | 96 | 1 | Cleveland | Excellent | Same | |
| A-16 | 25.86 | 550 ± 30 | 250 ± 5 | 96 | 1 | Cleveland | Excellent | Same | |
| A-17 | 26.02 | 550 ± 30 | 250 ± 5 | 96 | 1 | Cleveland | Excellent | Same | |
| A-18 | 25.90 | 550 ± 30 | 250 ± 5 | 96 | 1 | Cleveland | Excellent | Same | |
| A-19 | 26.09 | 550 ± 30 | 250 ± 5 | 96 | 1 | Cleveland | Excellent | Same | |

## A5.3 DANGER COEFFICIENT AND TOTAL DANGER SUM (TDS) AS APPLIED TO BERYLLIUM

The definition of danger coefficient is given in the Project Handbook (CL-697, Chap. IV, p. 17) as follows: "The danger coefficient of a substance is a measure of the ability of that substance to absorb thermal neutrons compared with the ability of some reference substance to absorb thermal neutrons, or

$$K_s(i) = \frac{S_i}{S_s} = \frac{N_0\sigma_i/A_i}{N_0\sigma_s/A_s} = \frac{\sigma_i \times A_s}{\sigma_s \times A_i} \qquad (1)$$

where

$K_s$ = danger coefficient with respect to reference or standard substance (usually natural uranium).

$S_i$ = effective area presented to neutrons by 1 g of impurity ($cm^2/g$).

$S_s$ = effective area presented to neutrons by 1 g of reference substance ($cm^2/g$).

$N_0$ = number of atoms in one mole of any substance ($6.03 \times 10^{23}$ atoms/mole).

$A_i$ = atomic weight of impurity (g/mole).

$A_s$ = atomic weight of reference substance (g/mole).

$\sigma_i$ = atomic cross-section of impurity for absorption of thermal neutrons ($cm^2$/atom).

$\sigma_s$ = atomic cross-section of reference substance for absorption of thermal neutrons ($cm^2$/atom)."

The total danger sum (TDS) is defined as the value which is equal to the sum of the products of the concentration of each impurity in a substance and the danger coefficient for that impurity. It is used to determine whether the total number of thermal neutrons absorbed by the material exceeds the allowable absorption. For example, the MTR beryllium specifications call for a maximum TDS of 30, which means that the increase in absorption of thermal neutrons due to impurities in beryllium metal shall not be greater than 30%

of the absorption due to beryllium alone. Since the cross-section of beryllium is 0.01 barn, the allowable cross-section due to impurities cannot be greater than 0.003 barn, giving a total absorption cross-section of no greater than 0.013 barn.

The following example shows the calculation of danger coefficient and TDS for boron in beryllium.[7]

Using formula (1) above,

$$K_s(i) = \frac{\sigma_i \times A_s}{\sigma_s \times A_i}$$

For boron in beryllium,

$$K_{Be}(B) = \frac{705 \times 9.02}{0.01 \times 10.82} = 58{,}800$$

for the concentration of boron in beryllium expressed in percent, or 5.88 for concentration in parts per million.

If we assume 0.0002% boron in beryllium, then the TDS is 58,800 × 0.0002 = 11.76. Similarly expressed as 2 ppm, 5.88 × 2 = 11.76.

### A.5.4 REFERENCE REPORTS


(1) English, J. L., *Interim Report on the Corrosion of Beryllium in Simulated Cooling Water for the Proposed Development Reactor,* ORNL-298, March 16, 1949.

(2) Olsen, A. R., *Initial Corrosion Tests on Brush Process QM and QRM Beryllium,* ORNL-733, July 14, 1950.

(3) Cunningham, J. E., *Deep Hole Drilling of Hot-pressed Beryllium Metal,* ORNL CF-50-4-110, April 20, 1950.

(4) Lane, J. A., *Thermal Stress in Beryllium Class II Pieces,* ORNL CF-50-6-9, June 2, 1950.

(5) Savage, H. C., *Thermal Stresses in Beryllium,* ORNL-571, February 9, 1950.

(6) Mann, M. M., *Specifications for Rough Sections of Beryllium for the MTR,* ORNL CF-50-3-114, March 22, 1950.

(7) Liss, B., *Danger Coefficients for Impurities in Several Substances,* NYO-108, March 6, 1950.

## Appendix 6

# FUEL ELEMENTS

### A6.1 DEVELOPMENT

The active ingredient of the MTR, 93.5% enriched uranium, is introduced as an aluminum alloy, 14% by weight, rolled to sheet form and clad on the edges and faces with 2S aluminum. The finished core dimension is to be 0.020 by 23.75 by 2.55 in. and is to be centered with respect to edges, ends, and faces. Dimensional details are given in Table A6.A. Eighteen of these fuel plates are then furnace brazed to grooved aluminum side plates to form subassemblies. A cross-section of such an assembly is shown in Chap. 2, Fig. 2.2.A.

In preliminary work at Battelle Memorial Institute[1] and later at ORNL,[2] it was shown that uranium-aluminum alloys containing up to 30% uranium by weight could be cast and rolled without too much difficulty. Two methods of alloying were studied. The direct method consisted in melting a mixture of metallic uranium and metallic aluminum. In the reduction method, some compound of uranium — $UF_4$, $U_3O_8$, or $UO_2$ — was added to a bath of melted aluminum, together with a flux (cryolite), where it was reduced by excess aluminum. While the reduction method was used in fabrication of plates for the MTR critical experiment, the direct method was the one finally adopted[3] because it was fast and produced a smaller volume of contaminated wastes.

For cladding,[4] the alloy was at first enclosed in an aluminum "picture frame," another sheet of aluminum was wrapped over both frame and core, and the assembly was hot rolled for a total reduction of 50%. The resulting laminate usually developed large blisters at the trailing end of the sheet (when tested by annealing for 1 hr at 900°F). This was considered to be the result of a poor fit between core and frame; air entrapped in voids at the core-frame interface became compressed during rolling. Frames for these first rollings were made with hand tools and consequently were quite crude.

Later work at ORNL[5] showed that entrapment of air could be avoided if the core and frame were blanked out with a punch and die, assembled, and then

## TABLE A6.A

### Fuel Assembly Plates

| | |
|---|---|
| **Overall dimensions** | |
| Length, internal plates (in.) | 24-5/8 ± 1/64 |
| Length, external plates (in.) | 28-5/8 ± 1/64 |
| Width, before curving (in.) | 2.845 ± 0.001 |
| Thickness (in.) | 0.060 ± 0.001 |
| **Core dimensions** | |
| Length (in.) | 23¾ ± ¼ |
| Width (in.) | 2.55 ± 0.01 |
| Thickness (in.) | 0.021 ± 0.001 |
| **Composition** | |
| Total uranium in U-Al alloy (%) | 14.1 (approx.) |
| $U^{235}$ enrichment (%) | 93.2 (approx.) |
| $U^{235}$ content per plate (g) | 7.70 ± 1% |
| **Quality** | |
| Blister free | |

hot rolled 50% prior to cladding. It was also shown[6] that there was a direct relationship between absolute humidity and blistering. During periods of high humidity, about $8 \times 10^{-4}$ lb/ft$^3$ $H_2O$, very low yields of blister-free plates were obtained. In addition to the obvious cure of air-conditioning the working area, two solutions to this problem were found. Either vacuum annealing of the clad plates or heating them, while coated with a slurry of brazing flux, to 1100°F allowed the blister-forming gas (hydrogen[7]) to escape.

The fuel subassembly went through three shape changes before the present design was frozen. The first design conception was that of a spiral, about 3 in. in O.D. by about 30 in. long, made from a 0.060- by 27- by 30-in. plate. An annulus of 0.120 in. between turns provided space for cooling water. Since flow tests showed this type of assembly to be mechanically unstable, it was discarded. The second type consisted of a plate which was folded back and forth with annuli of 0.120 in. between folds. This design was discarded for two reasons: (1) The cladding became too thin on the outside of the 180° bends, and (2) there was mechanical instability during flow testing.

The third and final design consisted of narrow 3-in. curved plates brazed to aluminum side plates with cooling annuli between the active plates. A complete list of drawings concerned with the fuel assembly is given at the end of this appendix.

## A6.2 CONSTRUCTION OF PLATES

A full report on the fabrication of the active plates and assemblies is being prepared,[8] but, since this type of element is still new to the reactor business, a brief summary of these operations is given here.

The flow sheet shown in Fig. A6.A shows that operations of melting, rolling, punching, cladding, hydrogen removal, shearing, machining, and forming take place in sequence. The ingots of 14.1% U-Al alloy are made in batches containing 800 g of uranium, each batch making two ingots. After being melted and thoroughly rinsed, the charge (usually aluminum, virgin uranium, recycled U-Al alloy, and reject clad plates) is cast into two graphite molds, forming ingots of about 4.5 by 8 by 1¼ in. usable size.

[redacted]

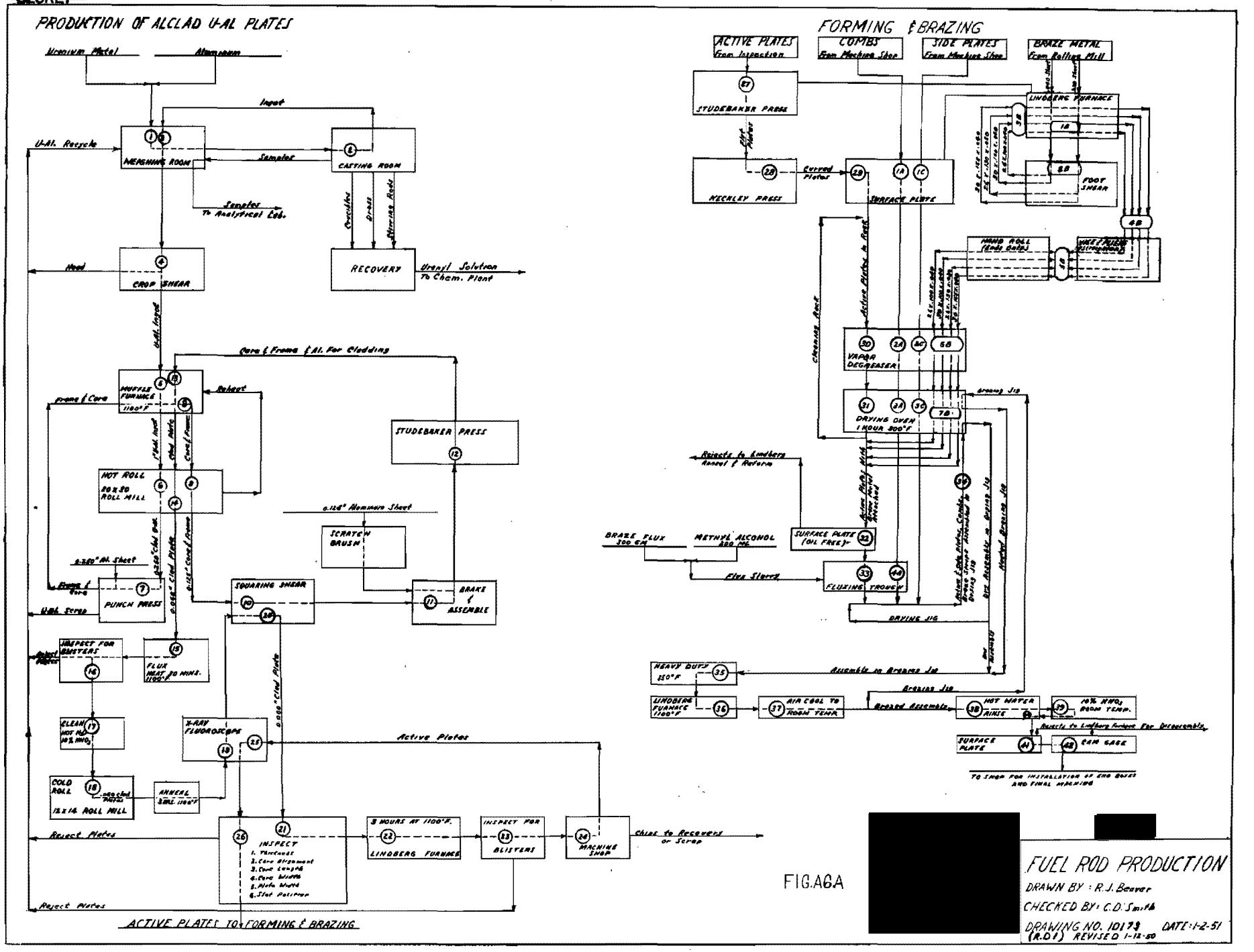



FIG.A6A

[redacted]

FUEL ROD PRODUCTION

DRAWN BY: R.J. Beaver

CHECKED BY: C.D. Smith

DRAWING NO. 10179 (R.D.) REVISED 1-12-50 DATE: 1-2-51

After preheating, the ingots are rolled to a thickness of 0.250 in. and width of 5 in. Using a double-acting punch and die, rectangular cores 2 by 2.35 in. with 1/8-in.-radius corners are blanked from the alloy slab. About thirty cores are produced per ingot. Using the same punch and die, two holes 4 by 8 in. are punched in 0.245-in. aluminum, forming two picture frames into which two cores are pressed. This assembly is preheated and rolled to 0.125 in. thickness and then cut to allow individual handling of each core. Each core and frame is then clad by insertion into 2S aluminum cover sheets 1/8 by 4 by 13 in. which are scratch-brushed and bent into a U shape with the scratched surface inside. The sandwich is then pressed flat and rolled.

In periods of high absolute humidity a considerable amount of hydrogen is absorbed by the U-Al alloy during melting and hot rolling. When this is present in sufficient quantities, it tends to collect at bond interfaces and around $UAl_4$ particles and cause blistering. To avoid this, it is necessary to coat each plate with an alcohol slurry of brazing flux and heat to 1100°F for ½ hr. The dissolution of the oxide layer on the surface apparently allows the hydrogen to escape.

After cold rolling, the exact location of the core in each plate is found by fluoroscopic examination, and the plate is then marked and sheared to rough plate dimensions. After being sheared, the plates are milled to finish dimensions and again checked by X ray so that any plates with exposed cores can be discarded. Finally, the milled plates are formed to a 5.5-in. radius in a 25-ton press.

Before further assembly, each plate is checked for longitudinal straightness and correctness of lateral curvature. This is, of course, exceedingly important because of the small clearance between the plates in the final assembly.

## A6.3 FABRICATION OF THE FUEL ELEMENTS

As shown in the flow diagram (Fig. A6.A), fabrication of fuel elements is a complex series of operations involving machining, furnace brazing, and arc welding. A finished fuel subassembly has already been shown in Fig. 2.2.B; a summary of the construction details follows.

The grooved side plates are prepared by gang milling sheets of 2S aluminum of proper thickness. The inside ends of these side plates are milled to provide proper fit for the end boxes, which are plug brazed into position. Combs milled from Alcoa No. 12 braze-clad aluminum sheet are placed at each end of the fuel assembly to aid in alignment and reinforcement at the centerline.

Before final assembly, all parts except the fuel plates are carefully cleaned by processing, as follows: (1) vapor degreasing, (2) air drying for 1 hr at 300°F, (3) acid dipping (10% $HNO_3$), (4) water rinsing, and (5) air drying as above. The fuel plates are degreased only.

In final assembly braze strips (silicon-aluminum alloy) are fastened to the edges of the fuel plates by bending over the ends. The brazing strip is then fluxed, and the fuel plate is inserted between the side plates, which are held in a jig. After all plates are inserted and the combs are in place, the jig is closed to put the side plates into correct position. The whole assembly is then dried and charged into a box type electric furnace for preheating to 800°C. After being preheated, the assembly is placed in another furnace at 1100°C to accomplish final brazing.

After cooling, the assembly is carefully inspected for spacing, plate curvature, overall dimensions and properly brazed joints. If the assembly passes this inspection, the end adapters are inserted, aligned, and plug welded into place ready for final machining.

The final machining determines the position of the fuel assembly bodies relative to each other in the reactor core according to the bottom and top supporting grids. The sequence of operations is as follows:

1. Sides. The plug welds are milled flush.
2. Top. A shaped cutter is used to mill the assembly body ends to 5½-in.-radius flush with the surface of the top plate. This operation trims the side plates and removes excess plug material. About 2-in. from the end of the assembly body the cutter is raised gradually and continues along the assembly length, providing a curved side plate edge.
3. Bottom. The bottom is prepared similarly by a second shaped cutter.

4. Top adapter. The assembly is aligned in a special machining fixture, using the top and bottom plate centerline surfaces as references. The sequence is:

   *a.* Machining of cylindrical surface.

   *b.* Machining of slot to fit a retaining ring for the helical spring and beveled ring which will be fitted later.

   *c.* Machining of the inside to the required diameter.

   *d.* Cutting away of excess metal at end.

   *e.* Drilling of a 3/8-in. hole diametrically to provide a grip for the assembly handling tools.

5. Bottom adapter. Using the same fixture as a guide, the various bosses are milled to specification.

## A6.4 REFERENCE REPORTS


(1) Grenell, L. H., and Eastwood, L., *Report on Preparations and Properties of U-Al Alloys*, CT-3294, October 10, 1945.

(2) Smith, C. D., and Adamson, G. M., *Production of U-Al Alloys*, ORNL CF-47-1-198, January 15, 1947.

(3) Mann, M. M., *MTR Project Progress Report for Period Ending August 31, 1950*, ORNL-859, November 6, 1950.

(4) Smith, C. D., *Preliminary Report on Rolling and Cladding U-Al Alloys*, ORNL CF-46-10-353, October 31, 1946.

(5) Kerze, F., *Technical Division, Section IV Progress Report for Period Ending September 15, 1948*, ORNL CF-48-10-131, October 6, 1948.

