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SUPPLEMENTARY REPORT TO REACTOR SAFEGUARD COMMITTEE

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LIST OF APPENDICES

- A. "Report of the Atomic Energy Commission Reactor Safeguard Committee on the Proposed High Neutron Flux, Water Cooled Thermal Reactor" by Edward Teller dated February 10, 1948, less its Addendum I.
- B. "Cooling after Shutdown" by J. R. Huffman, dated June 7, 1948, ORNL CF-48-6-155.
- C. "Information for Reactor Safeguard Committee Meeting Sept. 9, 1948" by John R. Huffman and S. Untermyer dated August 26, 1948, ORNL CF-48-8-348.
- D. "Materials Testing Reactor Design Data Sheets" by D. J. Mallon dated Oct. 28, 1949, ORNL CF-49-10-232.
- E. "Reduction in Reactivity of Heterogeneous Reactor by Expulsion of Coolant," by S. Untermyer, ANL-WPB-60.
- F. "Reactor Hazards" by J. J. Katz and H. R. Hoekstra dated January 27, 1950, ANL-WMM-596.
- G. "Meteorology of the Reactor Test Site, Arco, Idaho" prepared by U.S. Weather Bureau, Sept. 16, 1949, less the detailed tabular data.
- H. "Argon Activity in Reactor Cooling Air," by J. A. Lane dated Nov. 17, 1949, ORNL CF-49-11-180.
- I. "Stack Height Calculations."
- J. Excerpt from "Pacific Coast Uniform Building Code" 1949.
- K. "Earthquake Information of Possible Application at Arco, Idaho" by J. Stewart Williams.

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I. INTRODUCTION

The purpose of this report is to present a review of the hazards inherent in the design of the Materials Testing Reactor, to the Atomic Energy Commission for its consideration and suggestions, if any, for improvement of the safety features of the reactor. The report emphasizes the hazards involved in the operation of the reactor and the safety features incorporated into the design to nullify or reduce the danger from these hazards. The report is issued by the Steering Committee with the approval of the Idaho Operations Office since the Steering Committee is the group most familiar with the reactor design.

Construction of the reactor at the Arco Reactor Testing Station by the Fluor Corporation was started in May, 1950 with completion scheduled for the Fall of 1951. At present the design of the reactor is being completed jointly by Oak Ridge National Laboratory and Argonne National Laboratory. The detailed design drawings are being prepared by the Chemical Plants Division of the Blaw-Knox Construction Company of Pittsburgh, Pa. The reactor project includes the necessary facilities for operation of the reactor at a remote site but does not include the Chemical and Metallurgical Plants necessary to keep the reactor in operation. These plants will be designed by another group of engineers and located at another site at the Reactor Testing Station. Therefore, no information concerning these plants is included in this report.

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The Materials Testing Reactor is essentially the same as the High Flux Reactor which was partially designed at the Oak Ridge National Laboratory during 1946 and 1947. The Reactor Safeguard Committee considered the High Flux Pile in relation to the possibility of its location at the DuPage site of the Argonne National Laboratory at a meeting at Oak Ridge National Laboratory on February 8 and 9, 1948, reference "Report of the Atomic Energy Commission Reactor Safeguard Committee on the Proposed High Neutron Flux, Water Cooled Thermal Reactor," by Edward Teller, dated February 10, 1948. A copy of this report without its Addendum I is attached to this report as Appendix A.

At the Oak Ridge meeting, the Safeguard Committee raised the objection to the High Flux Pile operated at 30,000 kilowatts, that the fuel assemblies might melt after shutdown and possibly release as high as 9,000,000 curies of gaseous fission products. It was indicated that every effort should be made to prevent release of these gaseous fission products. The most probable source of difficulty was considered to be the loss of water after shutdown with subsequent melting of the fuel assemblies by the heat liberated by radioactive decay of the fission products. It was considered possible that accidents or sabotage might occur which would cut off all three of the independent sources of cooling water.

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A report was prepared by Dr. J. R. Huffman on the heat evolved after shutdown. This is attached as Appendix B. A second report based on theoretical and experimental work was prepared by Dr. J. R. Huffman and Mr. S. Untermeyer and presented to the Committee for consideration at its meeting on September 9, 1948 in Washington. Since the Committee had already decided that a reactor to be operated at 30,000 kilowatts at the DuPage site would be unsafe, it did not act on the recommendations made by Dr. Huffman and Mr. Untermeyer. A copy of their report is attached as Appendix C.

During the latter part of 1948, the Atomic Energy Commission decided to open a Reactor Testing Station at a remote site. Subsequently, the site near Arco, Idaho was chosen and the decision has been made by the Commission to build the Materials Testing Reactor at that site. As noted above, construction is underway.

The actual site chosen for construction is located in Section 14 of township 3 north, range 30 east in Butte County, Idaho, about four miles northwest of the Naval Proving Ground firing range. This is about 45 miles west of Idaho Falls, and is in the northwest part of the Snake River lava plains.

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The important changes which have been made in the High Flux Pile design, as previously reviewed by the Committee, to convert the design to the present Materials Testing Reactor design are:

- A. Inclusion of additional safety features to provide greater certainty of water coverage of the fuel assemblies.
- B. Introduction of a 4 7/8" x 4 7/8" square experimental through hole next to the lattice and the replacement of one of the shim-safety rods by a vertical through hole.
- C. Introduction of eighteen vertical 2 1/2" diameter holes to be placed in a ring just outside of the reactor tank.
- D. Elimination of the beryllium oxide balls and replacement by graphite balls.

In general, the remainder of this report follows the general outline presented in Appendix H of the "Summary Report of the Reactor Safeguard Committee" issued in 1949 with the addition of Conclusions and Recommendations, Section II and III. Section IV of this report includes a general description of the reactor and its control system.

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II. CONCLUSIONS

It is concluded that:

- A. The Materials Testing Reactor is operable with extremely minute probability of danger to the surrounding population.
- B. The control system can control an instantaneous change in k of at least 1%.
- C. Release of the gaseous fission products by melting of the fuel plates due to failure of the water system to keep the lattice covered by water is extremely unlikely.
- D. Further tests of the safety of the reactor should be made to determine experimentally the very fast changes in k that can be produced by malfunction, sabotage, or accident, and the effect of large changes in k on the reactor.

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III. RECOMMENDATIONS

It is recommended that:

- A. The Materials Testing Reactor be built as designed at the Idaho Reactor Testing site.
- B. The mock-up be used in experiments to check the changes in k that can be produced and the effects of large changes in k on the reactor.

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IV. DESCRIPTION OF THE REACTORA. Introduction

General descriptions of the physics and engineering design features of the High Flux Pile were given in the reports "Physics of the High Flux Pile - II" by E. Greuling, H. Soodak, and A. M. Weinberg dated March 27, 1947, MonP-272, and "Feasibility Report on Clinton High Flux Pile" by J. R. Huffman, M. C. Leverett, H. W. Newson, and A. M. Weinberg dated December 1, 1947, MonT-433. Reference is made to these reports for a general discussion of the design of the reactor and its facilities as planned for construction at Oak Ridge National Laboratory. The following paragraphs present a brief description of the reactor and its safety features. Appendix D presents the reactor Design Data Sheets with lists of the specific data concerning the reactor.

B. Reactor Proper

The fuel for the reactor will be enriched uranium alloyed with aluminum and made up into 2S aluminum clad plates .060" thick, see Figure 1. Eighteen plates approximately 2.5" wide and 23 5/8" long will be made into fuel assemblies with .118" spacing between the fuel plates, see Figure 2. Each plate will be built up like a picture in a frame. The 18 per cent uranium alloy will be rolled out, punched to size, inserted in a continuous aluminum frame, and rolled hot between backing and cover plates. This hot rolling results

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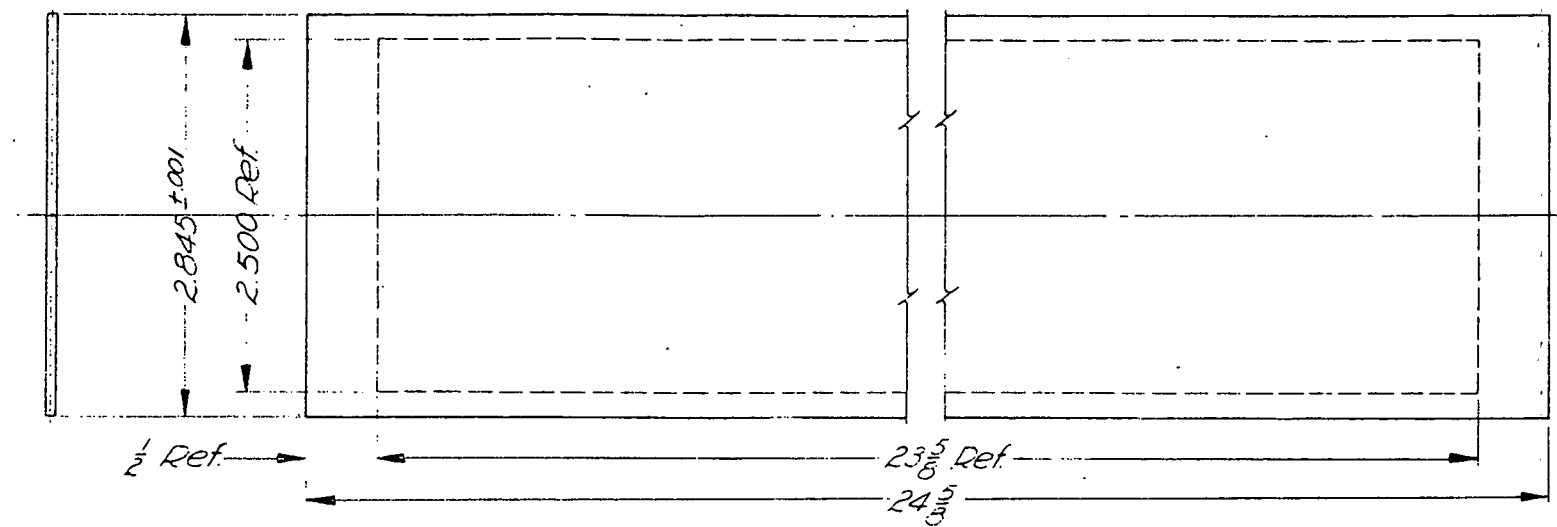
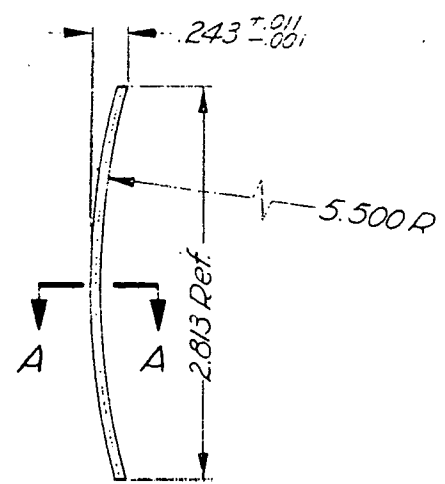