(6) Frye, J. H., Jr., *Metallurgy Division Quarterly Progress Report for Period Ending July 31, 1949*, ORNL-407, October 14, 1949.

(7) Frye, J. H., Jr., *Metallurgy Division Quarterly Progress Report for Period Ending July 31, 1950*, ORNL-827, December 4, 1950.

(8) Smith, C. D., Drosten, F. W., and Kerze, F., *Production of Fuel Assemblies for the MTR*, ORNL-951.

## A6.5 REFERENCE DRAWINGS

| DWG. NO. | TITLE |
|---|---|
| Casting Drawings | |
| TD-1543 | Upper Adapter |
| TD-1542 | Lower Adapter |
| Preassembly Machining Drawings | |
| TD-1166 | Upper Adapter |
| TD-1165 | Lower Adapter |
| Brazing Alignment Drawing | |
| TD-1167 | |
| Final Machining Drawing | |
| TD-1472 | |
| Fuel Assembly Drawings | |
| TD-966A | Side Plate |
| TD-1533 | Comb |
| TD-1534 | Standard Fuel Plate |
| TD-1535 | Outside Fuel Plate |
| TD-256 | Beveled Ring |
| TD-257 | Retaining Ring |
| TD-251 | Upper End Spring |

## Appendix 7

# PROPERTIES OF SHIELDING MATERIALS

During the analysis of the MTR shield, the values shown in Table A7.A were employed for the pertinent material properties:

**TABLE A7.A**

| | $\mu$ ($cm^{-1}$) | $L^2$ ($cm^2$) | $\tau$ ($cm^2$) | $\rho$ (g/cc) | $\lambda_n$* (cm) |
|---|---|---|---|---|---|
| Lead | 0.45 | | | 11.1 | |
| Iron | 0.235 | 1.7 | | 7.85 | 9 |
| Water | 0.025 | | | 1.0 | |
| Barytes concrete | 0.10 | | 350 | 3.5 | 8.2 |

*Asymptotic relaxation length for fission neutrons.

Some other useful information has been compiled in the balance of this appendix. Two sets of curves entitled "Mass Absorption Coefficients" are included, both of which are taken from ORNL-421. Figures A7.A and B, comprising Supplement 1 to this report, give the macroscopic total cross-section divided by density. Figures A7.C and D, which form Supplement 2, present the macroscopic absorption cross-section divided by density. In the latter case, an averaged probability for energy absorption in Compton interaction has been added to the photoelectric and pair cross-sections.

Table A7.B, a convenient summary of some declassified nuclear data, is taken from "Thermal Neutron Cross Sections and Related Data" by H. R. Kroeger* and is used with permission.

Figure A7.E presents the energy dependence of the same linear absorption coefficients, based on the total cross-sections of some materials of particular interest.

*Nucleonics*, Vol. 5, No. 4, pp. 51-54, October, 1949.

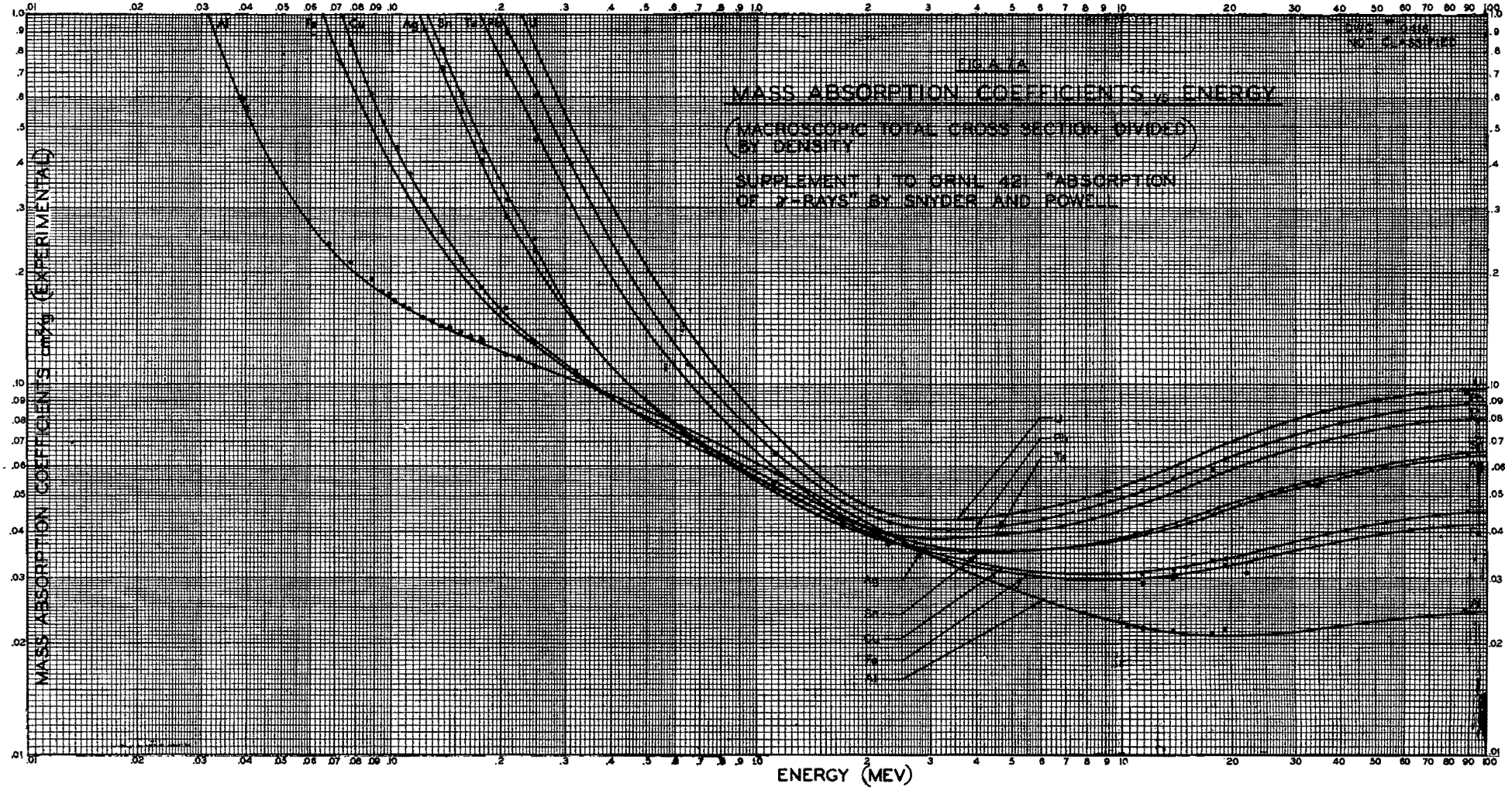


FIG. A7.A
MASS ABSORPTION COEFFICIENTS vs. ENERGY
(MACROSCOPIC TOTAL CROSS SECTION DIVIDED BY DENSITY)
SUPPLEMENT I TO ORNL 42 "ABSORPTION OF γ-RAYS" BY SNYDER AND POWELL
MASS ABSORPTION COEFFICIENTS cm²/g (EXPERIMENTAL)
ENERGY (MEV)
W
Pb
Au
Ta
Sn
Cd
Ag
Pd
Rh
Ru
Mo
Nb
Zr
Sr
Rb
Br
Se
As
Ge
Ga
Zn
Cu
Ni
Co
Fe
Mn
Cr
V
Ti
Sc
Ca
K
Cl
S
P
Si
Al
Mg
Na
Ne
F
O
N
C
B
Be
Li
He
H

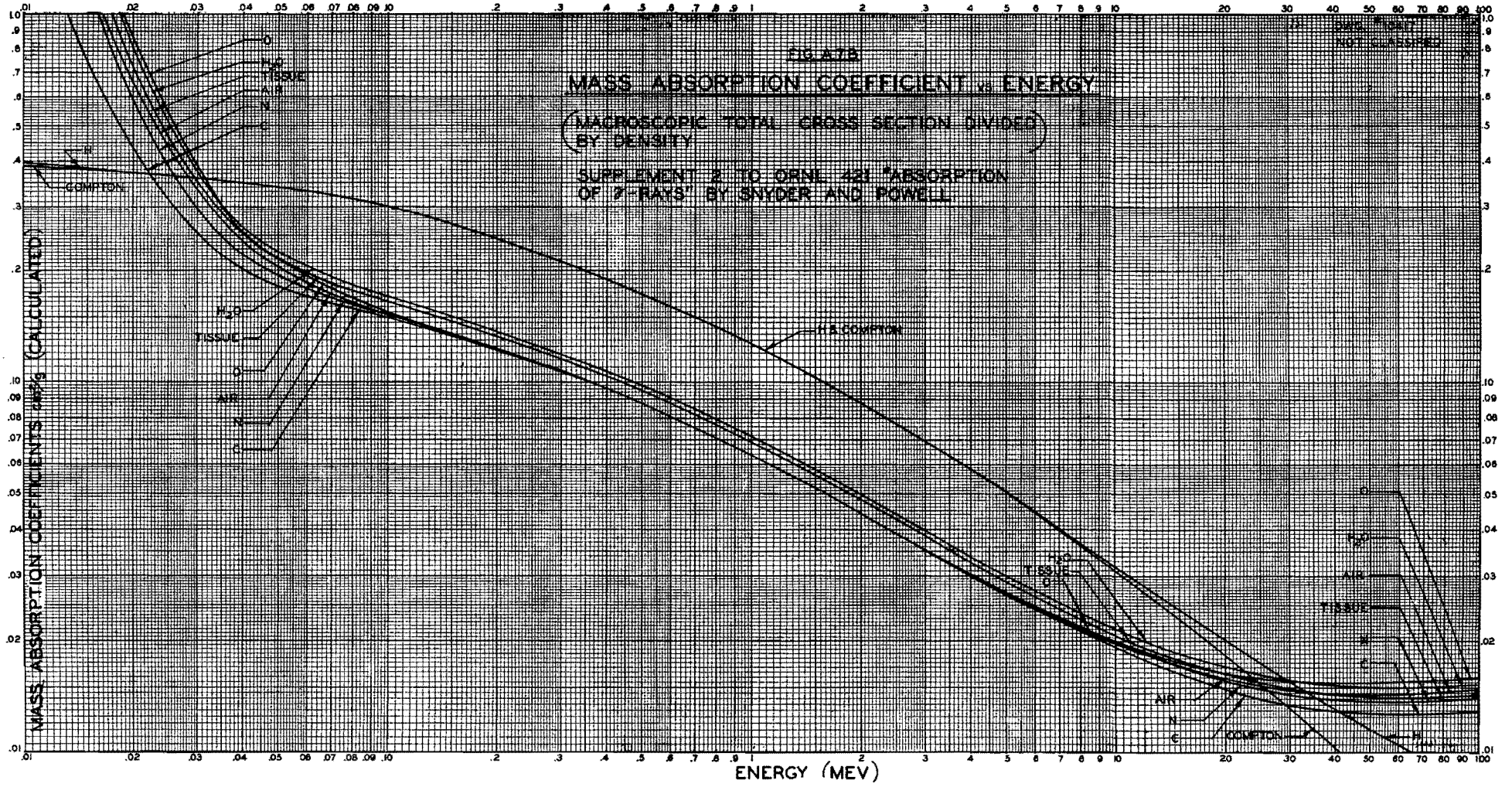


FIG. A7.3
MASS ABSORPTION COEFFICIENT vs ENERGY
(MACROSCOPIC TOTAL CROSS SECTION DIVIDED BY DENSITY)
SUPPLEMENT 2 TO ORNL 421 "ABSORPTION OF γ-RAYS" BY SNYDER AND POWELL
MASS ABSORPTION COEFFICIENTS cm²/g (CALCULATED)
ENERGY (MEV)
COMPTON
H.S. COMPTON
TISSUE
AIR
N
C
O
H₂O
PAIR PRODUCTION NOT CALCULATED

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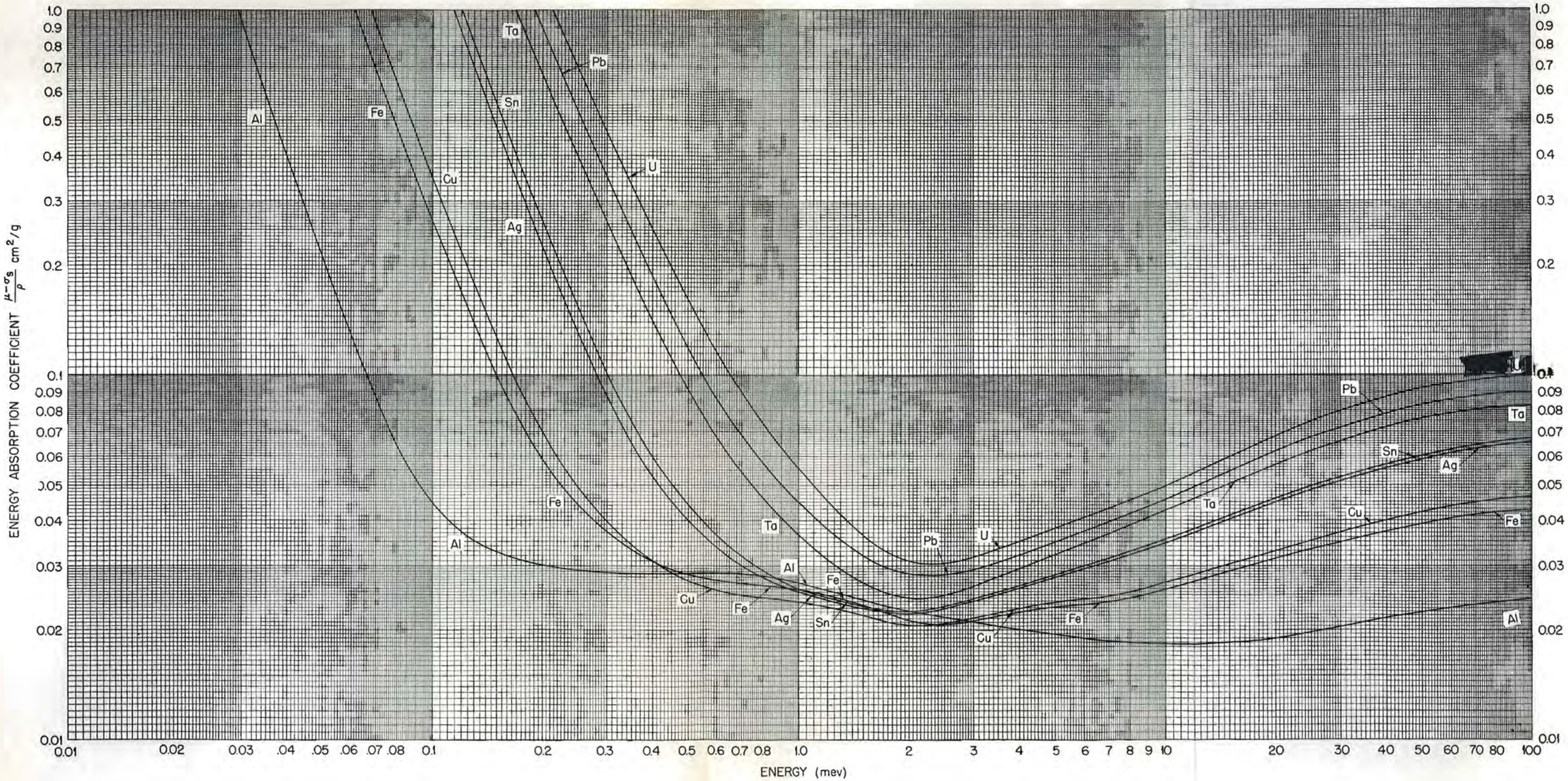
DWG. 8118 R1

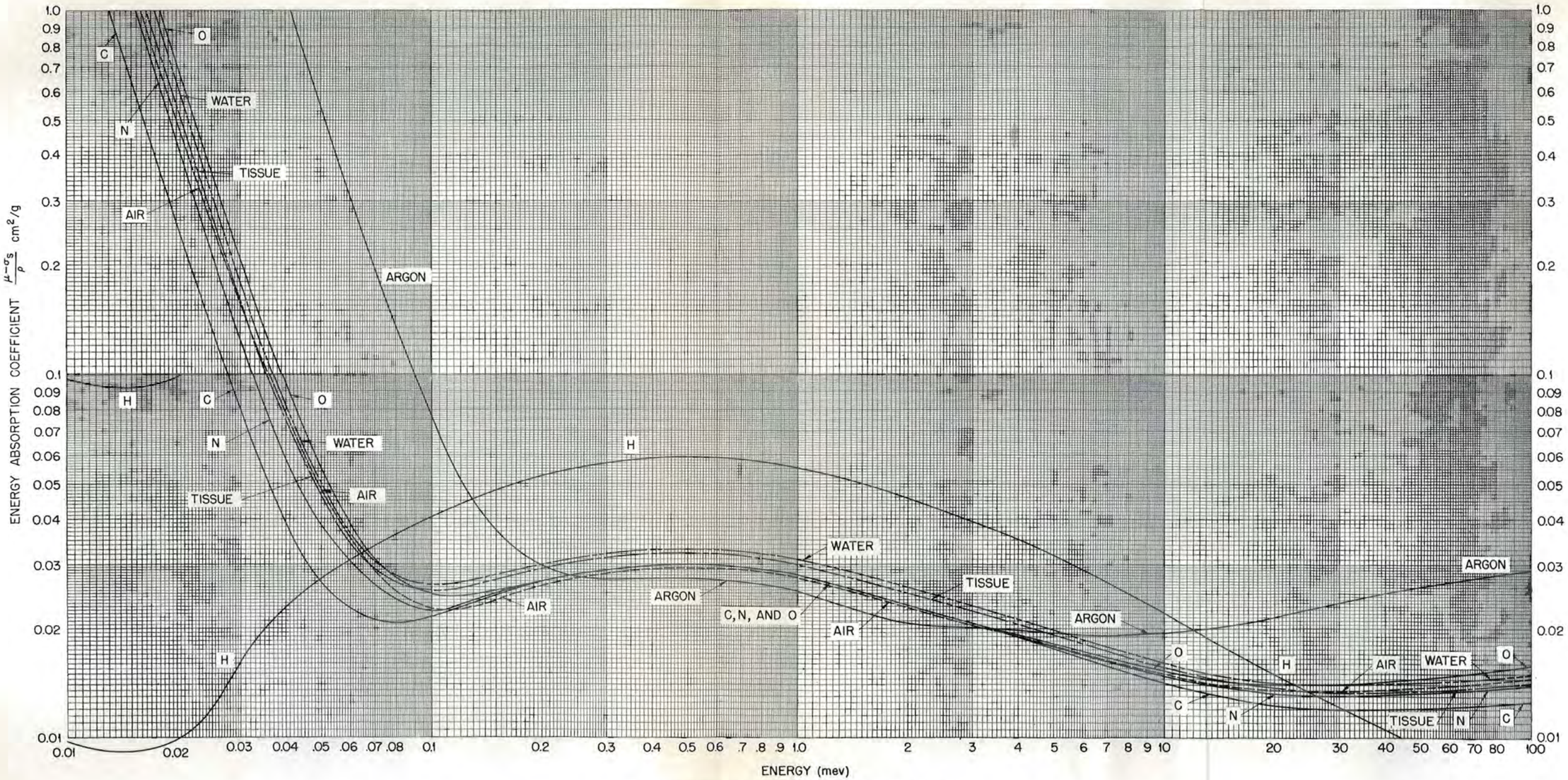



ENERGY ABSORPTION COEFFICIENTS VS ENERGY

(Macroscopic Cross-Section for Energy Absorption Divided by Density)

Supplement 3 to ORNL 421 "Absorption of $\gamma$-Rays" by Snyder and Powell.

UNCLASSIFIED
DWG. 8117 R1

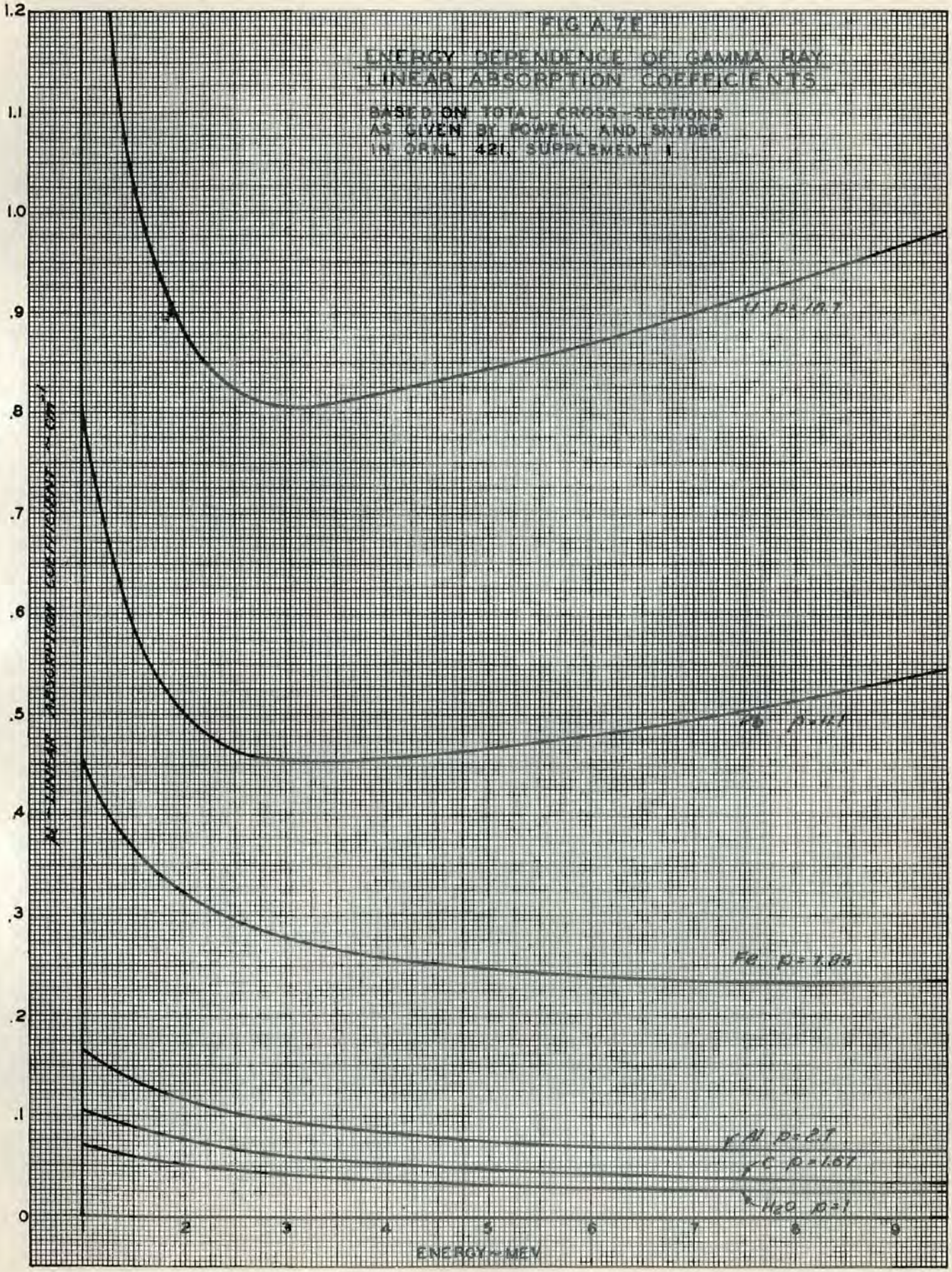



ENERGY ABSORPTION COEFFICIENTS VS ENERGY

(Macroscopic Cross-Section for Energy Absorption Divided by Density)

Supplement 4 to ORNL 421 "Absorption of $\gamma$-Rays" by Snyder and Powell

DWG. #10532
NOT CLASSIFIED



## TABLE A7.B

Symbols which are undefined in the table are:

$b$ = average cosine of the scattering angle relative to the initial direction; $\sigma_T$ = total microscopic cross-section for collision; $\sigma_S$ = microscopic cross-section for a scattering collision; $\sigma_{TR}$ = microscopic cross-section for a transport collision; $\sigma_A$ = microscopic cross-section for an absorption collision.

Cross-sections for hydrogen and deuterium are for H and D atoms in combination with O ($H_2O$ and $D_2O$). The density given for $_{86}Rn^{222}$ is for liquid radon.

The atomic weights and densities are from "Handbook of Chemistry and Physics" (Chemical Rubber Publishing Co., Cleveland). The macroscopic cross sections, mean free paths, slowing-down powers, and moderating powers are based on thermal neutron cross-section data gathered from the following sources:

1. K. Way, G. Haines, *Thermal neutron cross sections for elements and isotopes*, CNL-33, 2/29/48.
2. E. Melkonian, L. J. Rainwater, W. W. Havens, *Neutron beam spectrometer studies of oxygen*, M-2554.
3. R. G. Nucholls, *The total scattering cross-sections of deuterium and oxygen for fast neutrons*, MDDC-37, 6/17/46.
4. K. Way, G. Haines, *Tables of neutron cross-sections*, MonP-405, 10/28/47 (supplement issued 4/20/48).
5. H. H. Goldsmith, H. W. Ibser, *Neutron cross-sections of the elements*, MUC-HHG-7, 12/5/45.

| Atomic Number Z | Name | Chemical Atomic Weight A | Density ($\rho$) gm/cc, 20°C 760 mm Hg | Nuclei per cc ($\times 10^{-24}$) N | (1-b) | Avg. Log Energy Loss $\xi$ | Thermal (0.025 ev) Neutron Cross-Sections ($\sigma$) in barns | | | | Thermal (0.025 ev) Macroscopic Cross-Sections, $\Sigma = \sigma N$ | | | | Thermal (0.025 ev) Mean Free Path cm | | | | Slowing Down Power $N\sigma_S\xi$ | Moderating Ratio $(\sigma_S/\sigma_A) \times \xi$ |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | $\sigma_T$ | $\sigma_S$ | $\sigma_{TR}$ | $\sigma_A$ | $\Sigma_T$ | $\Sigma_S$ | $\Sigma_{TR}$ | $\Sigma_A$ | $\lambda_T$ | $\lambda_S$ | $\lambda_{TR}$ | $\lambda_A$ | | |
| 1 | H | 1.008 | .00008 | .00005 | .3329 | 1.0 | 46 | 46 | 15 | 0.32 | | | | | | | | | | 140 |
| 1 | D | 2.0147 | .00017 | .00005 | .6662 | .7261 | 5.3 | 5.3 | 3.5 | 0 | | | | | | | | | | |
| 2 | He | 4.003 | .00016 | .000025 | .8320 | .4281 | 1.5 | 1.5 | 1.2 | .008 | | | | | | | | | | 80 |
| 3 | Li | 6.940 | .534 | .0463 | .9031 | .2643 | 66.5 | 1.5 | 1.4 | 65 | 3.1 | .07 | .07 | 3.0 | .33 | 14 | 15 | .33 | .018 | .006 |
| 4 | Be | 9.02 | 1.84 | .1229 | .9255 | .2078 | 6.1 | 6.1 | 5.6 | .0085 | .75 | .75 | .69 | .001 | 1.3 | 1.3 | 1.5 | 950 | .156 | 150 |
| 5 | B | 10.82 | 2.535 | .1411 | .9379 | .1756 | 719 | 3.8 | 3.6 | 715 | 100 | .54 | .51 | 100 | .01 | 1.9 | 2.0 | 0.01 | .094 | .001 |
| 6 | C | 12.01 | 1.67 | .0838 | .9440 | .1589 | 4.8 | 4.8 | 4.5 | .0045 | .40 | .40 | .38 | .0004 | 2.5 | 2.5 | 2.6 | 2600 | .064 | 170 |
| 7 | N | 14.008 | .00116 | .00005 | .9520 | .1373 | 11.7 | 10 | 9.5 | 1.7 | | | | | | | | | | .81 |
| 8 | O | 16.000 | .00133 | .00005 | .9580 | .1209 | 4.1 | 4.1 | 3.9 | .001 | | | | | | | | | | 500 |
| 9 | F | 19.00 | .00158 | .00005 | .9646 | 1025 | 4 | 4 | 4 | 0.01 | | | | | | | | | | 41 |
| 10 | Ne | 20.183 | .00084 | .000025 | .9667 | .0967 | 2.8 | | | | | | | | | | | | | |
| 11 | Na | 22.997 | .9712 | .0254 | .9708 | .0852 | 4.5 | 4 | 4 | 0.45 | .11 | .10 | .10 | .011 | 8.8 | 9.8 | 9.8 | 88 | .009 | .76 |
| 12 | Mg | 24.32 | 1.741 | .0431 | .9724 | .0807 | 3.0 | 2.7 | 2.6 | 0.3 | .13 | .12 | .11 | .013 | 7.8 | 8.6 | 8.9 | 77 | .009 | .73 |
| 13 | Al | 26.97 | 2.699 | .0603 | .9751 | .0730 | 1.6 | 1.4 | 1.4 | 0.22 | .097 | .084 | .084 | .013 | 10 | 12 | 12 | 77 | .006 | .47 |
| 14 | Si | 28.06 | 2.42 | .0519 | .9760 | .0702 | 2.5 | 2.3 | 2.2 | 0.2 | .13 | .12 | .11 | .010 | 7.7 | 8.4 | 8.8 | 96 | .008 | .81 |
| 15 | P | 30.98 | 2.34 | .0455 | .9783 | .0637 | 4.4 | 4.1 | 4.0 | 0.3 | .20 | .19 | .18 | .014 | 5.0 | 5.4 | 5.5 | 73 | .012 | .87 |
| 16 | S | 32.06 | 2.0 | .0376 | .9790 | .0616 | 1.4 | 1 | 1 | 0.4 | .053 | .038 | .038 | .015 | 19 | 27 | 27 | 67 | .002 | .15 |
| 17 | Cl | 35.457 | .00295 | .00005 | .9810 | .0558 | 55 | 20 | 20 | 35 | | | | | | | | | | .032 |
| 18 | A | 39.944 | .00166 | .000025 | .9832 | .0497 | 3 | 2 | 2 | 1 | | | | | | | | | | .099 |
| 19 | K | 39.096 | .87 | .0134 | .9828 | .0507 | 4.0 | 1.5 | 1.5 | 2.5 | .054 | .020 | .020 | .034 | 19 | 50 | 50 | 30 | .001 | .030 |
| 20 | Ca | 40.08 | 1.54 | .0231 | .9832 | .0495 | 4 | 3.5 | 3.4 | 0.5 | .092 | .081 | .079 | .012 | 11 | 12 | 13 | 86 | .004 | .35 |

TABLE A7.B (Cont'd)