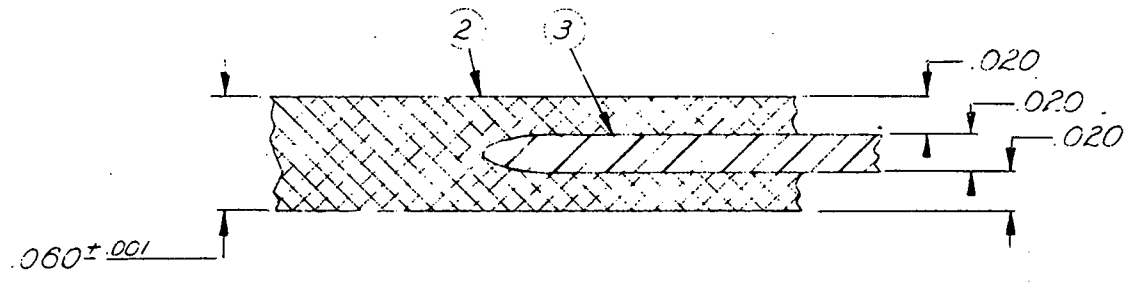
PLATE BEFORE BENDING
Scale: 1"=1"



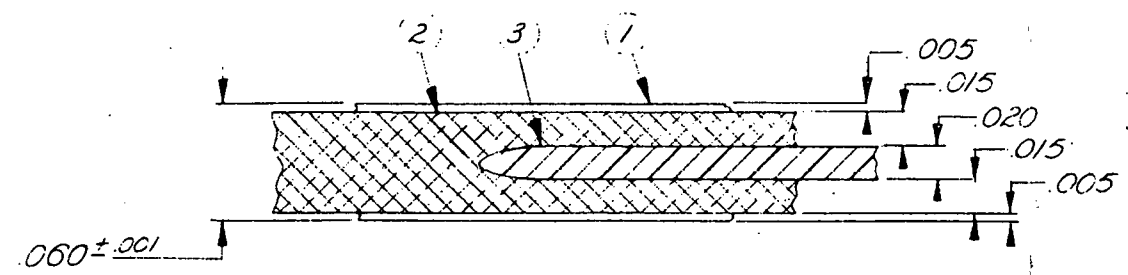
END VIEW
AFTER BENDING

Scale: 1"=1"

ITEM	MATERIAL	QTY
1	72S Aluminum	2
2	2S Aluminum	1
3	2S Alloy	1



SECTION A-A
Scale: Approx. 12 times size



OPTIONAL CLADDING
Scale: Approx. 12 times size

STANDARD FUEL PLATE

FIG. 1

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in complete welding of the plates and frame into a continuous heterogeneous plate. After trimming and bending slightly, the plates are brazed into aluminum side plates by use of a silica flux, suitable ceramic jigs and furnace brazing. These assemblies will be equipped with aluminum end boxes giving a completed assembly roughly 40 inches long by three inches square. Each assembly will contain about 140 grams of uranium²³⁵. Twenty-one or twenty-three of these fuel assemblies will be stacked together with their axes vertical to make the active section of the reactor approximately 73 x 23.4 x 60 cm. The reactor will be cooled and moderated with water flowing through the fuel elements of the active lattice at 30 ft/sec. The assemblies, of course, are held in place by suitable grids and supports. Outside of the active section, there is at least 30 cm of beryllium formed in a circular pattern; this in turn is held in an aluminum tank. The beryllium is largely ground and sintered-hot-pressed metal in the form of blocks through which there are many cooling holes. Figure 3 is a cutaway pictorial drawing of the various parts of the reactor core and the aluminum tank which contains the active core and beryllium reflector. The beryllium and fuel assemblies are also cooled by water flowing between the various sections. Experimental beam holes pass through the beryllium to the edge of the active lattice. These beam holes end in blanked off tubes welded into the inside of the aluminum tank.

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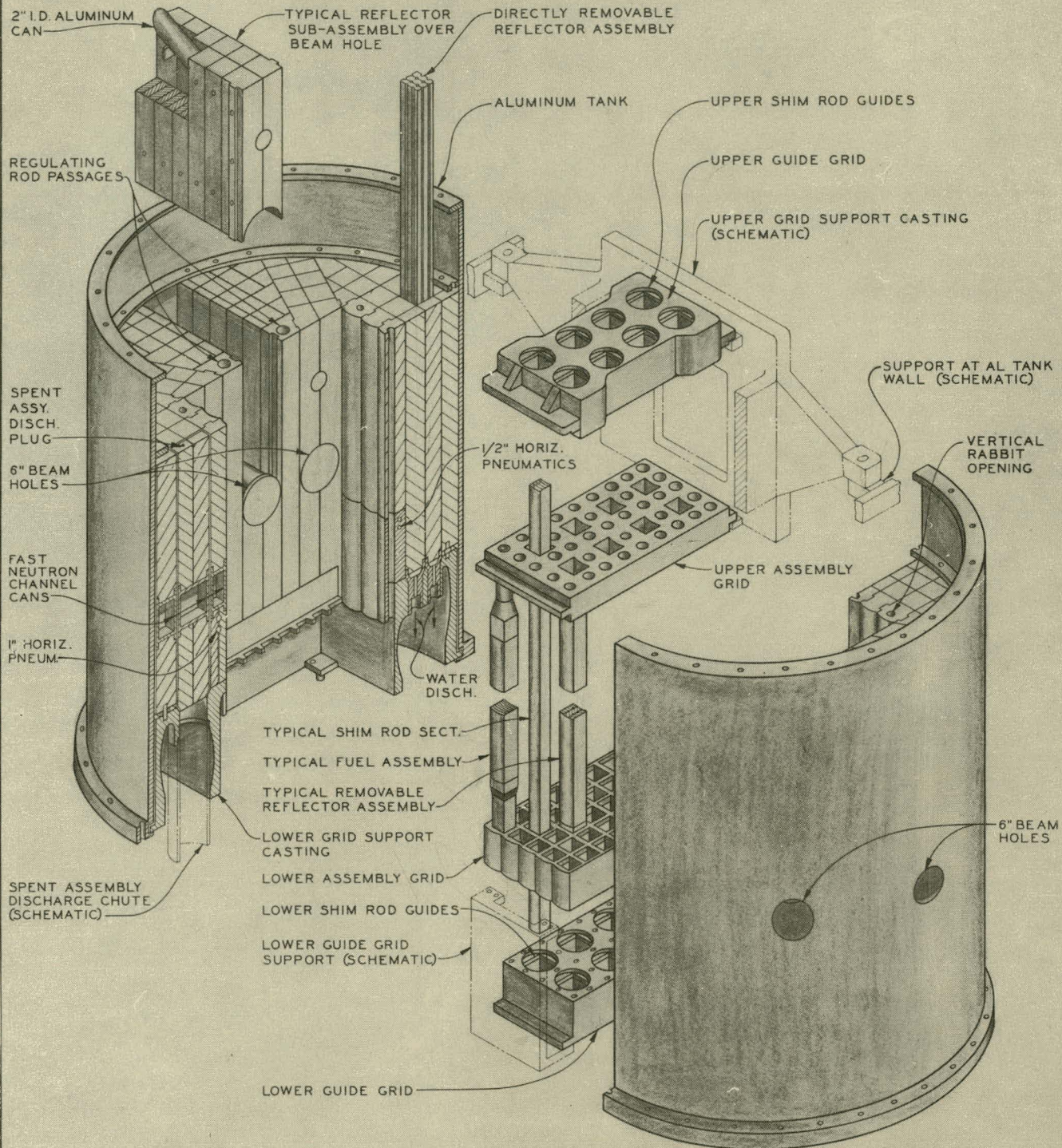