| Atomic Number Z | Name | Chemical Atomic Weight A | Density ($\rho$) gm/cc, 20°C 760 mm Hg | Nuclei per cc (x $10^{-24}$) N | (1-b) | Avg. Log Energy Loss $\xi$ | Thermal (0.025 ev) Neutron Cross Sections ($\sigma$) in barns | | | | Thermal (0.025 ev) Macroscopic Cross Sections, $\Sigma = \sigma N$ | | | | Thermal (0.025 ev) Mean Free Path cm | | | | Slowing Down Power $N\sigma_S\xi$ | Moderating Ratio $(\sigma_S/\sigma_A) \times \xi$ |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | $\sigma_T$ | $\sigma_S$ | $\sigma_{TR}$ | $\sigma_A$ | $\Sigma_T$ | $\Sigma_S$ | $\Sigma_{TR}$ | $\Sigma_A$ | $\lambda_T$ | $\lambda_S$ | $\lambda_{TR}$ | $\lambda_A$ | | |
| 21 | Sc | 45.10 | 2.5 | .0334 | .9851 | .0441 | | | | 22 | | | | .74 | | | | 1.4 | | |
| 22 | Ti | 47.90 | 4.5 | .0566 | .9860 | .0415 | 11 | 6 | 6 | 5 | .62 | .34 | .34 | .28 | 1.6 | 2.9 | 2.9 | 3.5 | .014 | .050 |
| 23 | V | 50.95 | 5.96 | .0705 | .9868 | .0391 | 12 | 7 | 7 | 4.5 | .85 | .49 | .49 | .32 | 1.2 | 2.0 | 2.0 | 3.2 | .019 | .061 |
| 24 | Cr | 52.01 | 6.92 | .0801 | .9871 | .0383 | 6.5 | 4 | 4 | 2.5 | .52 | .32 | .32 | .20 | 1.9 | 3.1 | 3.1 | 5.0 | .012 | .061 |
| 25 | Mn | 54.93 | 7.2 | .0789 | .9878 | .0363 | 15.3 | 2.3 | 2.3 | 13 | 1.2 | .18 | .18 | 1.0 | .83 | 5.5 | 5.5 | .95 | .007 | .006 |
| 26 | Fe | 55.85 | 7.85 | .0847 | .9880 | .0357 | 13.5 | 11 | 11 | 2.5 | 1.1 | .93 | .93 | .21 | .88 | 1.1 | 1.1 | 4.7 | .033 | .16 |
| 27 | Co | 58.94 | 8.71 | .0890 | .9886 | .0338 | 41 | 5 | 5 | 36 | 3.7 | .45 | .45 | 3.2 | .27 | 2.3 | 2.3 | .31 | .015 | .005 |
| 28 | Ni | 58.69 | 8.9 | .0913 | .9885 | .0340 | 22 | 17 | 17 | 4.5 | 2.0 | 1.6 | 1.6 | .41 | .50 | .65 | .65 | 2.4 | .053 | .13 |
| 29 | Cu | 63.57 | 8.89 | .0842 | .9894 | .0314 | 10.2 | 7.2 | 7.1 | 3 | .86 | .61 | .60 | .25 | 1.2 | 1.7 | 1.7 | 4.0 | .019 | .075 |
| 30 | Zn | 65.38 | 7.19 | .0662 | .9897 | .0305 | 4.5 | 3.6 | 3.6 | 0.9 | .30 | .24 | .24 | .060 | 3.4 | 4.2 | 4.2 | 17 | .007 | .12 |
| 31 | Ga | 69.72 | 5.9 | .0510 | .9904 | .0287 | 18 | 16 | 16 | 2.2 | .92 | .82 | .82 | .11 | 1.1 | 1.2 | 1.2 | 8.9 | .023 | .21 |
| 32 | Ge | 72.60 | 5.46 | .0453 | .9907 | .0275 | 9 | 6 | 6 | 2.8 | .41 | .27 | .27 | .13 | 2.5 | 3.7 | 3.7 | 7.9 | .007 | .059 |
| 33 | As | 74.91 | 5.73 | .0461 | .9910 | .0267 | 8.8 | 4.5 | 4.5 | 4.3 | .41 | .21 | .21 | .20 | 2.5 | 4.8 | 4.8 | 5.1 | .006 | .028 |
| 34 | Se | 78.96 | 4.5 | .0343 | .9915 | .0253 | 28 | 13 | 13 | 15 | .96 | .45 | .45 | .52 | 1.0 | 2.2 | 2.2 | 1.9 | .011 | .022 |
| 35 | Br | 79.916 | 3.12 | .0235 | .9916 | .0250 | 11.5 | 5 | 5 | 6.5 | .27 | .12 | .12 | .15 | 3.7 | 8.5 | 8.5 | 6.5 | .003 | .019 |
| 36 | Kr | 83.7 | .00348 | .000025 | .9920 | .0239 | 27 | 27 | 27 | 0.05 | | | | | | | | | | 13 |
| 37 | Rb | 85.48 | 1.532 | .0108 | .9921 | .0234 | 12.6 | 12 | 12 | 0.6 | .14 | .13 | .13 | .006 | 7.4 | 7.7 | 7.7 | 150 | .003 | .47 |
| 38 | Sr | 87.63 | 2.6 | .0179 | .9923 | .0228 | 11.3 | 9.5 | 9.4 | 1.8 | .20 | .17 | .17 | .032 | 5.0 | 5.9 | 5.9 | 31 | .004 | .12 |
| 39 | Y | 88.92 | 3.8 | .0257 | .9924 | .0225 | 4.2 | 3 | 3 | 1.2 | .11 | .08 | .08 | .031 | 9.3 | 13 | 13 | 33 | .002 | .057 |
| 40 | Zr | 91.22 | 6.44 | .0425 | .9926 | .0220 | 8.4 | 8 | 8 | 0.4 | .36 | .34 | .34 | .017 | 2.8 | 2.9 | 2.9 | 59 | .007 | .44 |
| 41 | Cb | 92.91 | 8.4 | .0545 | .9928 | .0216 | 8.2 | 7 | 7 | 1.2 | .45 | .38 | .38 | .065 | 2.2 | 2.6 | 2.6 | 15 | .008 | .13 |
| 42 | Mo | 95.95 | 10.2 | .0640 | .9930 | .0209 | 9.6 | 7 | 7 | 2.6 | .61 | .45 | .45 | .17 | 1.6 | 2.2 | 2.2 | 6.0 | .009 | .056 |
| 43 | Tc | 99 | | | .9932 | .0202 | | | | | | | | | | | | | | |
| 44 | Ru | 101.7 | 12.2 | .0723 | .9934 | .0197 | 9 | 6 | 6 | 3 | .65 | .43 | .43 | .22 | 1.5 | 2.3 | 2.3 | 4.6 | .009 | .039 |
| 45 | Rh | 102.91 | 12.5 | .0732 | .9935 | .0195 | 155 | 6 | 6 | 149 | 11 | .44 | .44 | 11 | .088 | 2.3 | 2.3 | .092 | .009 | .001 |
| 46 | Pd | 106.7 | 12.16 | .0686 | .9937 | .0188 | 10.5 | 4.5 | 4.5 | 6 | .72 | .31 | .31 | .41 | 1.4 | 3.2 | 3.2 | 2.4 | .006 | .014 |
| 47 | Ag | 107.880 | 10.5 | .0586 | .9938 | .0186 | 66 | 6 | 6 | 60 | 3.9 | .35 | .35 | 3.5 | .26 | 2.8 | 2.8 | .28 | .007 | .002 |
| 48 | Cd | 112.41 | 8.65 | .0463 | .9940 | .0178 | 2500 | 6 | 6 | 2500 | 120 | .28 | .28 | 120 | .009 | 3.6 | 3.6 | .009 | .005 | 0 |
| 49 | In | 114.76 | 7.28 | .0382 | .9941 | .0175 | 192 | 2 | 2 | 190 | 7.3 | .08 | .08 | 7.3 | .14 | 13 | 13 | .14 | .001 | 0 |
| 50 | Sn | 118.7 | 7.3 | .0370 | .9943 | .0169 | 4.6 | 4 | 4 | 0.6 | .17 | .15 | .15 | .022 | 5.9 | 6.8 | 6.8 | 46 | .003 | .11 |
| 51 | Sb | 121.76 | 6.691 | .0331 | .9945 | .0165 | 9 | 4 | 4 | 5 | .30 | .13 | .13 | .17 | 3.4 | 7.6 | 7.6 | 6.0 | .002 | .013 |
| 52 | Te | 127.61 | 6.25 | .0295 | .9947 | .0157 | 9 | 5 | 5 | 4 | .27 | .15 | .15 | .12 | 3.8 | 6.8 | 6.8 | 8.5 | .002 | .020 |
| 53 | I | 126.92 | 4.94 | .0234 | .9947 | .0158 | 10 | 3 | 3 | 7 | .23 | .07 | .07 | .16 | 4.3 | 14 | 14 | 6.1 | .001 | .007 |
| 54 | Xe | 131.3 | .00546 | .000025 | .9949 | .0153 | 37 | | | | | | | | | | | | | |

## TABLE A7.B (Cont'd)

| Atomic Number Z | Name | Chemical Atomic Weight A | Density ($\rho$) gm/cc, 20°C 760 mm Hg | Nuclei per cc ($\times 10^{-24}$) N | (1-b) | Avg. Log Energy Loss $\xi$ | Thermal (0.025 ev) Neutron Cross Sections ($\sigma$) in barns | | | | Thermal (0.025 ev) Macroscopic Cross Sections, $\Sigma = \sigma N$ | | | | Thermal (0.025 ev) Mean Free Path cm | | | | Slowing Down Power $N\sigma_S\xi$ | Moderating Ratio $(\sigma_S/\sigma_A) \times \xi$ |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | $\sigma_T$ | $\sigma_S$ | $\sigma_{TR}$ | $\sigma_A$ | $\Sigma_T$ | $\Sigma_S$ | $\Sigma_{TR}$ | $\Sigma_A$ | $\lambda_T$ | $\lambda_S$ | $\lambda_{TR}$ | $\lambda_A$ | | |
| 55 | Cs | 132.91 | 1.873 | .0085 | .9949 | .0151 | 50 | 24 | 24 | 26 | .43 | .20 | .20 | .22 | 2.4 | 4.9 | 4.9 | 4.5 | .003 | .014 |
| 56 | Ba | 137.36 | 3.78 | .0166 | .9951 | .0146 | 9.3 | 8 | 8 | 1.3 | .15 | .13 | .13 | .022 | 6.5 | 7.5 | 7.5 | 46 | .022 | .090 |
| 57 | La | 138.92 | 6.15 | .0267 | .9952 | .0145 | 28 | 20 | 20 | 8 | .75 | .53 | .53 | .21 | 1.3 | 1.9 | 1.9 | 4.7 | .008 | .036 |
| 58 | Ce | 140.13 | 6.9 | .0297 | .9952 | .0143 | 29 | 23 | 23 | 6 | .86 | .68 | .68 | .18 | 1.2 | 1.5 | 1.5 | 5.6 | .010 | .055 |
| 59 | Pr | 140.92 | 6.475 | .0277 | .9952 | .0142 | 15 | 5 | 5 | 10 | .42 | .14 | .14 | .28 | 2.4 | 7.2 | 7.2 | 3.6 | .002 | .007 |
| 60 | Nd | 144.27 | 6.96 | .0291 | .9953 | .0139 | 100 | | | | 2.9 | | | | .34 | | | | | |
| 61 | Pm | 147 | | | .9954 | .0137 | | | | 60 | | | | | | | | | | |
| 62 | Sm | 150.43 | 7.7 | .0308 | .9955 | .0134 | 8900 | | | 8900 | 270 | | | 270 | .004 | | | .004 | | |
| 63 | Eu | 152.0 | 5.24 | .0208 | .9956 | .0132 | 6000 | | | 6000 | 120 | | | 120 | .008 | | | .008 | | |
| 64 | Gd | 156.9 | 7.94 | .0305 | .9957 | .0128 | 40000 | | | 40000 | 1200 | | | 1200 | .001 | | | .001 | | |
| 65 | Tb | 159.2 | 8.33 | .0315 | .9958 | .0126 | | | | 10 | | | | .32 | | | | 3.2 | | |
| 66 | Dy | 162.46 | 8.56 | .0317 | .9959 | .0124 | 1200 | | | 1200 | 38 | | | 38 | .026 | | | .026 | | |
| 67 | Ho | 164.94 | 8.76 | .0320 | .9959 | .0122 | 65 | 5 | 5 | 60 | 2.1 | .16 | .16 | 1.9 | .48 | 6.3 | 6.3 | .52 | .002 | .001 |
| 68 | Er | 167.2 | 9.16 | .0330 | .9960 | .0120 | 200 | | | | 6.6 | | | | .15 | | | | | |
| 69 | Tm | 169.4 | 9.34 | .0332 | .9960 | .0119 | 120 | 20 | 20 | 100 | 4.0 | .66 | .66 | 3.3 | .25 | 1.5 | 1.5 | .30 | .008 | .002 |
| 70 | Yb | 173.04 | 7.01 | .0244 | .9961 | .0116 | 60 | | | | 1.5 | | | | .69 | | | | | |
| 71 | Lu | 174.99 | 9.74 | .0335 | .9962 | .0115 | 170 | 30 | 30 | 140 | 5.7 | 1.0 | 1.0 | 4.7 | .18 | 1.0 | 1.0 | .21 | .012 | .002 |
| 72 | Hf | 178.6 | 11.4 | .0384 | .9962 | .0113 | 125 | 20 | 20 | 105 | 4.8 | .77 | .77 | 4.0 | .21 | 1.3 | 1.3 | .25 | .009 | .002 |
| 73 | Ta | 180.88 | 16.6 | .0553 | .9963 | .0111 | 25 | 5 | 5 | 20 | 1.4 | .28 | .28 | 1.1 | .73 | 3.6 | 3.6 | .90 | .003 | .003 |
| 74 | W | 183.92 | 19.3 | .0632 | .9963 | .0109 | 23 | 5 | 5 | 18 | 1.5 | .32 | .32 | 1.1 | .69 | 3.2 | 3.2 | .88 | .003 | .003 |
| 75 | Re | 186.31 | 20 | .0647 | .9964 | .0108 | 98 | 13 | 13 | 85 | 6.3 | .84 | .84 | 5.5 | .16 | 1.2 | 1.2 | .18 | .009 | .002 |
| 76 | Os | 190.2 | 22.48 | .0712 | .9965 | .0106 | 32 | 15 | 15 | 17 | 2.3 | 1.1 | 1.1 | 1.2 | .44 | .94 | .94 | .83 | .011 | .009 |
| 77 | Ir | 193.1 | 22.42 | .0699 | .9965 | .0104 | 464 | 14 | 14 | 450 | 32 | .98 | .98 | 31 | .031 | 1.0 | 1.0 | .032 | .010 | 0 |
| 78 | Pt | 195.23 | 21.37 | .0659 | .9966 | .0103 | 18 | 10 | 10 | 8 | 1.2 | .66 | .66 | .53 | .84 | 1.5 | 1.5 | 1.9 | .007 | .013 |
| 79 | Au | 197.2 | 19.3 | .0589 | .9966 | .0102 | 102 | 6.5 | 6.5 | 95 | 6.0 | 38 | .38 | 5.6 | .17 | 2.6 | 2.6 | .18 | .004 | .001 |
| 80 | Hg | 200.61 | 13.546 | .0407 | .9966 | .0100 | 415 | 15 | 15 | 400 | 17 | .61 | .61 | 16 | .059 | 1.6 | 1.6 | .061 | .006 | 0 |
| 81 | Tl | 204.39 | 11.86 | .0349 | .9967 | .0098 | 12 | 9 | 9 | 3 | .42 | .31 | .31 | .11 | 2.4 | 3.2 | 3.2 | 9.5 | .003 | .029 |
| 82 | Pb | 207.21 | 11.347 | .0330 | .9968 | .0097 | 8 | 8 | 8 | 0.2 | .26 | .26 | .26 | .007 | 3.8 | 3.8 | 3.8 | 150 | .003 | .39 |
| 83 | Bi | 209.00 | 9.8 | .0282 | .9968 | .0096 | 6.5 | 6.5 | 6.5 | 0.015 | .18 | .18 | .18 | .0004 | 5.5 | 5.5 | 5.5 | 2400 | .002 | 4.2 |
| 84 | Po | 210 | | | .9968 | .0096 | | | | | | | | | | | | | | |
| 85 | At | 211 | | | .9968 | .0095 | | | | | | | | | | | | | | |
| 86 | Rn | 222 | 4.4 | .0119 | .9970 | .0091 | | | | | | | | | | | | | | |

**TABLE A7.B$_1$ (Cont'd)**

| Atomic Number Z | Name | Chemical Atomic Weight A | Density ($\rho$) gm/cc, 20°C 760 mm Hg | Nuclei per cc (x $10^{-24}$) N | (1-b) | Avg. Log Energy Loss $\xi$ | Thermal (0.025 ev) Neutron Cross Sections ($\sigma$) in barns | | | | Thermal (0.025 ev) Macroscopic Cross Sections, $\Sigma = \sigma N$ | | | | Thermal (0.025 ev) Mean Free Path cm | | | | Slowing Down Power $N\sigma_S\xi$ | Moderating Ratio $(\sigma_S/\sigma_A) \times \xi$ |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | $\sigma_T$ | $\sigma_S$ | $\sigma_{TR}$ | $\sigma_A$ | $\Sigma_T$ | $\Sigma_S$ | $\Sigma_{TR}$ | $\Sigma_A$ | $\lambda_T$ | $\lambda_S$ | $\lambda_{TR}$ | $\lambda_A$ | | |
| 87 | Fr | 223 | | | .9970 | .0090 | | | | | | | | | | | | | | |
| 88 | Ra | 226.05 | 5 | .0133 | .9970 | .0089 | | | | | | | | | | | | | | |
| 89 | Ac | 227 | | | .9970 | .0089 | | | | | | | | | | | | | | |
| 90 | Th | 232.12 | 11.5 | .0298 | .9971 | .0087 | | | | | | | | | | | | | | |
| 91 | Pa | 230.9 | | | .9971 | .0087 | | | | | | | | | | | | | | |
| 92 | U | 238.07 | 18.7 | .0473 | .9972 | .0084 | | | | | | | | | | | | | | |
| 93 | Np | 237 | | | .9972 | .0085 | | | | | | | | | | | | | | |
| 94 | Pu | 239 | | | .9972 | .0084 | | | | | | | | | | | | | | |
| 95 | Am | 241 | | | .9972 | .0083 | | | | | | | | | | | | | | |
| 96 | Cm | 242 | | | .9972 | .0083 | | | | | | | | | | | | | | |

## Appendix 8

# SHIELDING CALCULATION AIDS

The data in Figs. A8.A through F are included with the expectation that they may be of service in reading Section 4.6.3 as well as in connection with the solution of other problems. The nomenclature employed in the figures is presented in the following tabulation.

The shield is presumed to consist of one or more slabs of thickness $t_1$, $t_2$, $t_3$, ... . The source elements are assumed to be isotropic emitters.

The following symbols are used:

$\phi_E$ = undirected energy flux (Mev/cm$^2$-sec)

$E$ = energy of the primary $\gamma$ rays (Mev)

$B$ = build-up factor to correct for multiple scattering

$Q_0$ = source strength of a point source ($\gamma$ rays/sec)

$Q_L$ = source strength of a line source ($\gamma$ rays/cm-sec)

$Q_A$ = source strength of a plane source ($\gamma$ rays/cm$^2$-sec)

$Q_V$ = source strength of a volume source ($\gamma$ rays/cm$^3$-sec)

$\mu_R$; $\mu_s$; $\mu_1$, $\mu_2$ = absorption coefficients of receptor; source; shield (cm$^{-1}$)

$b_1 = \mu_1 t_1 + \mu_2 t_2 + \ldots$

$b_2 = b_1 + \mu_s Z$

$b_3 = b_1 + \mu_s h$

$Z$ = effective self-absorption distance (cm)

$C$ = radius of disk or cylindrical source (cm)

$l$, $a$, $h$, $d$ = dimensions as indicated (cm)

$$F_0(b) = \int_b^\infty \frac{e^{-y}\,dy}{y}$$

$$F_1(b) = b \int_b^{\infty} \frac{e^{-y}}{y^2}$$

Note: $F_1(0) = 1$

$$F(\phi, b) = \int_0^{\phi} e^{-b} \sec \phi \, d\phi$$

Note: $F(\phi, 0) = \phi$ (in radians)

**NOT CLASSIFIED** Dwg. # 8182

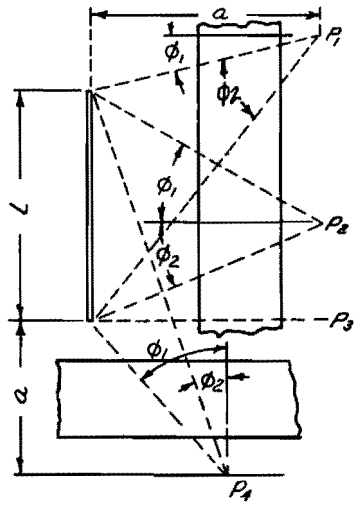
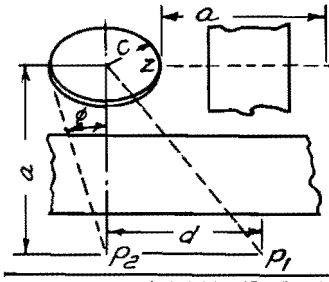
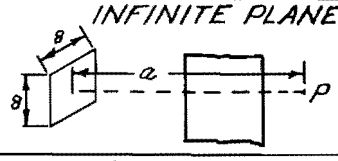
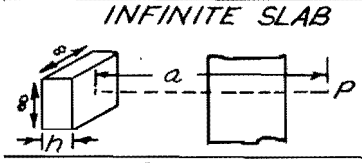
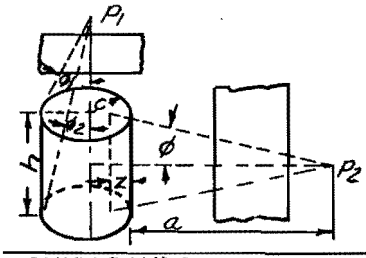
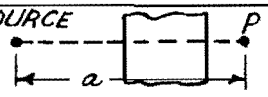
| SOURCE | ENERGY FLUX (mev/cm²/sec) | SPECIAL CONDITIONS |
|---|---|---|
| STRAIGHT LINE  | @$P_1$ $\phi_E = B\,\frac{Q_L E}{4\pi a}\left[\mathcal{F}(\phi_2,b_1)-\mathcal{F}(\phi_1,b_1)\right]$ | |
| | @$P_2$ $\phi_E = B\,\frac{Q_L E}{4\pi a}\left[\mathcal{F}(\phi_2,b_1)+\mathcal{F}(\phi_1,b_1)\right]$ | |
| | @$P_2$ $\phi_E = B\,\frac{Q_L E}{2\pi a}\,\mathcal{F}(\phi,b_1)$ | $\phi_1=\phi_2$ |
| | @$P_2$ $\phi_E = B\,\frac{Q_L E}{2\pi a}\,\mathcal{F}(\frac{\pi}{2},b_1)$ | $L=\infty,\ \phi_1=\phi_2=\frac{\pi}{2}$ |
| | @$P_3$ $\phi_E = B\,\frac{Q_L E}{4\pi a}\,\mathcal{F}(\phi_1,b_1)$ | $\phi_2=0$ |
| | @$P_3$ $\phi_E = B\,\frac{Q_L E}{4\pi a}\,\mathcal{F}(\frac{\pi}{2},b_1)$ | $\phi_2=0,\ L=\infty$ |
| | @$P_4$ $\phi_E = B\,\frac{Q_L E}{4\pi a}\left[\mathcal{F}(\phi_1,b_1)-\mathcal{F}(\phi_2,b_1)\right]$ | |
| DISC. SOURCE  | @$P_1$ $\phi_E = B\,\frac{Q_A E}{4}\left\{\ln_e \frac{c^2+a^2-d^2+\sqrt{(c^2+a^2-d^2)^2+4a^2d^2}}{2a^2} - \frac{2b_1}{a}\left[\sqrt{a^2+d^2+c^2-\sqrt{a^2+d^2}}\right]+\frac{1}{2}\left(\frac{cb_1}{a}\right)^2\right\}$ | |
| | @$P_2$ $\phi_E = B\,\frac{Q_A E}{2}\left[F_0(b_1)-F_0(b_1\sec\phi)\right]$ | |
| | @$P_3$ $\phi_E = B\,\frac{Q_A E c^2}{4(a+z)^2}\,e^{-bz}$ | |
| | | |
| INFINITE PLANE  | $\phi_E = B\,\frac{Q_A E}{2}\,F_0(b_1)$ | |
| | | |
| INFINITE SLAB  | $\phi_E = B\,\frac{Q_V E}{2\mu_S}\left[F_1(b_1)-F_1(b_3)\right]$ | |
| | | |
| CYLINDRICAL SOURCE  | @$P_1$ $\phi_E = B\,\frac{Q_V E}{2\mu_S}\left[F_1(b_1)-F_1(b_3)+\frac{F_1(b_3\sec\phi_1)}{\sec\phi_1}-\frac{F_1(b_1\sec\phi_1)}{\sec\phi_1}\right]$ | Upper Limit |
| | @$P_1$ $\phi_E = B\,\frac{Q_V E}{2\mu_S}\left[F_1(b_1)-F_1(b_3)+\frac{F_1(b_3\sec\phi_2)}{\sec\phi_2}-\frac{F_1(b_1\sec\phi_2)}{\sec\phi_2}\right]$ | Lower Limit |
| | @$P_2$ $\phi_E = B\,\frac{Q_V E c^2}{2(a+z)}\,\mathcal{F}(\phi,b_2)$ | $P_2$ opposite center |
| | | |
| POINT SOURCE  | $\phi_E = B\,\frac{Q_0 E e^{-b}}{4\pi a^2}$ | |

FIGURE A8.A

DWG. 8183

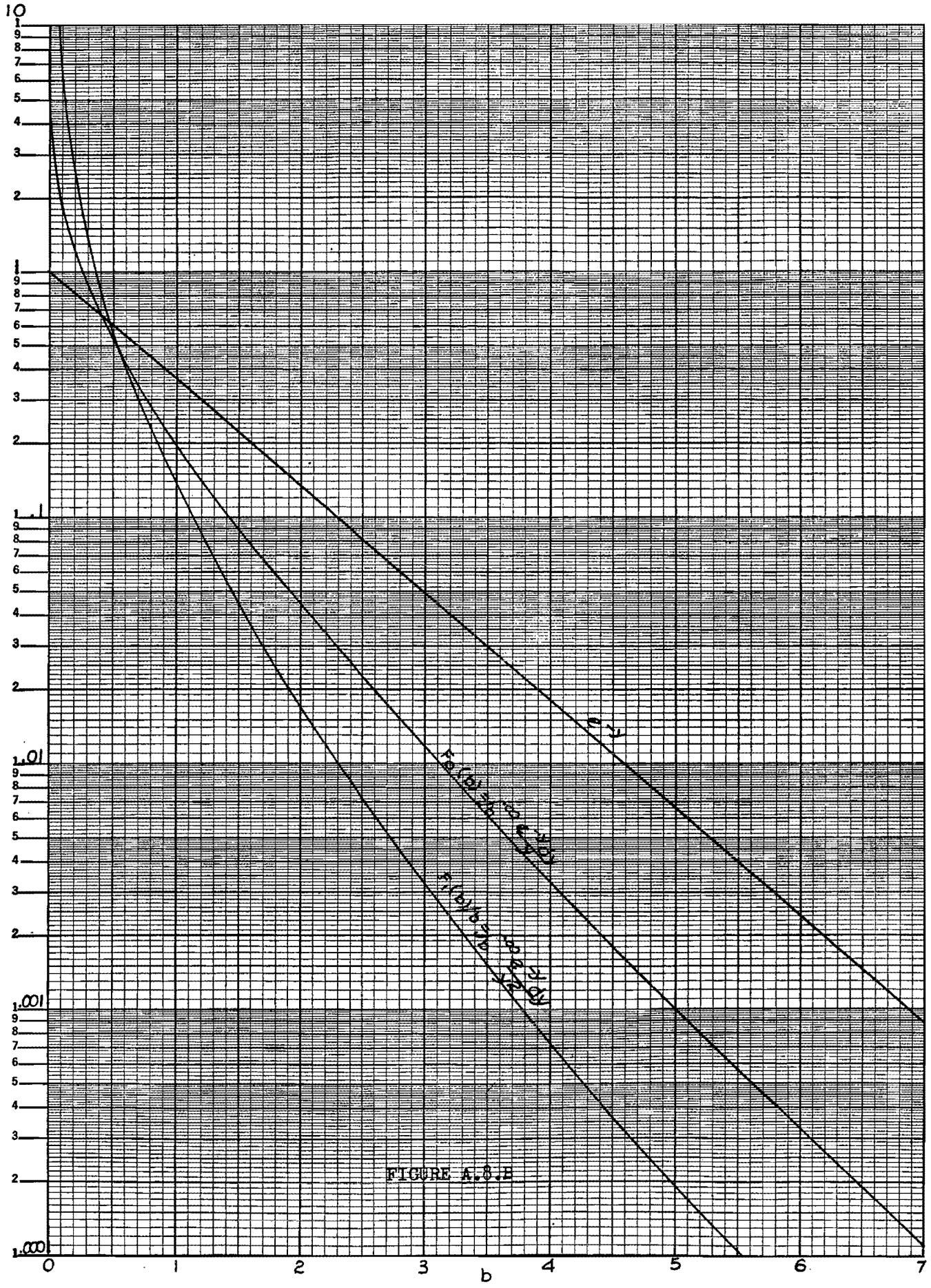



FIGURE A.8.E

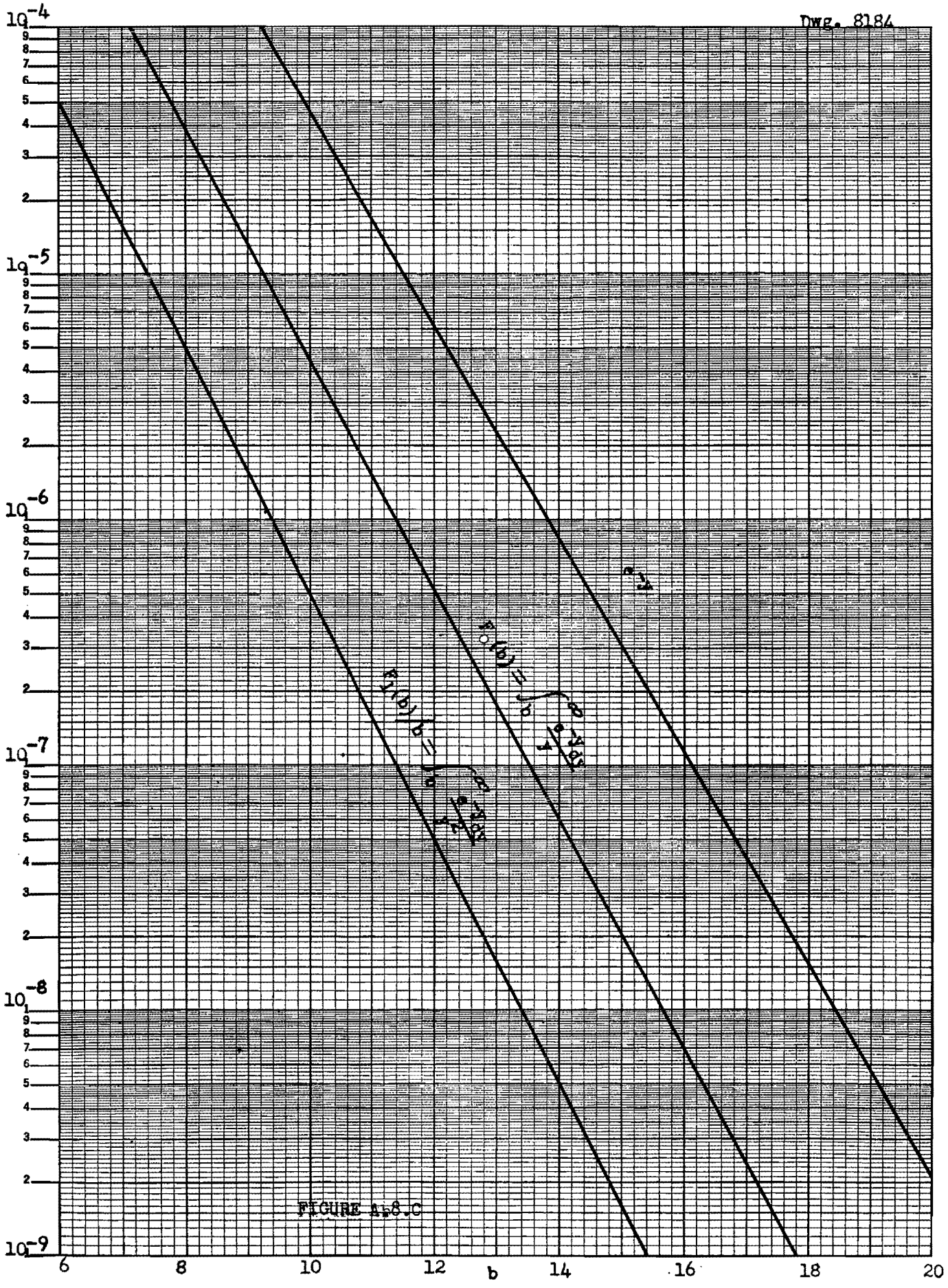

Dwg. 8184
$10^{-4}$
$10^{-5}$
$10^{-6}$
$10^{-7}$
$10^{-8}$
$10^{-9}$
6
8
10
12
14
16
18
20
b
$F_2$
$F_0(b) = \int_0^b e^{-3Ax} dx$
$F_1(b) = \int_0^b e^{-2Ax} dx$

FIGURE A8.C

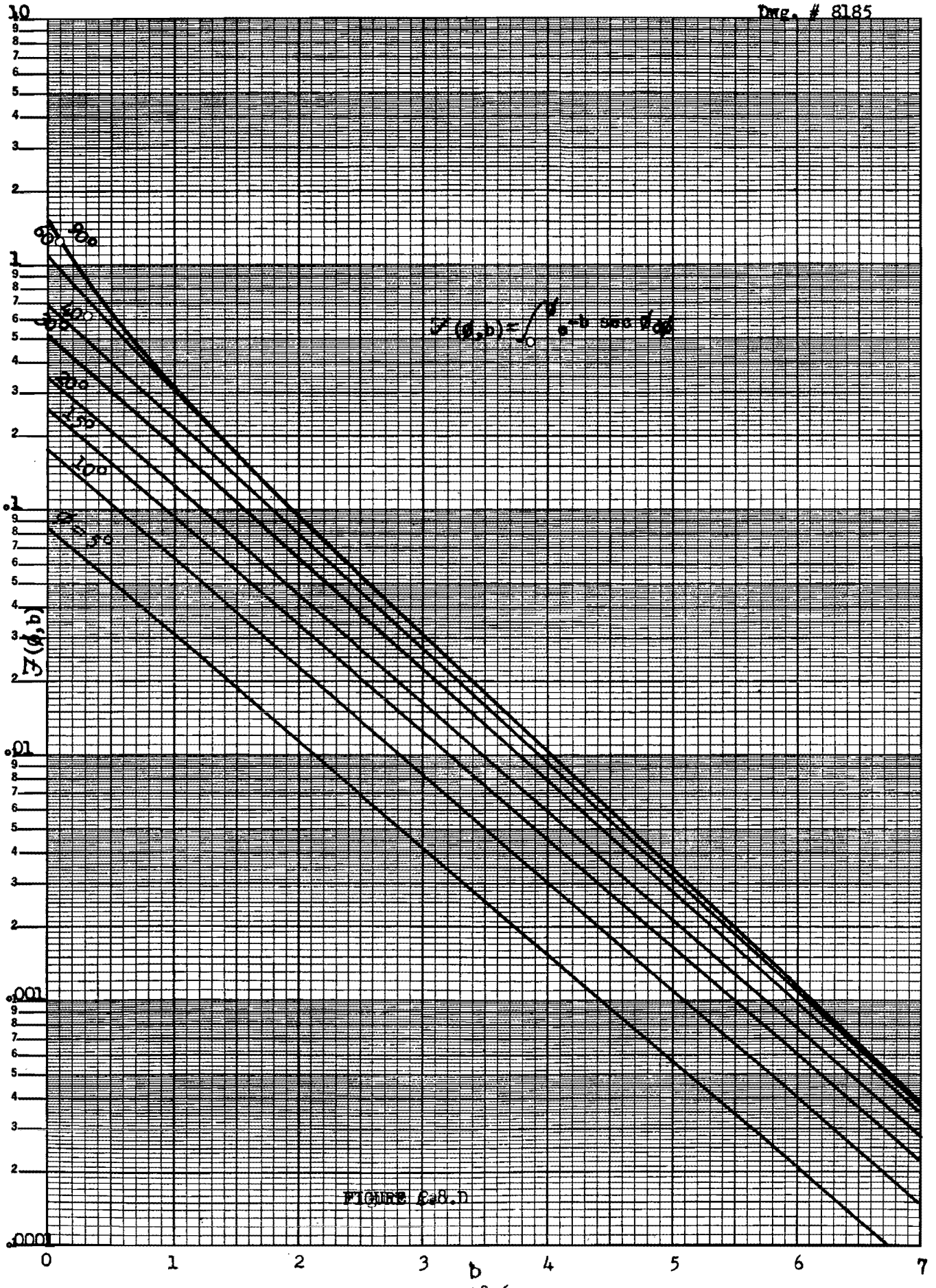

Dwg. # 8185
$\mathcal{F}(\phi,b) = \int_0^{\phi} e^{-b \sec \phi} d\phi$
$\mathcal{F}(\phi,b)$
b
0
1
2
3
4
5
6
7
10
1
.1
.01
.001
.0001