FIG 3
 PICTORIAL CROSS SECTION
 REACTOR TANK

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MATERIALS TESTING REACTOR

FUEL ASSEMBLY

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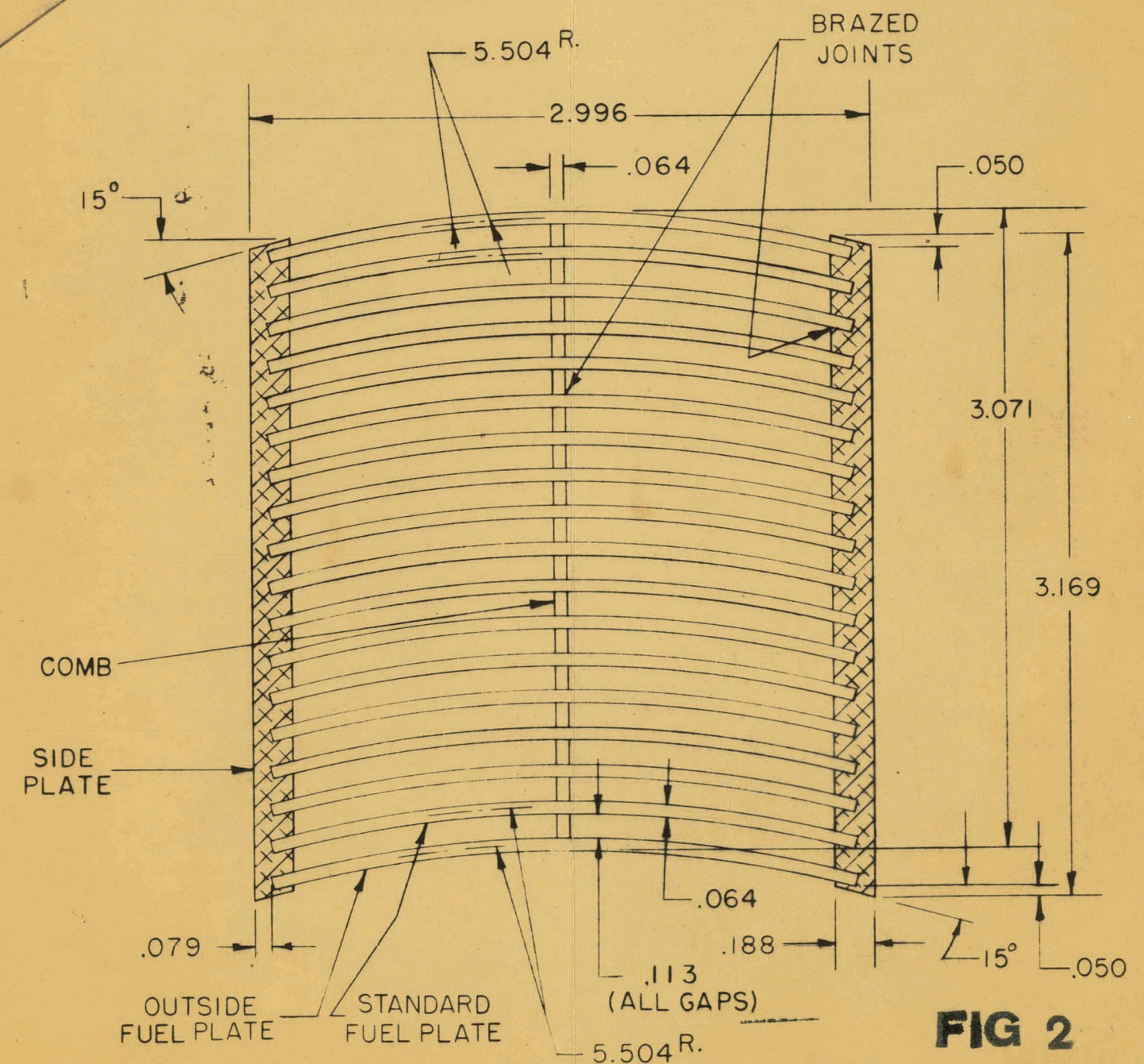
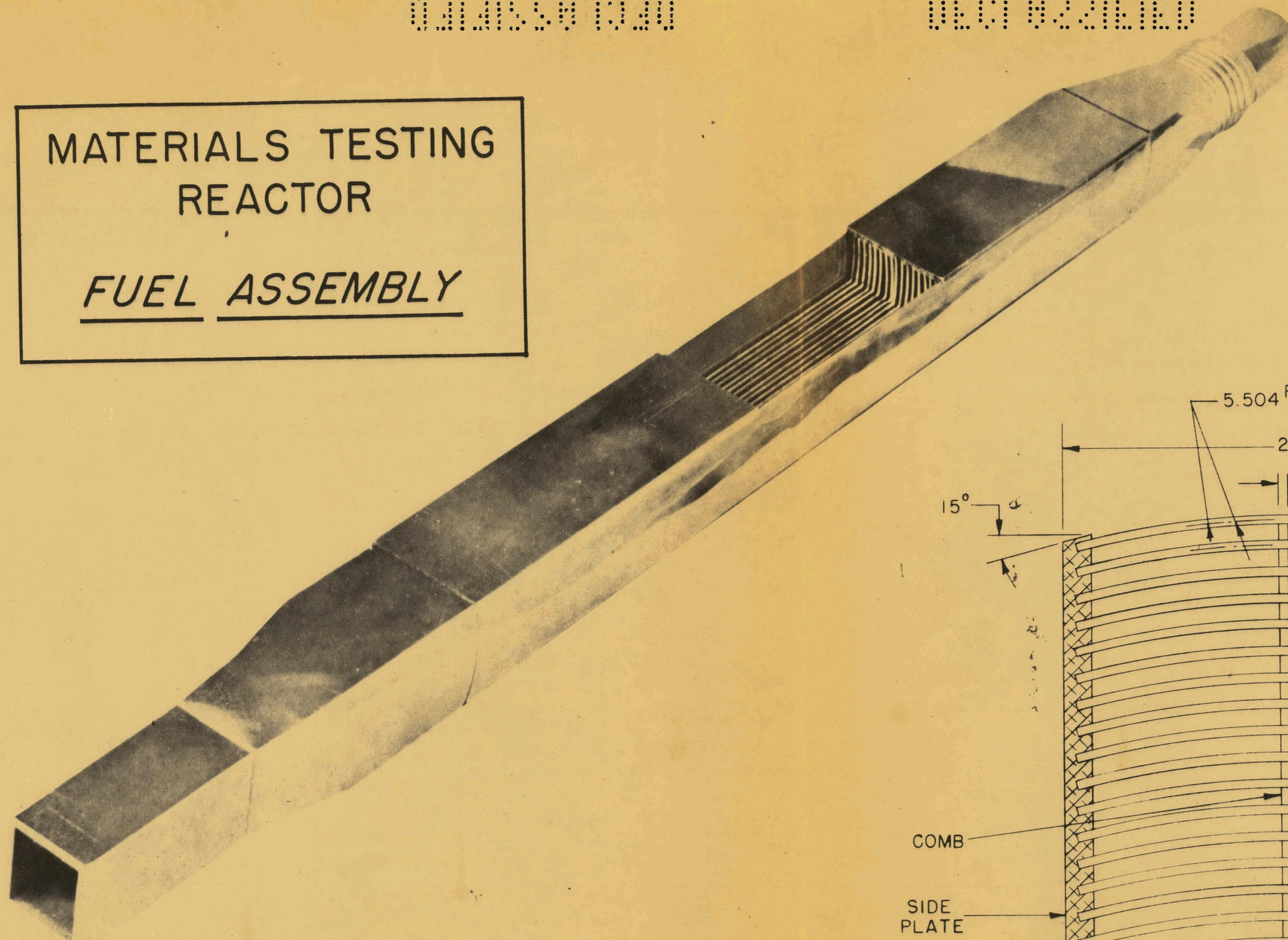


FIG 2

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The aluminum tank is surmounted by a second aluminum tank, an expansion joint, and by a steel tank. These tanks serve as water inlet calming tanks and provide sufficient water depth for shielding when the top plug is removed for the charging and discharging operations during shutdown. The aluminum tank is made from 2S aluminum alloy 1-inch thick, completely welded and machined to close tolerances. The so-called top plug seals the top of the reactor. This plug carries the motors, controls, and operating mechanisms for the regulating and shim-safety rods and also acts as a shield during operation.

Outside of the aluminum tank there will be a space filled with 1-inch diameter graphite balls for one to two feet to form a square and outside of these balls are several feet of block graphite. The graphite balls have replaced the beryllium oxide balls mentioned in the older reports and drawings. Outside of the graphite there will be a thermal shield consisting of two 4-inch thick steel plates and a barytes concrete biological shield approximately 9 feet thick. 1850 lbs/min of air is drawn between the thermal shield plate and through the block graphite and balls for cooling. Figure 4, DRP-63, is a drawing of the horizontal section of the reactor through the midheight plane. This drawing also illustrates the horizontal experimental facilities at that level.

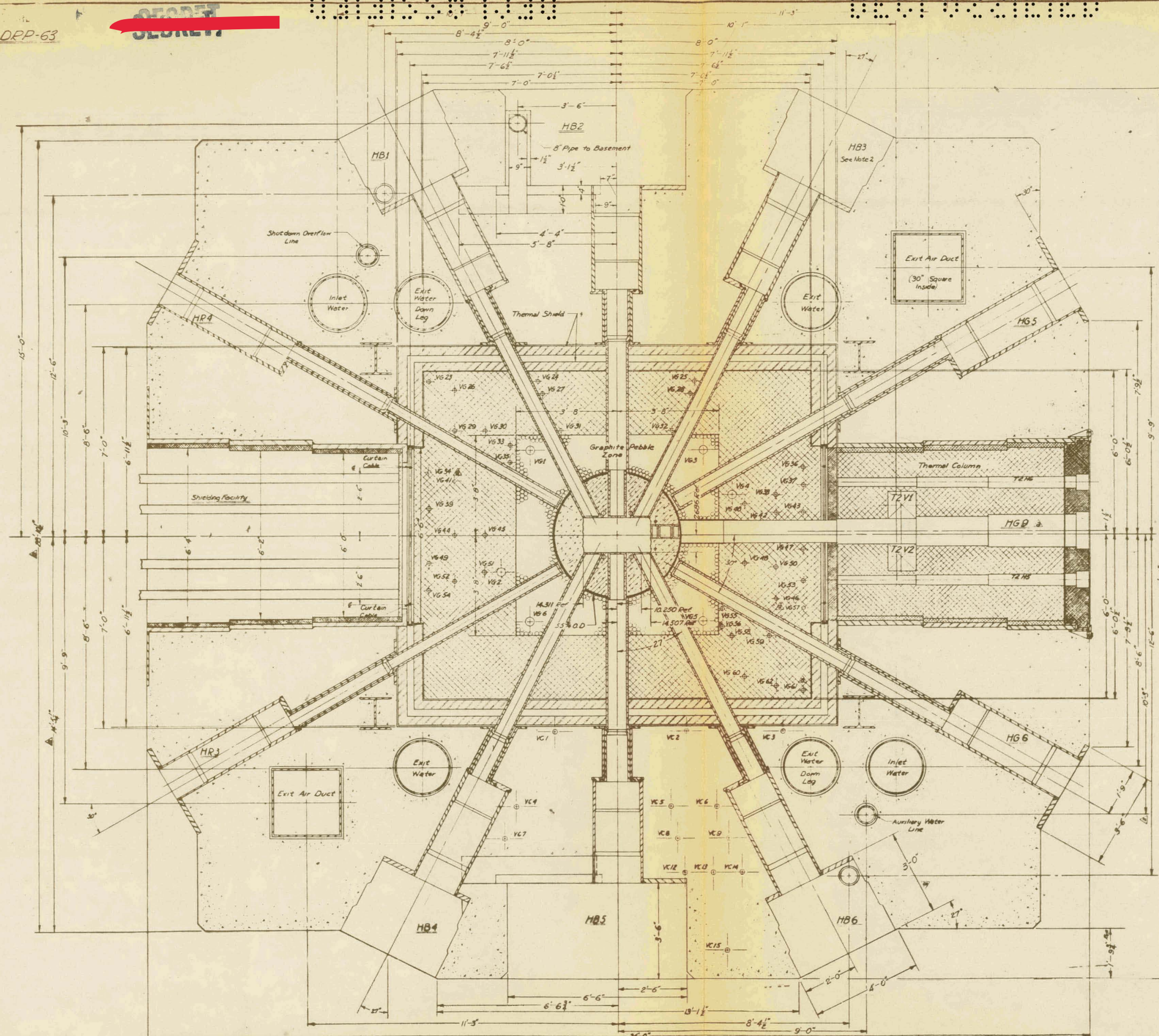
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REFERENCE DRAWINGS

- DRP-47 Structural Steel Proposal
 DRP-61 Reactor Structure-Top Plan
 DRP-62 Reactor Structure-Plan View
 DRP-64 Experimental Facilities Written disclosures
 DRP-65 Graphite Reactor Tank-Vertical Section thru E-W
 DRP-66 Reactor Structure-Vertical Section thru N-S
 DRP-67 Reactor Structure-Vertical Section thru E-W
 DRP-73 Experimental Shielding Facility, Horizontal Section
 DRP-77 Reactor Structure, Layout MB1, E-W Liners
 DRP-78 Reactor Structure, Vertical Section, MB1, 4'x6' DB1, 3'x6' Liner Details
 DRP-79 Reactor Structure Horizontal Section, MB5
 DRP-80 Reactor Structure, Vertical Section, MB1, MB2, 4'x6' DB2, 5' Liner Details
 DRP-81 Reactor Structure, Vertical Section, MB3, 4'x6' Liner Details
 DRP-83 Reactor Structure, Thermal Column Graphite Arrangement
 DRP-87 Reactor Structure, Experimental Facility Plug Liner Orientation

NOTES

1. For small vertical hole location see
 Drawg. DRP-61
 2. MB3 Cubicle incomplete.

CAUTION

FIG 4

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APPD.	DATE	OAK RIDGE NATIONAL LABORATORY
TECHNICAL DIV.	P.O. BOX 9	OAK RIDGE, TENN.
MTB	105 BLDG	
REACTOR STRUCTURE		
HORIZ SECTION REACTOR &		
FUNDAMENTAL DIMENSIONS		
DRW. BY	DATE	SCALE
DRW. NO.	REVISION	APP. DATE

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Figure 5, CSK 184E, is a drawing of the vertical section of the reactor through the active lattice. This figure indicates the water flow. The two inlet water lines enter the reactor structure from the bottom, pass to the top through the concrete shield and enter the top tank. After passing through the active lattice the water exits through two water lines at the bottom of the tank, turns and passes up above the top level of the lattice, then down and out through the concrete shield. The purpose of these loops in the exit lines is to assure that the water in the aluminum tank is not drained out in case of external water line failure. Siphon break lines pass from the top of the reactor to the top of each loop. These siphon break lines will not be equipped with valves. In case the aluminum tank ruptures, the water will leak into the thermal shield and concrete shield which will prevent rapid loss of the water. Excessive leakage of the water through the experimental holes in these shields is minimized by comparatively tight fitting plugs which are in place during operation.

Figure 6, DRP 68, is an illustration of the appearance of the outside of the reactor. Figure 7, MTR 3018E, illustrates the design of the experimental plugs. The experimental plugs for the six horizontal beam holes actually penetrate the aluminum tank and the beryllium reflector as shown in Figure 3. The aluminum tank is made with internal sleeves welded on the inside of the tank and open externally. The beryllium reflector is built around these beam hole

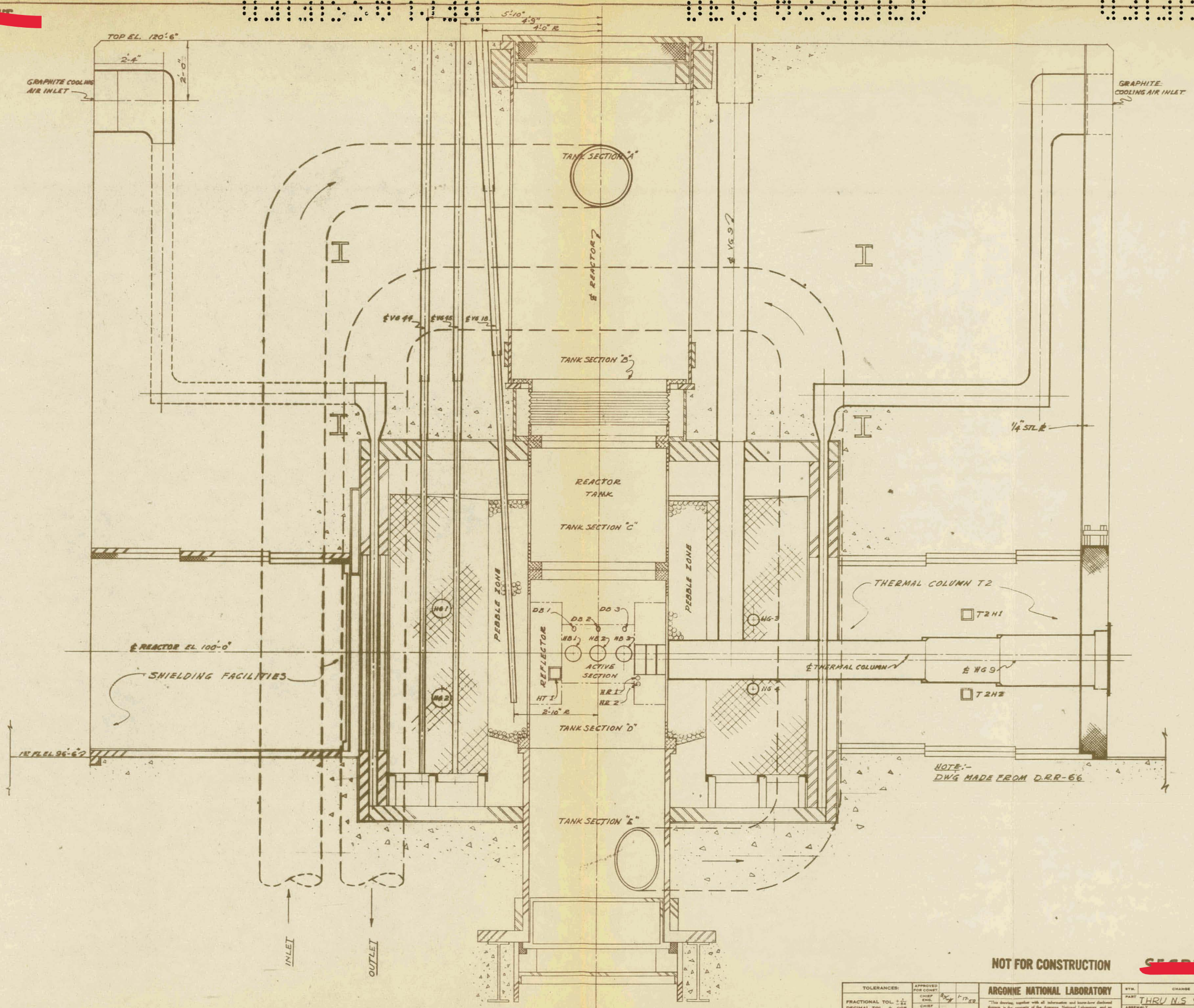


FIG 5

TOLERANCES:		APPROVED FOR CONST.	ARGONNE NATIONAL LABORATORY		SYM.	CHANGE	DATE	BY
FRACTIONAL TOL. $\pm .01$	CHIEF ENGR.	[Signature]	THRU N.S. & OF REACTOR		PART			
DECIMAL TOL. $\pm .001$	CHIEF DRAFTSMAN		VERTICAL SECTION		ASSEMBLY			
ANGLE $\pm .1^\circ$	CHECKED	1-17-58	SCALE		NO. RECD.			
UNLESS OTHERWISE NOTED	DRAWN	11-51 GAK.	SCALE		NO. RECD.			
	NOTED		SCALE		NO. RECD.			