FIGURE A8.D

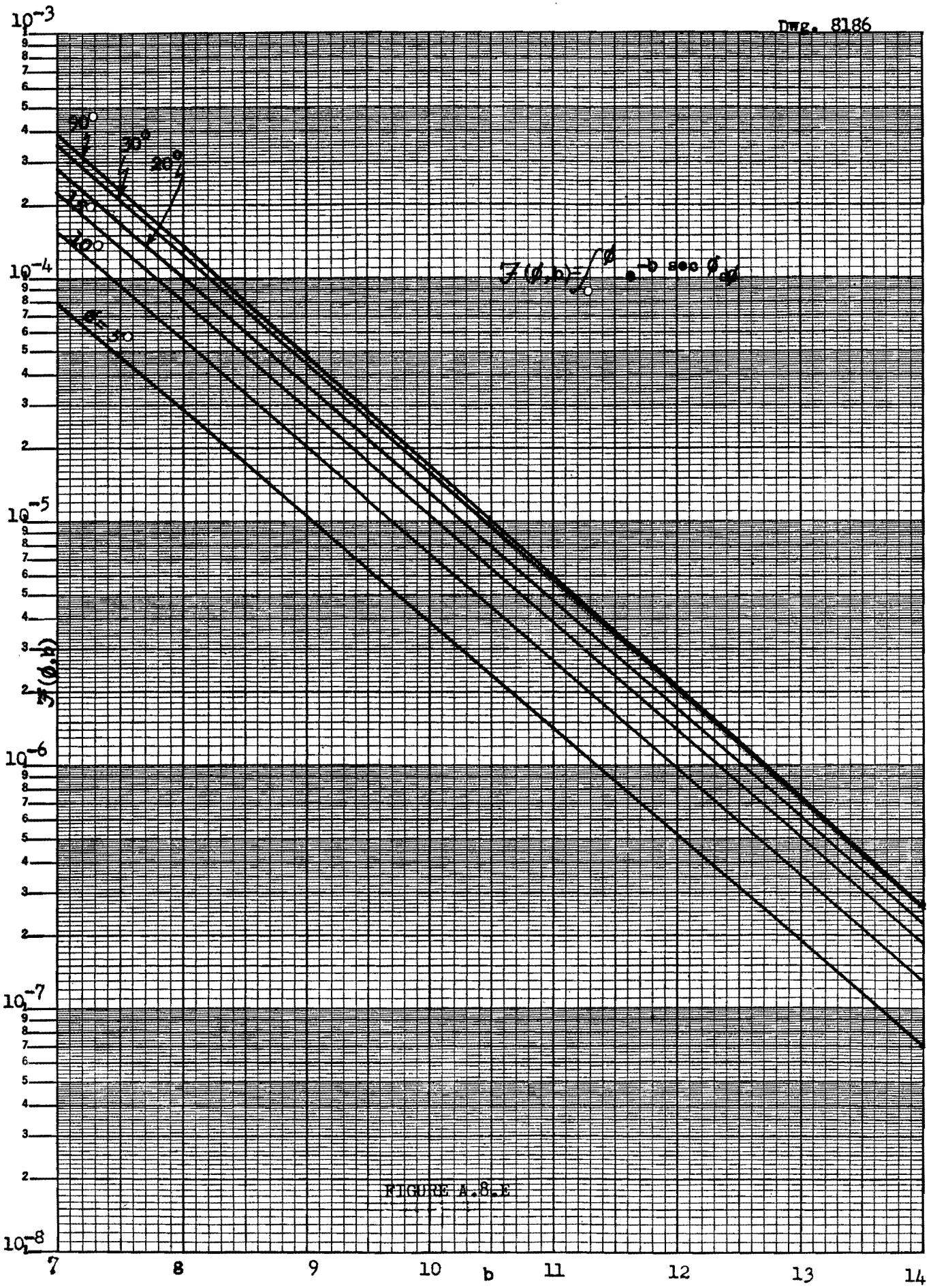

Dwg. 8186
$\mathcal{J}(\phi,b)=\int_0^{\phi} e^{-b\sec\phi}\,d\phi$
$\phi=50°$
$\phi=30°$
$\phi=20°$
$\phi=15°$
$\phi=10°$
$\phi=5°$
$\mathcal{J}(\phi,b)$
$10^{-3}$
$10^{-4}$
$10^{-5}$
$10^{-6}$
$10^{-7}$
$10^{-8}$
7
8
9
10
11
12
13
14
b

FIGURE A.8.E

Dwg. # 8187

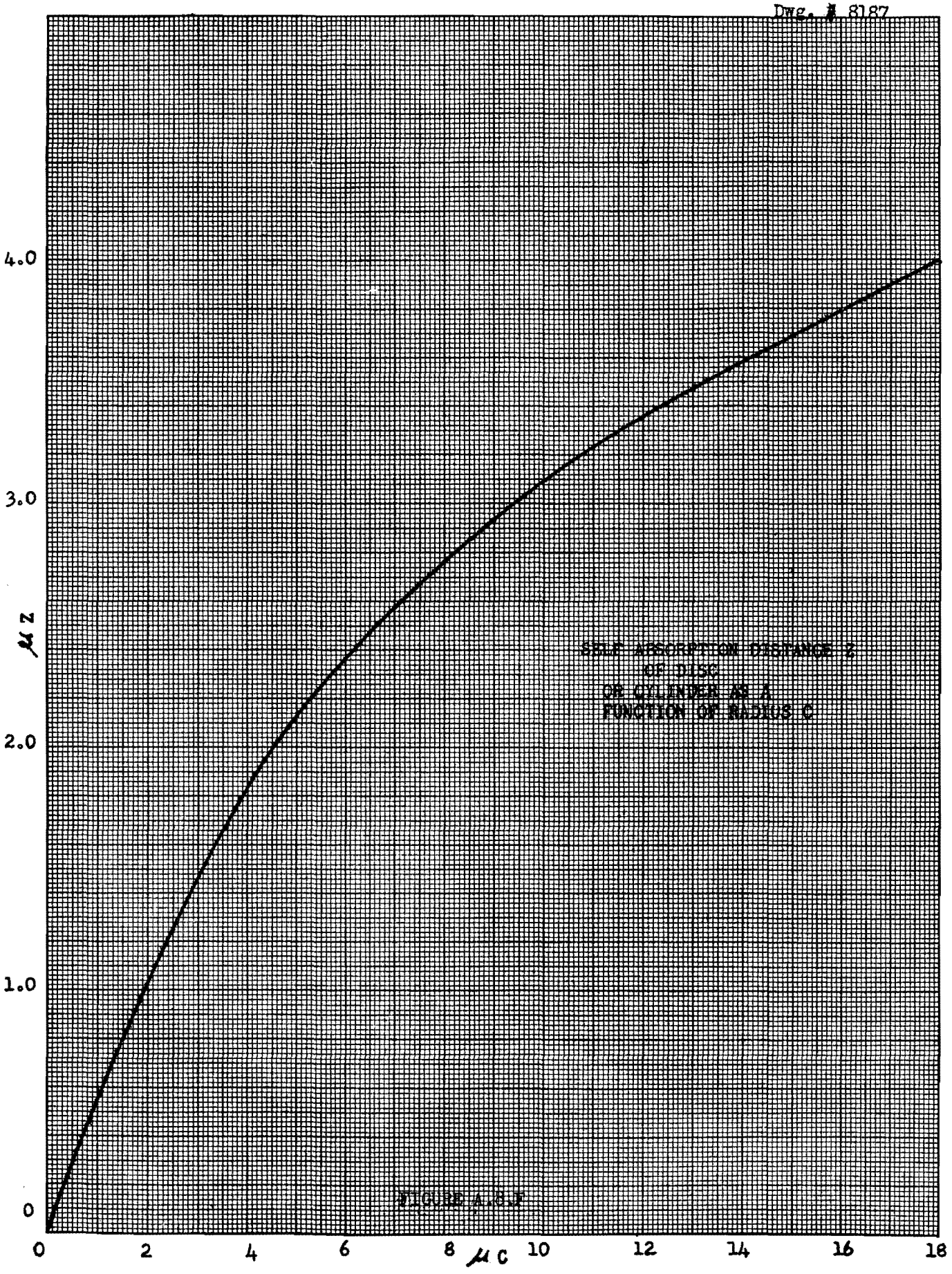



FIGURE A.8.F

## Appendix 9*

# PNEUMATIC RABBITS

### A9.1 GENERAL DESCRIPTION OF PROPOSED RE-ENTRY SHUTTLE SYSTEM

The layout of the re-entry shuttle system is shown in Figs. A9.A and B. The equipment provides for the introduction of the shuttles into the system in the basement and provides a means of propelling them to the inner end of the rabbit tube. After a predetermined irradiation time in the reactor, they are expelled either to the laboratories or to an unloading station in the basement. The major components of the system are a loading terminal, a shock absorber, a transfer unit, a laboratory selector, and unloading terminals. An electric circuit provides timing and operation of the solenoid-operated valves that control the flow of air and motion of the shuttle through the system.

The loading terminal provides the means for insertion of the loaded shuttle into the system. Two "S" bends in the section within the reactor structure reduce the streaming of radiation outward through the tube. The shuttle is stopped at the active lattice by a pneumatic shock absorber. After irradiation the shuttle is propelled out of the reactor to the transfer unit, from which it is sent either to the unloading terminal in the basement or to the laboratory selector. If sent to the laboratory selector, it is then directed to the unloading terminal in any one of the four laboratories provided with this facility.

### A9.2 PROPOSED EQUIPMENT AND FACILITIES FOR RE-ENTRY SHUTTLE SYSTEM

The following items are included in the design of the re-entry shuttle:

1. Rabbit tube HR-2.
2. Remotely operated solenoid valves to control direction of air flow in pressure and vacuum lines.
3. Electric air-controlled gate valve to permit passage of shuttle and to isolate the laboratory system from the reactor propelling pressure system.

*ANL contribution.

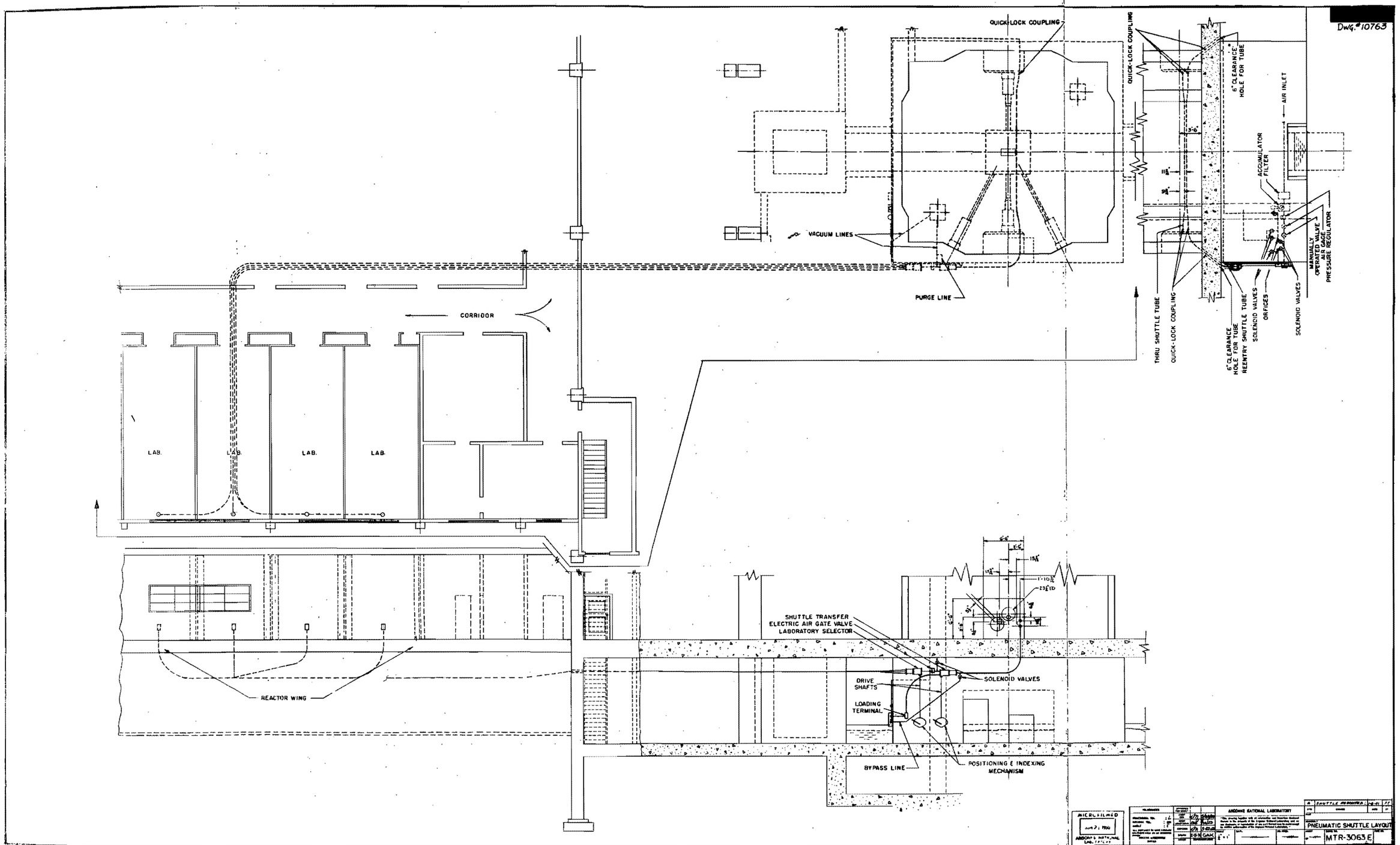
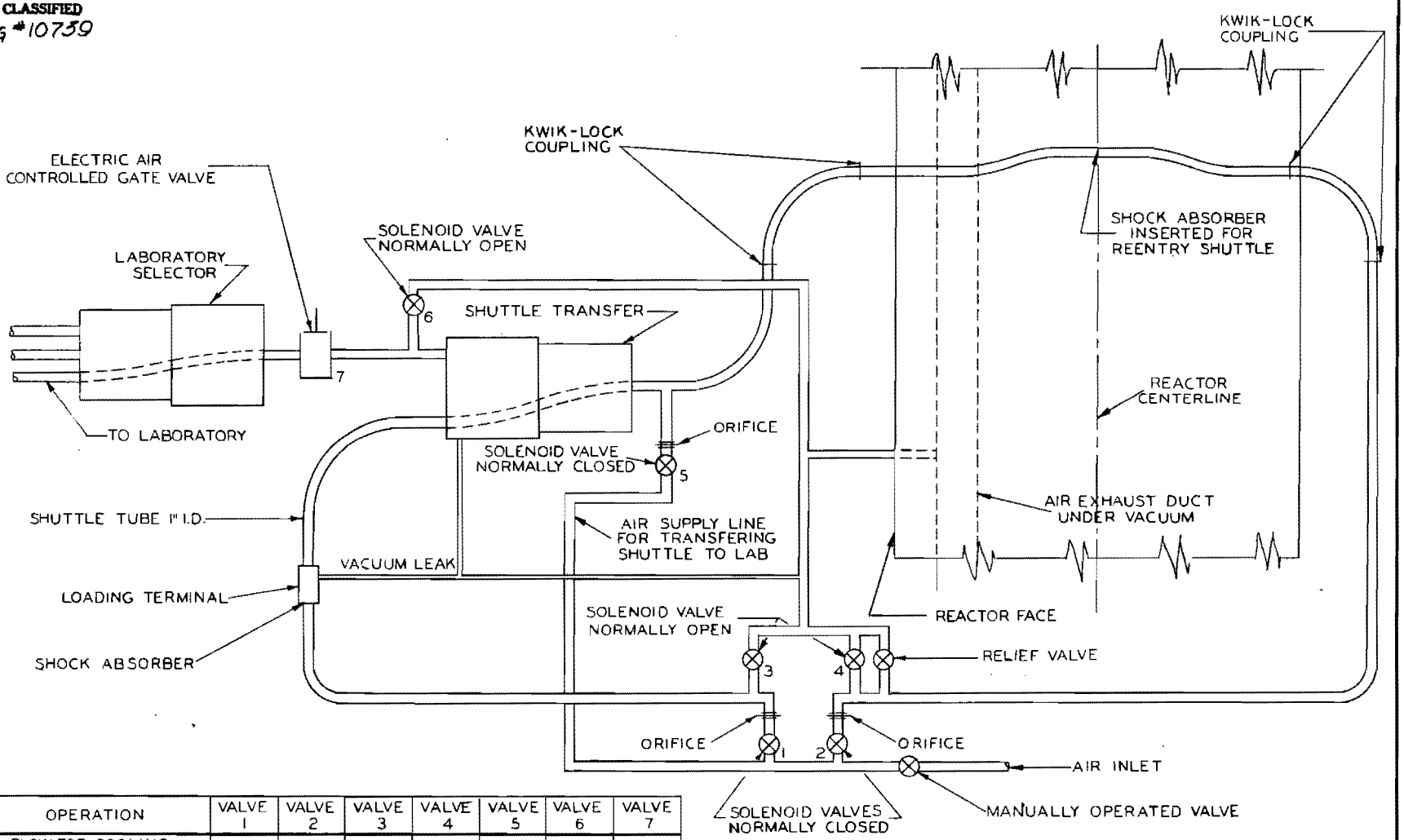

Dwg. #10763
QUICK-LOCK COUPLING
QUICK-LOCK COUPLING
6" CLEARANCE HOLE FOR TUBE
AIR INLET
ACCUMULATOR FILTER
MANUALLY OPERATED VALVE
AIR GAGE
PRESSURE REGULATOR
VACUUM LINES
PURGE LINE
THRU SHUTTLE TUBE
QUICK-LOCK COUPLING
6" CLEARANCE HOLE FOR TUBE
REENTRY SHUTTLE TUBE
SOLENOID VALVES
ORIFICES
SOLENOID VALVES
CORRIDOR
LAB.
LAB.
LAB.
LAB.
REACTOR WING
SHUTTLE TRANSFER
ELECTRIC AIR GATE VALVE
LABORATORY SELECTOR
SOLENOID VALVES
DRIVE SHAFTS
LOADING TERMINAL
BYPASS LINE
POSITIONING & INDEXING MECHANISM
ARGONNE NATIONAL LABORATORY
PNEUMATIC SHUTTLE LAYOUT
MTR-3063 E

FIG. A.9.A

NOT CLASSIFIED
Dwg #10759




| OPERATION | VALVE 1 | VALVE 2 | VALVE 3 | VALVE 4 | VALVE 5 | VALVE 6 | VALVE 7 |
|---|---|---|---|---|---|---|---|
| AIR FLOW FOR COOLING SHOCK ABSORBER | CLOSED | CLOSED | CLOSED | OPEN | OPEN | CLOSED | CLOSED |
| INSERTING SHUTTLE IN TERMINAL | CLOSED | CLOSED | OPEN | OPEN | CLOSED | OPEN | CLOSED |
| DRIVE SHUTTLE TO CENTER OF REACTOR | OPEN | CLOSED | CLOSED | CLOSED | CLOSED | CLOSED | CLOSED |
| COOL SHUTTLE | OPEN | CLOSED | CLOSED | OPEN | CLOSED | CLOSED | CLOSED |
| RETURN SHUTTLE TO TERM | CLOSED | OPEN | OPEN | CLOSED | CLOSED | CLOSED | CLOSED |
| SEND SHUTTLE TO LAB | CLOSED | OPEN | OPEN | CLOSED | CLOSED | OPEN | OPEN |
| AFTER PASSING TRANSFER STATION | CLOSED | CLOSED | OPEN | CLOSED | OPEN | CLOSED | OPEN |

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TOLERANCES: FRACTIONAL TOL ± 1/64; DECIMAL TOL ± .005; ANGLE ± 1/4°; ALL SURFACES TO HAVE AVERAGE (125 RMS) OR AS REQUIRED FINISH. UNLESS OTHERWISE NOTED

APPROVED FOR CONST.: CHIEF ENG. 2-7-50; CHIEF DRAFTSMAN 2/6/50; CHECKED 2-7-50; DRAWN 2-1-50 GAK; REPRESENTATIVE 2-7-50

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SYM. | CHANGE | DATE | BY

PART: SCHEMATIC LAYOUT

ASSEMBLY: PNEUMATIC SHUTTLE SYSTEM

SCALE: —//— MATL: —//— NO. REQD.: —//—

FIG. A.9.B

4. Loading terminal.

5. Shuttle transfer unit.

6. Laboratory selector.

7. Shock absorber to position re-entry shuttle.

8. Electric timing device.

9. Monitoring device to determine flux at center of shuttle tube.

The more important items in this list are discussed in the following paragraphs.