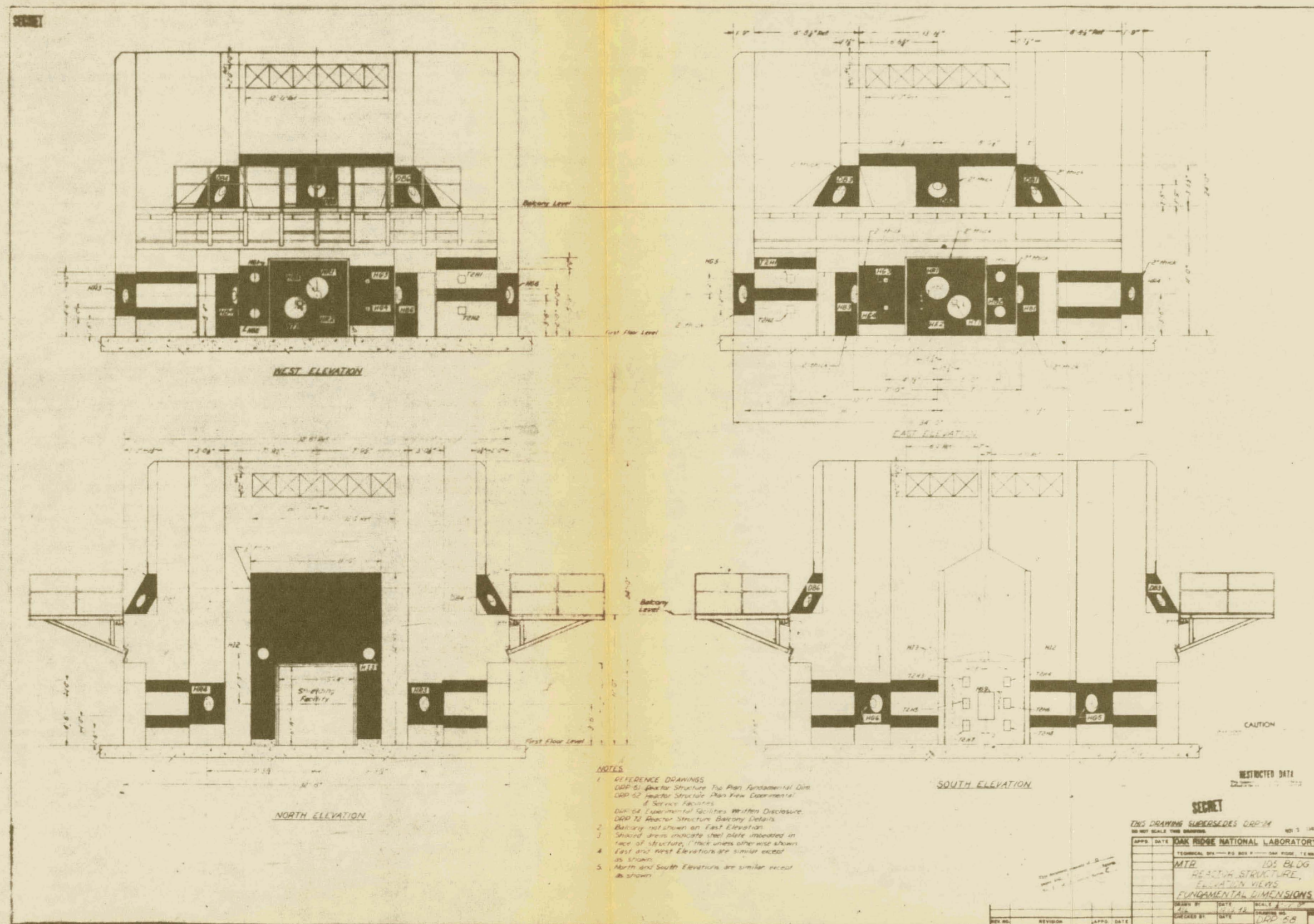


FIG 6

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liners or sleeves. The beam hole liners in the graphite zone are aluminum tubes which end just short of the aluminum tank and are held by the heavy steel liner in the biological or concrete shield, see Figure 7.

The plugs proper for these six holes will be made of aluminum tubes containing beryllium and graphite in those regions which will be in the beryllium and graphite regions within the reactor. The beryllium will be cooled by water passing through connections on the outer ends and by tubes through the graphite with water flow directly over the beryllium within the plug tube. Flow and temperature instruments will sound an alarm in the control room in case of water failure. The outside aluminum tube of the plug and the beam hole liner will be cooled by 20 to 30 cfm of air flowing between the two tubes from the outside of the reactor to the inner end of the hole liner just outside of the aluminum tank. This air will enter the main reactor cooling air stream at this point in the graphite ball zone.

The six main down beam holes and the HG9 beam hole, see Figure 8, will end at the outside of the aluminum tank. The six down beam holes have two inch diameter hollow cylinders as beam channels through the beryllium section within the reactor, and the HG9 hole has hollow sections in the beryllium, including the removable plug in the discharge chute, as shown in Figure 3.

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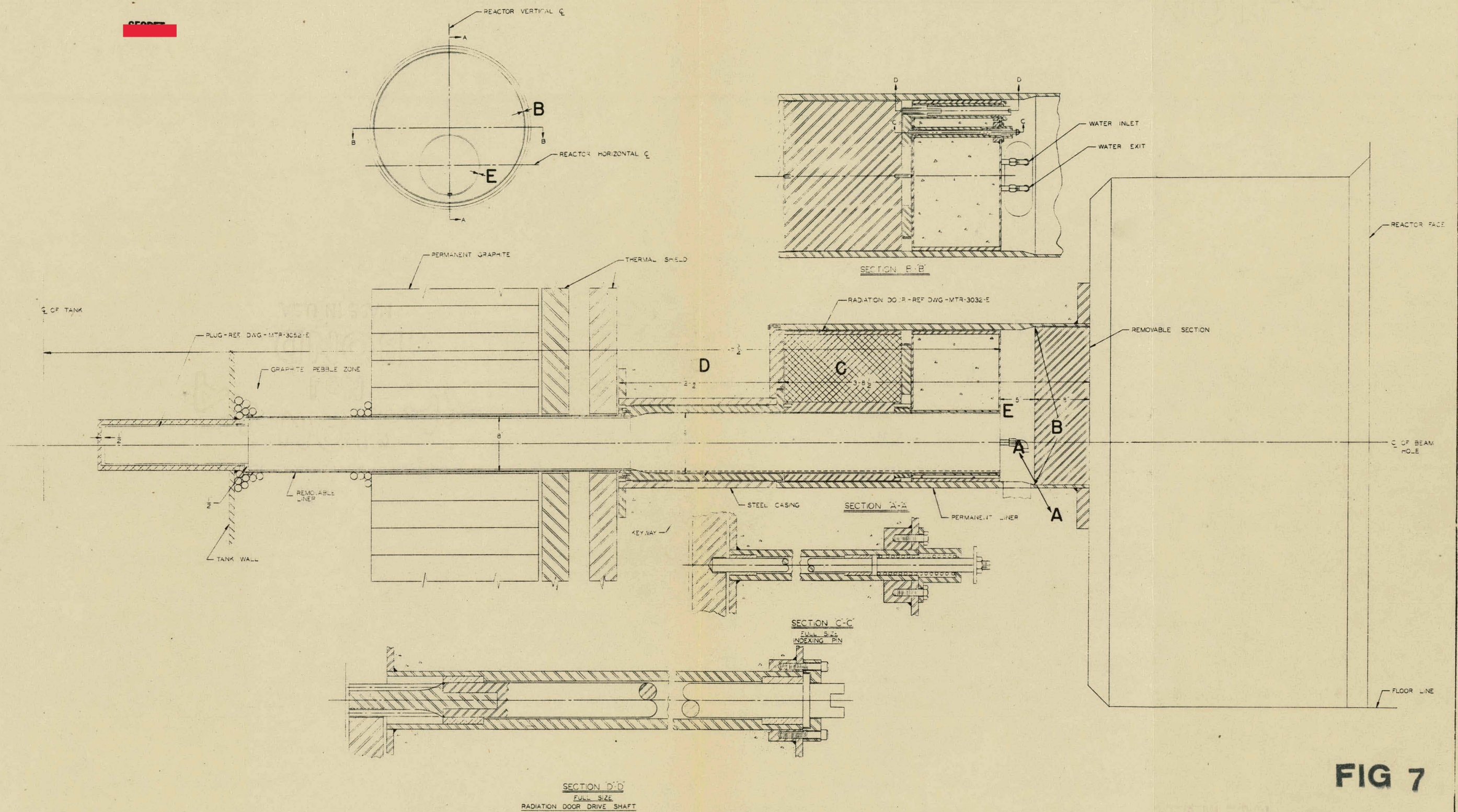


FIG 7

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The document contains restricted data within the meaning of the Atomic Energy Act of 1954 and information affecting the national defense of the United States within the meaning of the Espionage Act, 18 U.S.C. 793 and 794, and the transmission or the revelation of its contents in any manner to an unauthorized person is prohibited and may result in severe criminal penalties.

APPROPRIATE AGENCY	APPROPRIATE AGENCY
DATE	DATE
BY	BY
FOR	FOR
REMARKS	REMARKS
HORIZONTAL BEAM HOLE ASSY - H-B-E-N-B-S	
MTR-3053-E	

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None of these plugs can be removed while the reactor is in operation. During shutdown the main plugs can be removed to coffins and the coffins removed only when the 12" lead doors are closed. During the operation of the reactor, not only must the plugs be in position, but the heavy lead lined concrete outer doors must be in place.

The HG9 hole does not have a lead door but rather stepped plugs. Likewise, the vertical holes do not have lead doors nor is it necessary that the vertical plugs fill the holes for their entire length.

When the horizontal and down beam holes are to be used for experimental work, new plugs containing the experimental equipment will have to be built. These also must fill the holes as nearly full as the experiments will permit. It is estimated that the largest beam that will ever be permitted to escape from the reactor will be 0.1 square inch cross section.

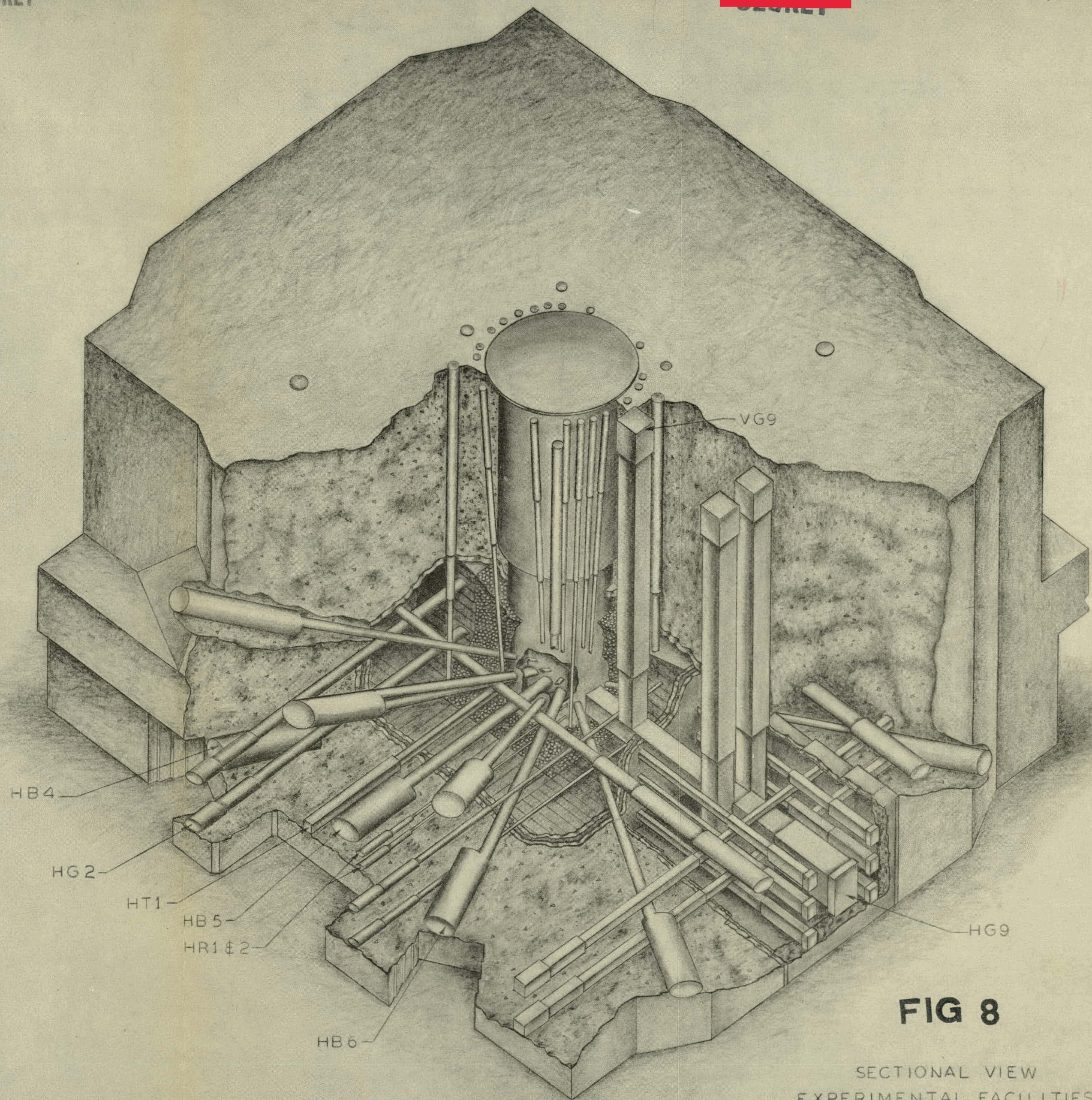
The neutron flux of the reactor will be about 2×10^{14} thermal and 1×10^{14} maximum fast ($E > 1$ mev) flux. The thermal flux will vary approximately as illustrated in Figure 1 of Appendix D. It is emphasized that this will be a "thermal" reactor.

Two features of the reactor should be mentioned. The first of these is that the maximum slow neutron flux exists not in the active core of the reactor but in the beryllium reflector approximately 5 cm from the core. This results in the maximum heat transfer

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**FIG 8**SECTIONAL VIEW
EXPERIMENTAL FACILITIES

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occurring at the edges of the active portion rather than in the center of the active portion. The actual heat flux varies from 322,000 Btu per hour square foot (24.2 calories per square cm sec) to 502,000 Btu per hour square foot (37.7 calories per square cm sec). The slow neutron flux density is higher by a factor of 2 at a distance of 5 cm from the active section than the average density within the active section.

The second point of interest is that the reactivity of the reactor is dependent upon the rate of growth and decay of Xe^{135} . The Xenon in turn results from the decay of I^{135} formed in the fission process. Under steady operating conditions at 30,000 kw, the loss of activity due to the Xe^{135} is about 4% δ^k/k . After steady operation at full power the fission product I^{135} , which has a half-life of 6.6 hours, decays continuously to the Xe^{135} , which has a half-life of 9.4 hours and a very high cross section. The net result is that on shutdown, the decay of the I^{135} to Xe^{135} results in a continuous loss of reactivity. One-half hour after shutdown, the reactor will have lost 9% of reactivity, and 2 hours after shutdown, it will have lost 16% of reactivity. The maximum loss of reactivity occurs about 11 hours after shutdown and amounts to 39% over the 4% loss due to the steady state concentration of Xenon under steady operation at 30,000 kw. After 11 hours the loss of reactivity decreases and becomes only 4% greater than during steady operation at 48 hours after shutdown and equals the normal operation reactivity 63 hours after shutdown.

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It has been considered unsafe to build into the reactor sufficient reactivity to override the Xenon at all times and thus permit shutdowns of any length. The actual design has been based on the premise that accidental shutdown could be corrected and the reactor started one-half hour after shutdown. If this is not done, it will be necessary to partially or completely recharge the reactor or to let it remain shut down for 48 hours. In order to overcome Xenon poisoning to this limited extent, 19% excess reactivity has been designed into the reactor and 5% $\delta k/k$ allowed for experimental work. All these changes in $\delta k/k$ are listed in the MTR Design Data Sheets attached as Appendix D.

As a consequence of the high heat transfer rates and the high reactivity of the reactor, the design of the reactor proper has required an extremely large amount of analysis and experimental development work. The design of the entire project, in fact, is based on the premise that adequate heat transfer must be provided and adequate control devices must be provided to really control the excess reactivity at all times.

C. Control Rods

The control system for the reactor consists of instruments such as ion chambers within the reactor to indicate the neutron intensity level, period meters, electronic instrumentation to interpret the data and control the shim-safety and regulating rods within the

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reactor. The two regulating rods consist of cadmium containing rods placed in the beryllium just outside of the active lattice, each rod being capable of controlling $.005 \delta^k/k$. These rods have an acceleration of 1.3 g and are operated by one horsepower motors. One rod only is used for control, while one is standby and remains fully withdrawn. Two other regulating rod positions are provided for use with other loading designs. These holes can be used as experimental holes when not used with rods. These rods are discussed in detail in part F 1 of this section.

The shim-safety rods consist of two types of complex rods one of which has a cadmium absorption section at the top followed by an enriched uranium section similar to the fuel assemblies but with only 14 fuel plates. Operation will require a minimum of three rods of this type. The other type can have thorium absorption sections on top and probably will have beryllium in the bottom sections. These will be located outside the active section. All these rods end just within the top of the active lattice, when they are in the full absorption position. These rods are attached by means of magnets to the motorized rods above, thus making integral units when the magnets are energized. When the safety system causes "scram", for any reason, the magnets become de-energized and the absorption sections of the shim-safety rods drop into the reactor core. One cadmium section has approximately $0.07 \delta^k/k$ absorption. Release time for the magnets is less than 30 milliseconds.

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A more complete description of the control system is given in a later section. The details of the system can be found in the following references, all prepared at Oak Ridge National Laboratory:

Ion Chamber	MonP 203
General Theory of Pile Control	MonP 271
Physics Division Quarterly	MonP 314
Ionization Chamber	MonP 316
Physics Division Quarterly	MonP 368
Guide to Literature on Pile Control	MonP 385
Physics Division Quarterly	MonP 437
Electrically Driven Servo	ORNL 2
Instrument for Measuring the Log. of Neutron Level and Period of a Pile	ORNL 110
Physics Division Quarterly	CNL 35

D. Water System

The following paragraphs present a brief description of the water system within the reactor and outline the main features of the water supply system. This system has been designed to assure that the reactor tank will be kept filled with water in spite of any foreseeable accidents or sabotage. Figure 9, Drawing MTR 3034-D, is a flow diagram of the water system. Figure 10 shows a pictorial view of the pipe lines within the reactor.