**A9.2.1 Rabbit Tube.** The proposed layout of the pneumatic shuttle system is shown in Fig. A9.A. The shuttle tube begins at an inlet air valve on the west wall of the reactor in the basement, continues around the south wall, passes up through the first floor of the Reactor Building, and enters the reactor face approximately 1 ft 10 in. east of the reactor north-south centerline at an elevation of approximaeely 98 ft 10 in. In the concrete shield it makes an offset bend and passes through the graphite and reactor tank approximately 1 ft 3 in. east of the reactor north-south centerline at an elevation of approximately 99 ft. Beyond the graphite, the tube again makes an offset bend to emerge from the north wall of the reactor in a location similar to its entry point in the south wall. From here it bends down through the floor and continues along the basement ceiling to the other inlet valve at the west wall. In the recesses in the north and south reactor faces at the first floor level, the curved sections of tubing are equipped with quick-lock couplings to facilitate their removal. The quick-lock couplings provide a ready means for disassembling the shuttle tube when it is desired to insert the shock absorber into the reactor or withdraw it into a coffin. The shuttle tube through the reactor tank and the thermal shield is constructed of aluminum. Beyond the reactor structure the section that is utilized by the shuttle is stainless steel, but the other sections are constructed of copper tubing.

**A9.2.2 Electric Air-controlled Gate Valve.** An electrically operated air-controlled gate valve, located between the shuttle transfer unit and the laboratory selector, separates the system into two sections and stops the flow of air into the laboratory while the shuttle is propelled into the reactor, irradiated, and expelled.

The laboratory section consists of the laboratory selector and the four tubes that terminate in separate laboratories. At each unloading terminal a

line to the contaminated-air system withdraws any air that is discharged from the shuttle tube during the delivery or unloading of a shuttle. This is a precaution to prevent escape of contaminated air into the laboratory building.

**A9.2.3 Shuttle Transfer.** The shuttle transfer design is illustrated in Fig. A9.C. The transfer unit consists of a housing that encloses the transfer tube, indexing gear, and guide disk. Four tie rods unite these parts into a single unit. Both the section of the tube connected to the laboratory selector and the section of tube leading from the loading station fasten to the gear end of the housing. At the other end a single tube leads to the reactor.

The shuttle transfer unit and the indexing mechanism, connected by means of an arrangement of bevel gears and shafting, provide for the indexing of the shuttle tube. One-third of a revolution of the indexing handle rotates the transfer unit through an angle of 180°.

**A9.2.4 Laboratory Selector.** No drawings have been prepared for the laboratory selector unit as its construction will be very similar to the shuttle transfer unit. It differs only in the arrangement and number of tubes required for servicing the laboratories. The selector can be located close to the transfer unit or it can be set up at some convenient location in the laboratory building.

**A9.2.5 Shock Absorber.** Experiments which require that the shuttle be delivered in a very short time, ½ sec or less, need a special shock absorber to stop the shuttle. A suitable design, illustrated in Fig. A9.D, has been engineered from the standpoint of high speed, adequate cooling, and ease of insertion and withdrawal. It consists of a series of alternate sections of bellows and straight tubes, approximately 12 ft long and fastened to the shock absorber body, to provide the means for insertion and location of the shock absorber in the shuttle tube. Three openings formed between the shock absorber and the inside diameter of the shuttle tube provide the space for the passage of cooling air when the shuttle rests against the shock absorber. They also form the small orifices that restrict the flow of air when the large orifice is closed during the slowing down of the shuttle. The large orifice is formed by the seat and the beveled surface of the valve. When the valve is open the larger portion of the air flows through the valve opening and then out the portholes to mix with the air flowing through the small orifices. The air continues to flow in the annulus formed between the body diameter and the

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Dwg. # 10762

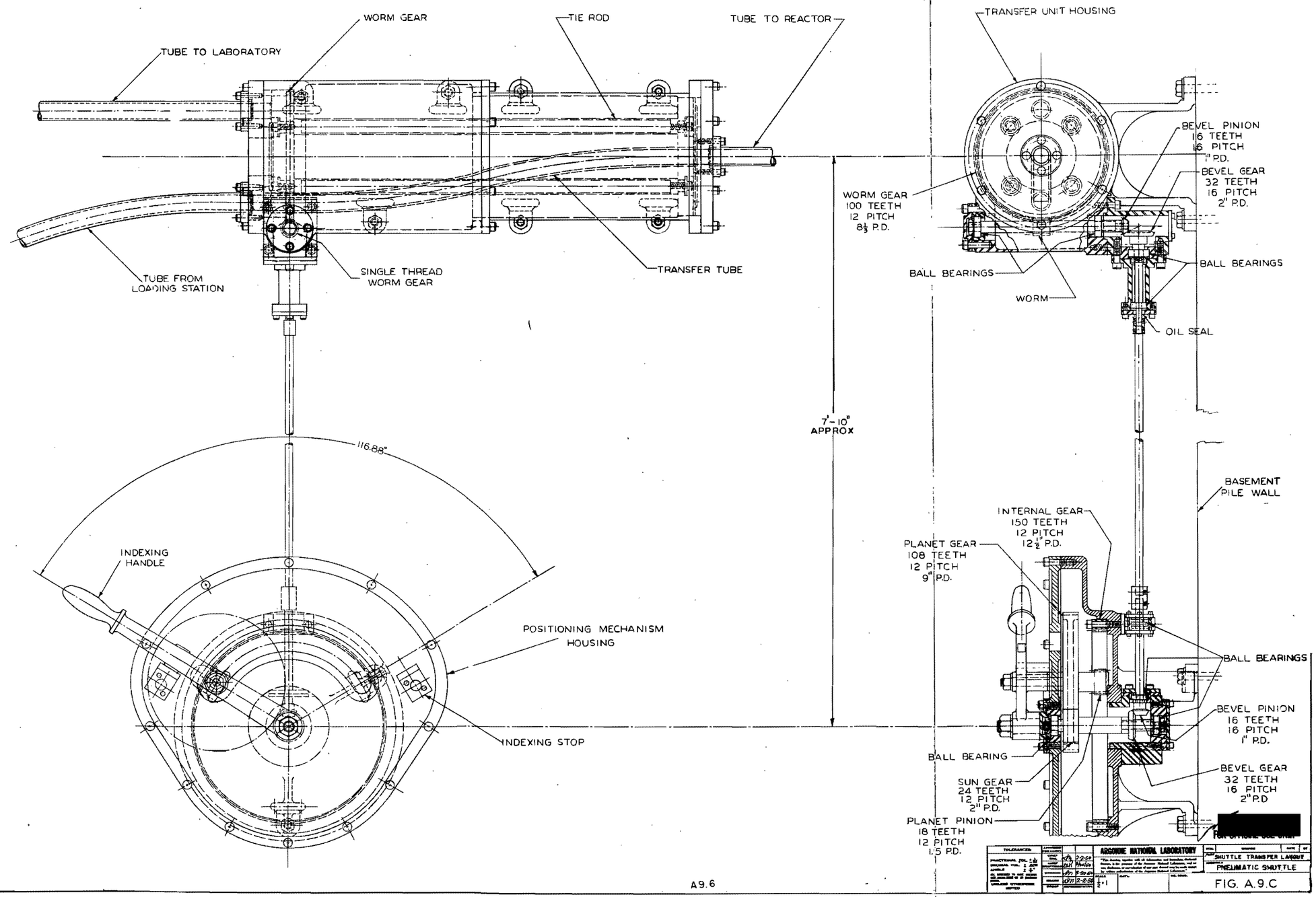


FIG. A.9.C

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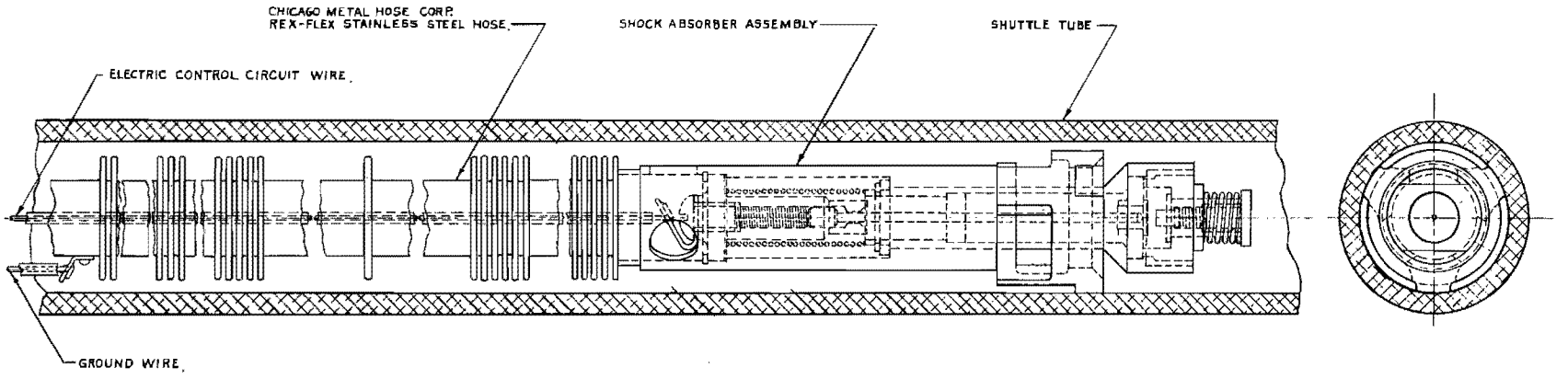


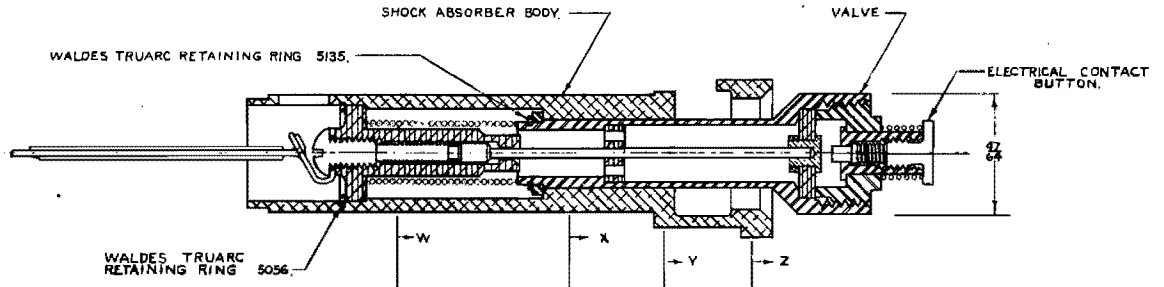


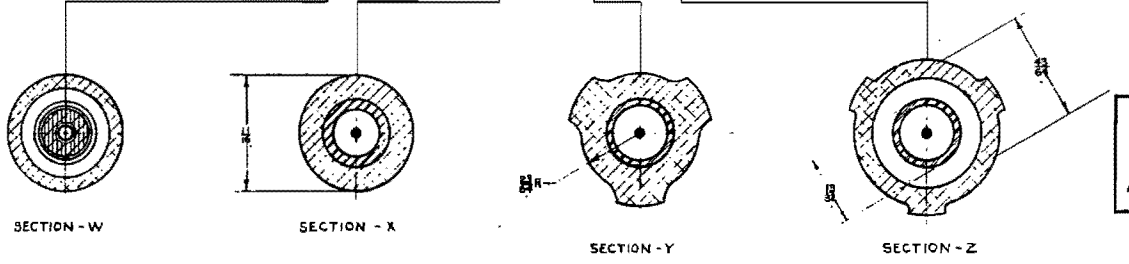


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| TOLERANCES: | APPROVED FOR CONST. | | ARGONNE NATIONAL LABORATORY | SYM. | CHANGE | DATE | BY |
|---|---|---|---|---|---|---|---|
| FRACTIONAL TOL. ± 1/64 | CHIEF ENG. | 3-2-50 | "This drawing, together with all information and know-how disclosed thereon, is the property of the Argonne National Laboratory, and no use, disclosure, or reproduction of any part thereof may be made except by written authorization of the Argonne National Laboratory." | PART | SINGLE CONTACT SHOCK ABSORBER. | | |
| DECIMAL TOL. ± .005 | CHIEF DRAFTSMAN | 3-2-50 | | ASSEMBLY | PNEUMATIC SHUTTLES. | | |
| ANGLE ± 1/4° | CHECKED | 3-2-50 | | | | | |
| ALL SURFACES TO HAVE AVERAGE 125 RMS FINISH OR AS SPECIFIED FORM. UNLESS OTHERWISE NOTED | DRAWN | 3-2-50 | SCALE 2:1 | MATL. | NO. REQD. | FIG. A.9.D | |
| | GROUP | REPRESENTATIVE | | | | | |

inside surface of the shuttle tube. Three holes located in the body in front of the bellows permit the air to divide into two streams, one flowing inside and the other outside the bellows. A spring enclosed in a sealed chamber in the shock absorber body holds the valve open until the pressure of the air in front of the shuttle overcomes the spring force. After the valve closes a greater pressure is required to force the air displaced by the shuttle through the small orifices. This increase in pressure acts as a retarding force to slow down and stop the shuttle. When the shuttle depresses a spring-backed button secured in the valve head, an electrical circuit for controlling the shuttle timing device is completed. The wires for the timing device enter through connections at the removable section of the shuttle tube. One wire is grounded to the flexible tubing and the other continues through the bellows and terminates at the electrical terminal mounted in an insulation bobbin that is pressed into the shock absorber body. The electrical circuit is completed by means of a spring, a plunger, and a rod that is secured in an insulation bushing mounted in the valve.

The parts that comprise the shock absorber should be made as light as possible in order to minimize the heat generated by gamma absorption. A continuous flow of air at the rate of 0.04 lb/sec is required to cool the shuttle and shock absorber.

**A9.2.6 Delivery Stations and Delivery-station Shock Absorbers.** Speed of delivery of the shuttle to the laboratory may be of prime importance; hence the delivery velocities may be high and adequate shock absorbing means must be provided at the delivery stations. For this purpose conventional pneumatic dash-pot delivery stations can be used, since heating due to nuclear reactions is not involved. In general these consist merely of a 3-ft blind length of tubing beyond the exit for the propelling air. Suitable doors are provided to remove the shuttles after delivery. The doors should not be opened until the delivery air has been shut off (by automatic timing of the delivery cycle), thereby placing the delivery stations under a slight vacuum.

**A9.2.7 Pneumatic Driving System.** The incoming air from the plant air compressor is cleaned, filtered, and dehumidified before it enters the accumulator tank. A relief valve controls the maximum pressure in the tank. A pressure regulator controls the pressure of the air prior to its entry into the pneumatic shuttle system through the solenoid-operated valves that govern

the direction of air flow. A secondary solenoid valve controls the flow of air after the shuttle transfer has been rotated and also when it is desired to transfer a shuttle to a laboratory. Solenoid valves at the extremities of the shuttle tube control the direction of the air after it has passed through the reactor and prior to its entering the reactor cooling-air exhaust duct. These two valves remain open except during a charging operation. A relief valve governs the maximum pressure allowed in the exhaust line during charging of a shuttle. A solenoid valve permits the escape of gas when the shuttle transfer tube has been rotated to the laboratory position.

**A9.2.8. Vacuum System.** Both ends of the shuttle tube are connected, through solenoid-operated valves, to a single pipe which terminates at the reactor cooling-air exhaust duct. Since the exhaust duct is under vacuum and the valves are normally open, one or both ends of the shuttle tube are always under a vacuum. This tends to clear the shuttle tube of dust particles and simplifies the periodic purge required to clear the lines. Vacuum leak lines, continually open to the exhaust line that leads to the reactor cooling-air exhaust duct, place the loading terminal and the shuttle transfer unit under a continuous negative pressure and provide against leakage of contaminated air into the Reactor Building.

Vacuum exhaust connections for the through shuttles are provided at the face of the reactor for the convenience of experimenters in setting up their equipment. These are part of the experimental services normally provided at all beam hole facilities.