1. Primary Cooling System.

The 20,000 gallons of water per minute needed to cool the reactor are pumped to an overhead working reservoir of 150,000 gallon capacity. The water then flows by gravity through measuring devices and through the reactor to a seal tank. As noted previously,

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MATERIALS TESTING REACTOR COOLING SYSTEM

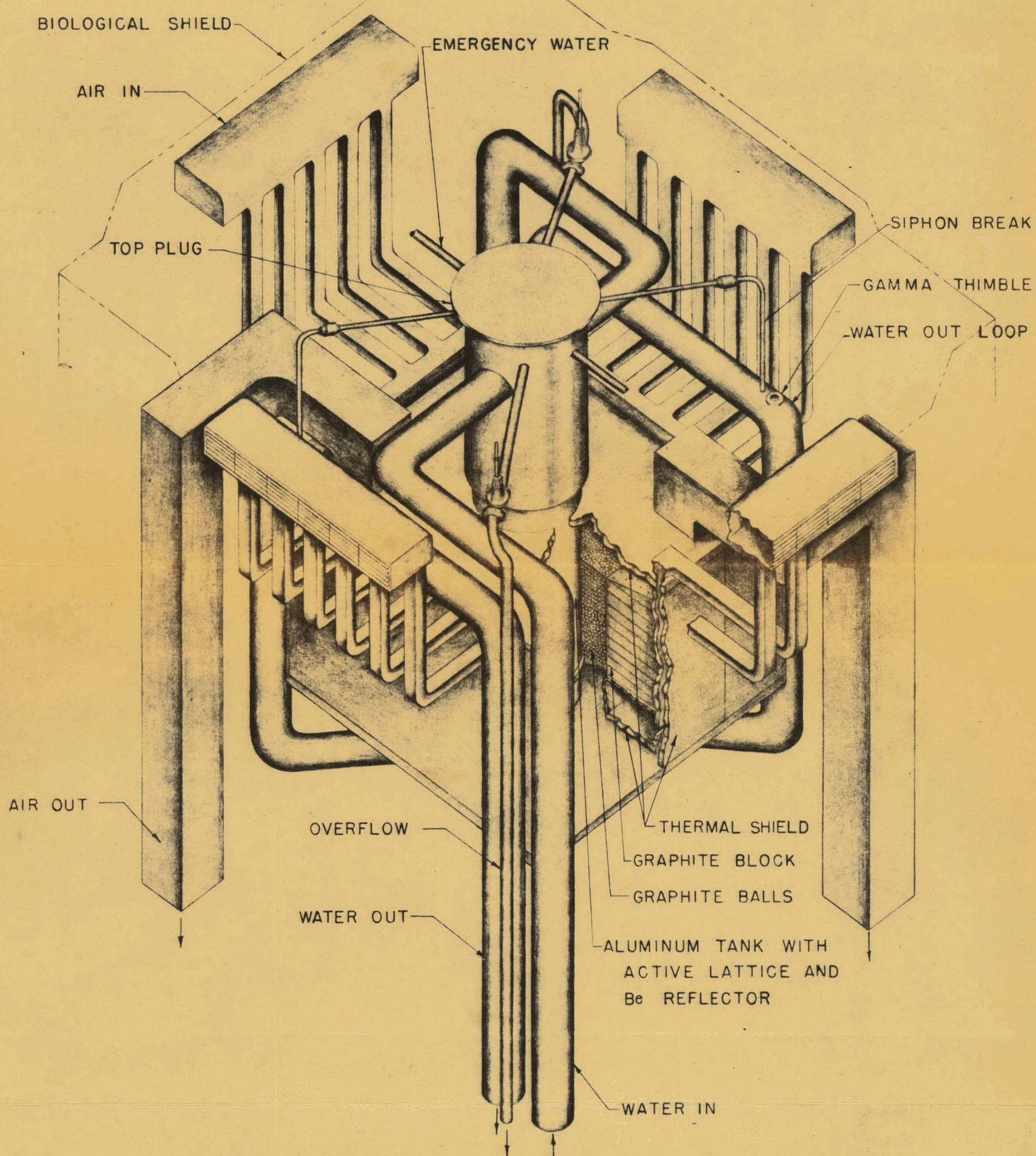


Fig-10

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AUTHOR WINKLER
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FIG 10

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the flow through the reactor is designed so that the active section will be kept covered by water in case of failure of any external water line. The water enters the reactor tank at the top by means of two pipes, passes down through the active uranium section and the beryllium and out of the bottom of the tank, then turns upward in two pipes to form loops within the shield which extend above the lattice. The water lines then turn downward and out through the bottom of the reactor and to the seal tank. Vent lines which will have no valves are provided between the top of the reactor tank and the tops of the exit water pipe loops. The water level in the seal tank will fluctuate during operation of the reactor. During shutdown a depth of 20' 6" above the center line of the active lattice in the reactor will be maintained by a weir. From the seal tank, the water is drawn by vacuum through spray evaporators for cooling and deaeration. Flow from the evaporators to a sump tank is by gravity. The water is then pumped to the working reservoir for another cycle.

Since the water must be kept very pure, all the tanks except the reactor tank itself and all the pipe lines in this primary system will be made of stainless steel or stainless steel lined piping with all piping joints welded. The pumps and valves in the pump room will be made of cast iron flanged to the stainless steel piping.

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The purpose of the 150,000 gallon overhead working reservoir is to give sufficient capacity to keep the reactor in operation during very brief electrical outages and to provide sufficient water to cool the reactor in case the power outage is long enough to require the reactor to be shut down. If a power outage over 30 seconds duration occurs, the reactor controls will be reversed and the reactor shut down. A secondary purpose of the reservoir is to provide sufficient water to fill the reactor thermal shield in case of failure of the aluminum tank.

The seal tank is designed to keep the water level in the reactor well above the active lattice at all times. The piping and tank will be made of fully welded stainless steel, the piping buried six feet or more, and the seal tank placed in a shielded room. The weir on the seal tank inlet will be 19' 6" above the top of the lattice in the reactor.

In case of power outage over a few seconds duration, the emergency power system will be started. This emergency power system will be described briefly later in this report. As soon as the power outage occurs, the circulating pumps will stop, but as soon as the emergency power system begins to function in a matter of seconds, an emergency pump will start to circulate 1,000 gpm from the seal tank to the reactor. This pump will continue operation during the emergency shutdown. The fission product decay and pumping energy will raise the water temperature 16.4°F in 24 hours after shutdown, neglecting all heat loss due to conduction.


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2. Demineralized and Filtered Water Systems

Additional safety is assured by the availability of water direct from the demineralized water pumps through an 8-inch pipe connected directly to the inlet pipes to the top of the reactor. This line is controlled by the reactor operator. The ground level demineralized water tank will hold 100,000 gallons. An additional 8-inch line will run directly from the 150,000 gallon overhead filtered water storage tank to the top of the reactor. This line will be equipped with a spool piece that must be manually installed before the line can be used. This line in turn will be fed by pumps from the 1,500,000 gallon raw water storage tanks. One of these pumps will be operated by either the regular electrical circuit or the emergency electrical power system, and another pump will be operated by a standby gasoline engine directly connected. Finally, the fire loop will be fed from the raw water tanks by means of a Diesel driven pump. One of the 850 gpm well pumps will be designed to operate by the emergency electrical system. This pump will not be automatically started but will be manually started for day operation only. The emergency electrical system will be powered by a central Diesel engine and generator. This Diesel will not run continuously but will be started automatically whenever a power outage of over a few seconds occurs. It is planned to use delay relays which will switch from primary power in about five seconds and automatically start the Diesel. The delay relays will prohibit switch-overs every time a power surge occurs. The standby Diesel power will start and come to full power in one minute or less.

E. Reactor Air System

In order to cool the graphite 1850 lbs/min. of air will be drawn into the top of the reactor, down between the thermal shield plates, up through the graphite, and down through ducts in the concrete shielding, see Figures 10 and 11. From the reactor, the air will go to fans and the stack. Additional air will be drawn in around the experimental hole plugs to cool these plugs. This air will leave the reactor with the main air stream.

Since the inlet and outlet air enters and leaves the thermal shield above the reactor lattice, water would be trapped in this shield in case of failure of the aluminum reactor tank. Thus, water would leak out around the experimental hole plugs and might overflow into the air outlet.

The air from the reactor will contain about 1500 curies of Argon⁴¹ per day or 3.4×10^{-5} curies per cu ft average temperature and pressure of air at the top of the stack. This air will be exhausted through a stack downwind from the reactor. The stack height has been fixed at 250 feet. This is discussed further in Section XIV of this report. One of the two auxiliary exhaust fans will be motor^{driven} and the other will be equipped with a gasoline engine drive that will start automatically on failure of the motor driven fan or the auxiliary standby Diesel generator to pick up the fan load during power outages. Ordinarily one of these auxiliary fans will be operated