## A9.3 OPERATION OF RE-ENTRY SHUTTLE SYSTEM

Prior to delivery of a shuttle to the reactor, the pneumatic shuttle system is under a slight vacuum since the solenoid valves at the ends of the shuttle tube are open to the reactor cooling-air exhaust system. The sequence of operations that take place during the delivery of a shuttle can be better understood by reference to Fig. A9.B.

Shuttles prepared for delivery to the reactor are introduced into the system at the loading terminal. After the cover is secured, the operator presses a switch which automatically closes all valves in the vacuum lines and

opens the solenoid valve in the pressure line to admit the air that propels the shuttle. The incoming air, metered through an orifice, increases the pressure in the section of pipe between the shuttle and the orifice, and drives the shuttle against the shock absorber at the center of the reactor. The air in front of the shuttle is compressed since it is not permitted to escape. The pressure continues to build up and eventually overcomes the spring force that normally holds the shock absorber valve open. With closure of the valve the flow area at the body section is reduced to that of the three small openings formed between the shock absorber body and the inside of the shuttle tube. This decrease in orifice area increases the pressure required to discharge the air displaced by the shuttle. The volume of air displaced is a function of the pressure and the velocity of the shuttle. In an effort to reach a state of equilibrium, the pressure acts as a retarding force on the shuttle. The maximum pressure build-up in the low-pressure section of the line is limited by a relief valve. When the shuttle contacts the shock absorber, an electrical circuit is completed which simultaneously operates an electric timing device and opens the solenoid valve that permits the gas to escape into the reactor cooling-air exhaust system. The steady flow of air through the tube cools the shock absorber and shuttle.

If the shuttle is to be sent to the laboratory, the shuttle transfer tube is rotated to the laboratory position during the time interval in which the shuttle is irradiated. Rotation of the shuttle transfer tube automatically closes the solenoid valve that has been supplying the air to propel and cool the shuttle and, after a fraction of a second delay, opens the solenoid valve in the air-supply line that by-passes the shuttle transfer unit. This now provides the air necessary to cool the shuttle and the shock absorber.

At the end of a preset exposure time for the shuttle, the electric circuit in the timing device is broken. This automatically reverses the solenoid valves that control the direction of air flow, thereby expelling the shuttle. At the same time, the solenoid valve on the vacuum side of the transfer unit opens to permit the air in front of the shuttle to discharge into the reactor cooling-air exhaust system. After a delay period, to ensure that the shuttle is beyond the transfer unit, the valve in the vacuum line closes and the shuttle transfer by-pass air-supply valve opens to drive the shuttle to the unloading terminal in the laboratory. After a suitable time delay to ensure delivery of the shuttle and to purge the system, the valves in the vacuum line open automatically. The unloading terminal cover can then be opened and the shuttle removed. The air-supply valve can still be open to permit a flow of air to cool the shock absorber.

**A9.3.1 Pressure Drop.** When the shuttle is positioned at the center of the reactor by the shock absorber, an approximate pressure drop of 38 lb/sq in. is required to force 0.04 lb of air per second through the shuttle tube. This is equivalent to 31.8 cfm of air at 70°F and atmospheric pressure of 14.7 lb/sq in.

**A9.3.2 Anticipated Operating Curves.** The curves shown in Fig. A9.E are not to be construed as the actual representation of the air pressures in the tube and the velocities of the shuttle. They merely indicate a trend and are helpful in understanding some of the problems in designing the mechanism for propelling and slowing down the shuttle in the pneumatic system.

The propelling force starts at the pressure produced by the vacuum in the reactor cooling-air system. A constant supply of air into the chamber increases the pressure behind the shuttle and drives it to the center of the reactor. The maximum pressure is reached when the shuttle velocity and the speed of the incoming air in the tube are equal. Maximum acceleration also occurs at this point. A decrease in the propelling pressure occurs when the shuttle travels faster than the air velocity in the tube. This is due to the volume behind the shuttle increasing at a greater rate than the build-up in pressure.

During the driving of the shuttle, no air is permitted to escape through the exhaust valves. The travel of the shuttle compresses the volume in front of the shuttle and thereby increases the back-pressure. The difference between the propelling pressure and the back-pressure is the unbalanced pressure that decreases the acceleration. This continues until the two pressures are equalized, at which time the acceleration becomes zero and the velocity becomes a maximum. Further increase in back-pressure results in deceleration of the shuttle. When the pressure in front of the shuttle overcomes a spring force in the shock absorber, the valve closes.

After the valve closes, a greater pressure is required to force the air, displaced by the shuttle, through the small orifices between the shock absorber body and shuttle tube. This accounts for the sharp rise in the back-pressure curve, the increase in deceleration, and the decrease in shuttle velocity. It should be noted that the velocity is practically zero when the shuttle is approximately 8 in. from its stop. The shuttle continues to stall until the propelling and back-pressure are again equal, at which time the shuttle is

NOT CLASSIFIED
DWG #10738

PNEUMATIC SHUTTLE SYSTEM
SPACE VS. TIME
PRESSURE VS. TIME

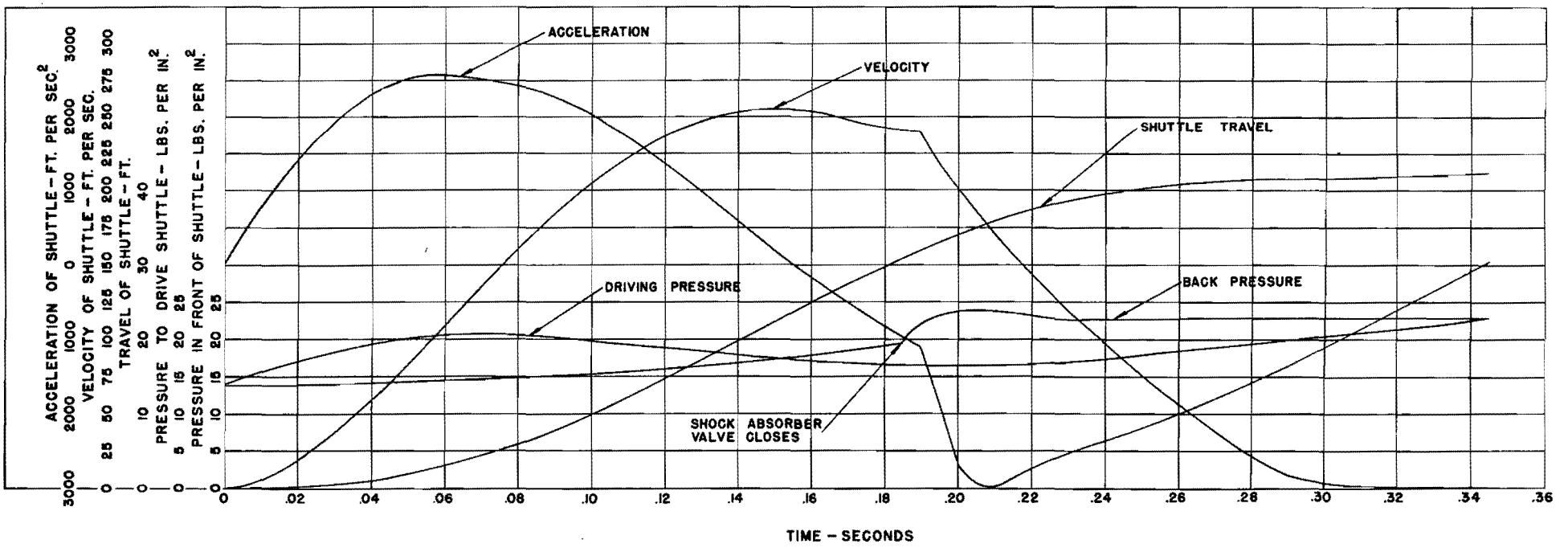


FIG. A.9 E
ANTICIPATED OPERATING CURVES

MTR-G-144

forced against the shock absorber valve head. The pressure in back of the shuttle then increases until the steady flow required to cool the shuttle and shock absorber is attained.

**A9.3.3 Shuttle Material.** Some of the characteristics of an ideal shuttle material are (1) low-activation cross-section, (2) low density, (3) high thermal conductivity, (4) stability to radiation, and (5) structural strength.

The first property is desirable to prevent the carrier's becoming active enough to require shielding during the unloading operation. Materials that satisfy this requirement are electrolytic iron, beryllium, zirconium, carbon, polystyrene, and bakelite.

Of the preceding materials, iron is attractive with respect to thermal conductivity, stability to radiation, and structural strength, but, because of its density, it has a high rate of gamma heat generation that introduces cooling difficulties. However, the cooling problem can be minimized by limiting the weight of the carrier to approximately 12 g. For a 3-in.-long shuttle, the maximum wall thickness would then be 0.01 in. An additional restriction on the use of iron is that in order to limit the induced activity to tolerable values it must not contain appreciable quantities of manganese.

The use of beryllium or zirconium would permit a larger wall thickness for a given weight carrier. The activity of these elements is controlled by the impurities aluminum and hafnium.

Some of the plastics are promising, but the inherent low thermal conductivity makes cooling of the sample difficult. Information on the effects of radiation on plastics is rather incomplete at this time. However, from experiments made with low neutron fluxes, it appears feasible to use bakelite and polystyrene as shuttle materials.

**A9.3.4 Cooling of the Shuttle.** The cooling required by the loaded carrier is a function of the total mass and the desired maximum temperature. Since it may become necessary to irradiate samples that have a low decomposition temperature, such as hydrocarbons, the cooling system should be capable of limiting the maximum sample temperature to approximately 300°F. In addition, the use of a plastic carrier would limit the temperature to a similar maximum. It is estimated that the upper limit on the mass of a loaded shuttle will be 25 g. To cool this shuttle and the shock absorber with air, a flow of 0.04 lb/sec is required. This value is based on a heat transfer area of 9.6 sq in. with a 3-in.-long carrier.

**A9.3.5 Shielding.** The exposure hazards in the shuttle system will be from activation of the propellant gas or the sample and carrier. Computations indicate that no permanent shielding will be required to satisfy health physics requirements, and temporary shielding will be adequate when needed to provide instrument background.

The air used as a propellant will contain active argon, nitrogen, and oxygen. To prevent the escape of active air it is necessary for the parts of the system that operate at a positive pressure to be leakproof.

**A9.3.6 Use of Air vs. $CO_2$ for Propellant.** Air is used as a medium to drive the pneumatic shuttle in and out of the reactor. It was selected because it is economical to use as a cooling medium for the sample and shock absorber. A comparison of $CO_2$ and air made on the basis of cost and radiation hazard indicates that air is preferable. The inherent disadvantage of air is the formation of 110-min argon. However, the exhaust lines in the proposed shuttle system will operate at a negative pressure so that the escape of active argon is not likely.

The exposure at the surface of a 1-in. shuttle tube containing active exhaust air is calculated to be 0.02 r per 8 hr. At short decay periods no decrease in exposure can be realized through the use of $CO_2$ because the significant fast flux at the shuttle tube gives appreciable $O^{16}(n,p)N^{16}$ activity.

## A9.4 REFERENCE REPORTS


(1) Nicoll, D. and Lennox, D. H., *Pneumatic Shuttle*, Design Report No. 44, ANL-ABS-16, March 27, 1950.

(2) Scott, D., *Pneumatic Rabbits in the X-Pile*, ORNL CF-47-11-35, October 31, 1947.

(3) Scott, D., *Status of the Pneumatic Rabbit Design in the High Flux Pile*, ORNL CF-48-6-134, June 9, 1948.


## A9.5 REFERENCE DRAWINGS

| DWG. NO. | TITLE |
|---|---|
| BKC-3150-538-1 through 10 | HR-1 and HR-2 - Rabbit Tubes |
| BKC-3150-540-1 through 5 | HR-3 and HR-4 - Rabbit Tubes |

## Appendix 10

## HYDRAULIC RABBITS

The hydraulic rabbit system for the MTR consists of two 1-in.-I.D. vertical tubes through the east end of the beryllium reflector (see Beryllium Layout, Figs. 2.4.A and B) and a 1.310-in.-I.D. tube through each of the regulating rod holes not in use.

The following description of the system is a tentative proposal for the design. The final design of the system may vary somewhat from this presentation, but the overall principles will be the same.

The proposed system is shown in Fig. A10.A. Shuttles from the laboratory, after being sealed, are lowered into the loading chute and pushed through valve C into the loading chamber. Valve C is then rotated into the insertion position and valve B is opened, permitting the rabbit shuttle to be driven to the top of the reflector where an orifice acts as a stop. (The shuttle can be stopped lower than this by insertion of appropriate dummies ahead of it.) When the irradiation is finished, the demineralized-water flow is decreased until the shuttle falls into the canal. This occurs when the demineralized-water flow into the reactor is so low that resistance of the shuttle to the flow is not sufficient to overcome gravity. With valve A closed, it is possible to discharge the shuttle into a bucket type coffin in the canal without letting any great volume of active water get into the canal. Since valve A is normally an open solenoid valve, it will not trap the shuttle in case of power failure but will allow it to fall into the canal.

Vertical holes in the upper support casting, in line with the rabbit tubes, permit the use of a long rod for pushing any shuttle which might stick above the canal.

The actual design of the shuttle is, of course, best done by the MTR operator or the experimenter. If a shuttle design similar to that of the hydraulic rabbit system of the X-10 reactor is used, the tubes should have a minimum radius of curvature of 48 in. The rabbit tube positions at the east end of the reflector will accommodate 1.190-in.-O.D. tubes, and the regulating rod holes will accommodate 1.500-in.-O.D. tubes. The inside diameters of these tubes have been set at 1.000 in. and 1.310 in., respectively.

DWG. # 10712 [redacted]

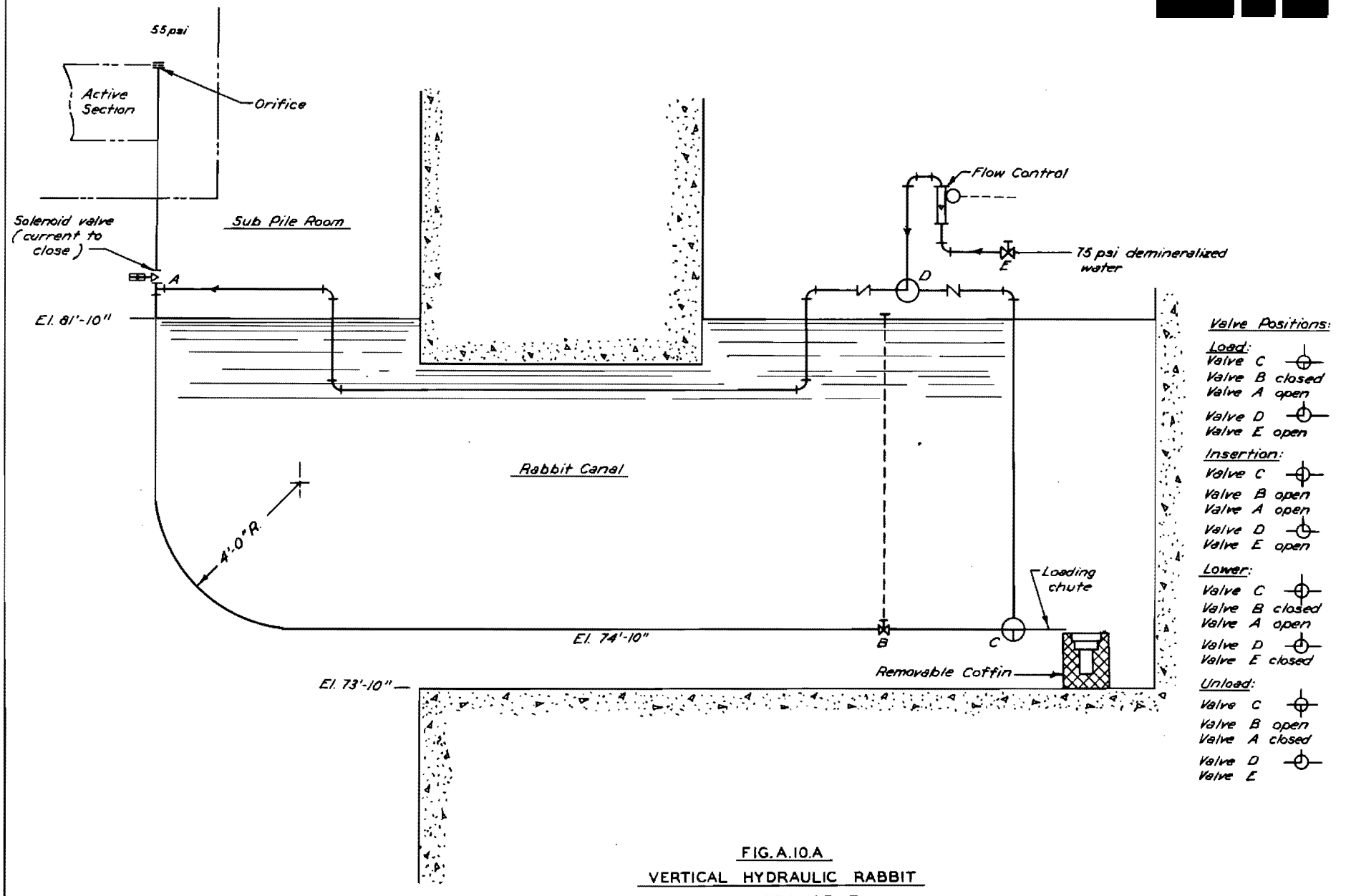


FIG. A.10.A
VERTICAL HYDRAULIC RABBIT
PIPING ARRANGEMENT

Demineralized water at 75 psi will be available at the rabbit canal, and approximately 5 gal of water per minute past the shuttle will be necessary for cooling. No shock absorbers are provided, but speed of insertion can be easily regulated by the demineralized-water flow.