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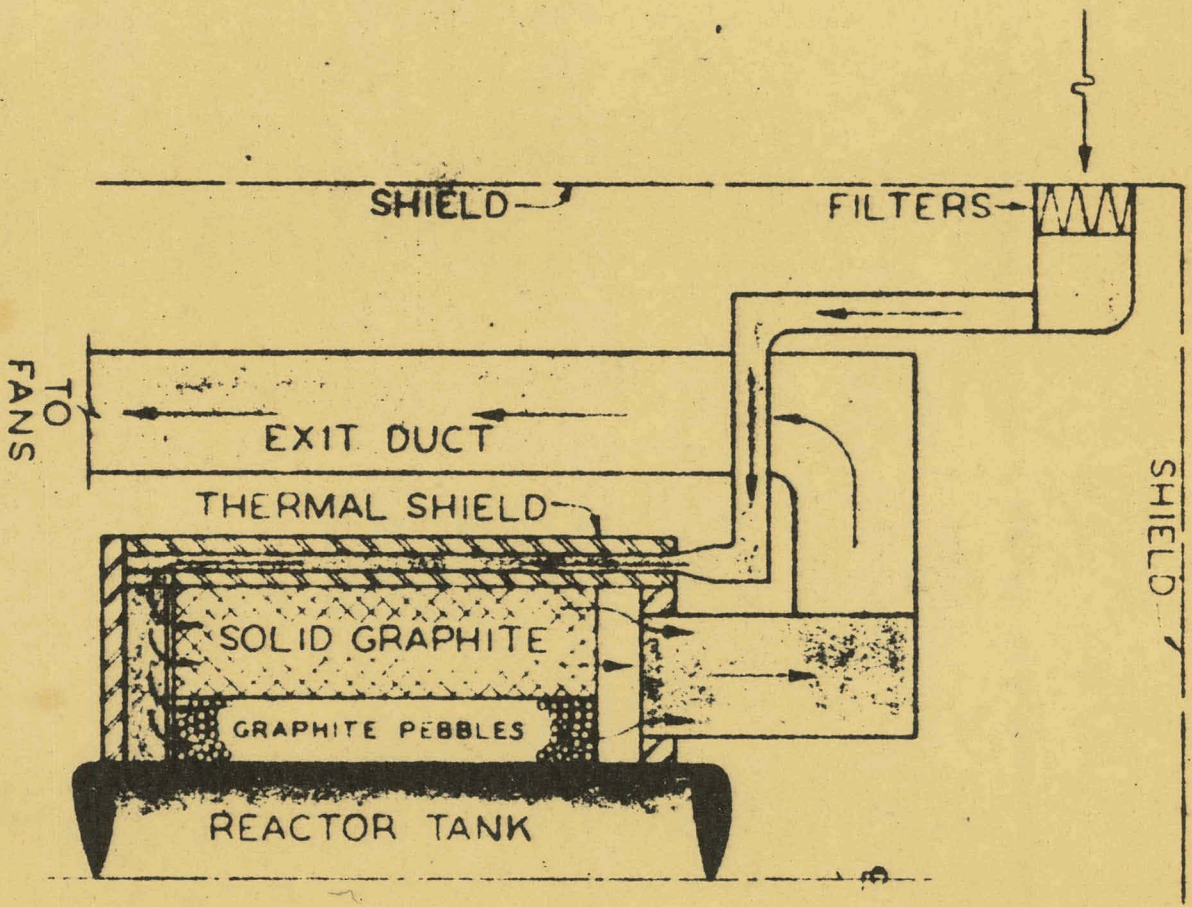


FIG 91

MTR AIR SYSTEM
SCHEMATIC FLOW DIAGRAM

FIG 11

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while the reactor is shut down. The fans will have a capacity of 5000 cfm each. The temperature of the graphite will be allowed to rise during power outages from the time of the power outage until the reactor is reversed approximately 30 seconds later.

Meteorological studies will be made continuously by the Idaho Operations Office to make certain that the amount of activity in the air at the ground in the reactor area, is below $10^{-6} \mu\text{c/cc}$.

Oxidation of the graphite at 300°C will give a weight loss of approximately .01 per cent per day. Therefore, 300°C has been chosen as the maximum temperature for any of the graphite. This temperature occurs only in very small regions near the aluminum tank and at the inside edge of the block graphite.

F. Control System.

The Materials Testing Reactor, being a thermal reactor with a small volume core with highly enriched fuel elements and light water moderator, is capable of very short periods, $\sim 10^{-3}$ sec. minimum in the prompt critical range. For this reason its control system design has many automatic safety features not included in the controls of existing reactors. A period of 0.1 second can be controlled. The features that assure periods within a controllable range are itemized later in this section.

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The safety devices will prevent any period shorter than one second from occurring. A period < 7 seconds blocks withdrawal of the shim-safety rods; a period < 5 seconds inserts the safety rods; and a period of one second or less "scrams" the reactor. The system can be described as a combination of manual control by the operator and automatic control from instrument signals with the automatic safety system capable of overriding the manual system if the operator manipulates the controls in an unsafe manner. The control system is designed to "fail safe" in the event of system defects or malfunctioning. Also other types of reactor operation troubles are tied into the control system to reduce reactivity and to "scram" the reactor in serious cases.

1. Components of the Reactor Control System and Instrumentation.

The principal parts of the control system are:

- a. Sensing elements
- b. Regulating rods
- c. Shim rods
- d. Rod drive mechanisms
- e. Magnets
- f. Servos
- g. Period meters
- h. Interlocks
- i. Amplifiers

The sensing elements are ion and fission chambers. They are sensitive to the neutron density being generated in the reactor and provide signals which energize the instruments and control mechanism.

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The fission chamber is sensitive to neutron densities of very low order and is especially useful during the time the reactor is being started up. A counting rate meter which indicates the rate of neutron production and a galvanometer are energized from this chamber. The fission chamber is initially positioned near the active section in the reactor. As the power level of the pile increases the chamber is removed to predetermined positions in the VM holes to prevent damaging the chamber due to high neutron fluxes. This chamber with its associated instrumentation provides an indication of a change of rate of neutron production and the power level of the pile throughout the full power range of the reactor.

The other chambers are compensated or proportional type. One of the compensated chambers energizes a galvanometer directly, which indicates neutron density. This instrument is independent of any external power source.

Other compensated chambers energize period meters which indicate the pile period and also provide signals which energize the rod controls when the pile period is in one of the critical ranges described later.


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The proportional chambers energize both recording instruments and pile control rods. The instruments record the power level of the pile and the control rods are actuated to shut down the pile when power level exceeds 1.5 full power. Similar chambers also energize the servo mechanism which actuates the regulating rods.

Amplifiers are used intermediately between the chambers and the instrument or control load since the power output from the chamber alone would be inadequate to meet the demand.

Servos of the worm gear reduction screw thread type actuated by electric motors are used to drive the shim-safety rods. The regulating rod servos are of the rack and pinion type actuated by D.C. motors. The amplidyne control to this motor automatically maintains the reactor at a selected flux intensity within the range of 1% to 100% of full rated power of 30,000 kw. The servo moves the regulating rod directly and can indirectly cause the movement of the shim-safety rods by various travel limit switches on the regulating rods.

Four regulating rod positions are provided in the beryllium reflector at the edge of the active section. One of these is sufficient for fine control of the reactor and a standby is kept in the fully retracted position for fast insertion in case of emergency. This will be treated in some detail in later paragraphs dealing with


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safety interlocks. The two spare regulating rod positions are for use with different loadings but can be used for experimental holes. Each regulating rod is $1\frac{1}{2}$ in. in diameter and consists of an aluminum tube $128\frac{1}{4}$ in. long with a 24-inch long cadmium segment sandwiched between concentric layers of 3S aluminum to make up the neutron absorbing part of the assembly. The rods are driven by a d-c motor through a pinion and rack arrangement. The motor is driven by current from an amplidyne. There will be no brake on the system so that with motor current failure the rod will coast to its bottom position. The regulating rod will have acceleration sufficient to attain a velocity of four feet per second in 0.1 second. The full stroke of the rod is approximately 27 inches, but limit switches tend to keep the travel within a 12-in. center range. One regulating rod controls about 0.5% of reactivity, which is less than the effect of delayed neutrons. The full travel 0.5% $\delta k/k$ is chosen so that in the event of misoperation, the servo mechanism cannot change the reactor condition to prompt critical. The shim-safeties start to withdraw when the regulating rod hits its upper limit switch, but they are limited to a much slower speed of withdrawal than the regulating rods. Thus the various safeties have time to cause reversal of shim-safety rods before prompt critical is reached.

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There are eight vertical positions in the core of the reactor for shim-safety rods. It is planned that only seven of these will be used as such. The remaining one will be used as an experimental facility for access to the maximum fast neutron flux. Actually, three shim-safeties are adequate to operate the reactor for the normal condition. These rods are made up of an active bottom section $23 \frac{5}{8}$ in., 60 cm long, that has 14 fuel plates in it as compared to 18 fuel plates in a standard fuel assembly, an absorption section $23 \frac{5}{8}$ in., 60 cm long, and a magnet and drive rod assembly. Three or more of the shim-safety rods will have absorption sections made of cadmium, and the others will have thorium absorption sections. However, in initial operation, all rods will have cadmium absorption sections. The magnet is fastened to the lower end of the drive rod which extends up through the top plug to the drive mechanism. When the magnet is energized, it holds the top of the absorption section thus making an integral rod of the assembly. Any condition calling for "scram" deenergizes these magnets and drops the shim-safety rods into the maximum absorption position.

The drives for each of these rods is an a-c induction motor. Since it is impossible to increase the speed of such a motor under load to greater than synchronous speed, there is a definite limitation on speed of withdrawal of the shim-safety rods. The circuit

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is so designed that four speeds are available for increasing reactivity during start-up. "High speed" is the condition where all rods are being withdrawn continuously at the same time; "Intermediate" speed is intermittent withdrawal of all rods. The time cycle is $\frac{1}{2}$ second withdrawal followed by $4\frac{1}{2}$ seconds stop (none being withdrawn). The "Second speed" is one rod continuously withdrawn, and "Low speed" is intermittent withdrawal of one rod. Insertion is always at high speed unless the operator blocks the insertion of particular rods by a manual selector switch. Withdrawal and insertion conditions will be described in more detail in later paragraphs dealing with interlocks.

2. Shim-Safety Rod Control and Interlocks

Shim-safety rod motions must be organized so as to carry out the three functions of running, protecting, and maintaining the reactor. These effects are in turn obtained from three groups of causes: instrument signals, operator initiative, and regulator rod position, see Figure 13. The demands of these causative agents are complex and often conflicting. It is the function of the control circuits to impose on these demands a priority sequence and then to permit exercise of the appropriate function. Figure 14 shows the reactor control rod arrangement as designed for the mock-up.

The schematic diagram, Figure 15, is intended to show the sequence and interrelations of the various actions of the control

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Dr. No. 5258

ION CHAMBER BLOCK DIAGRAM

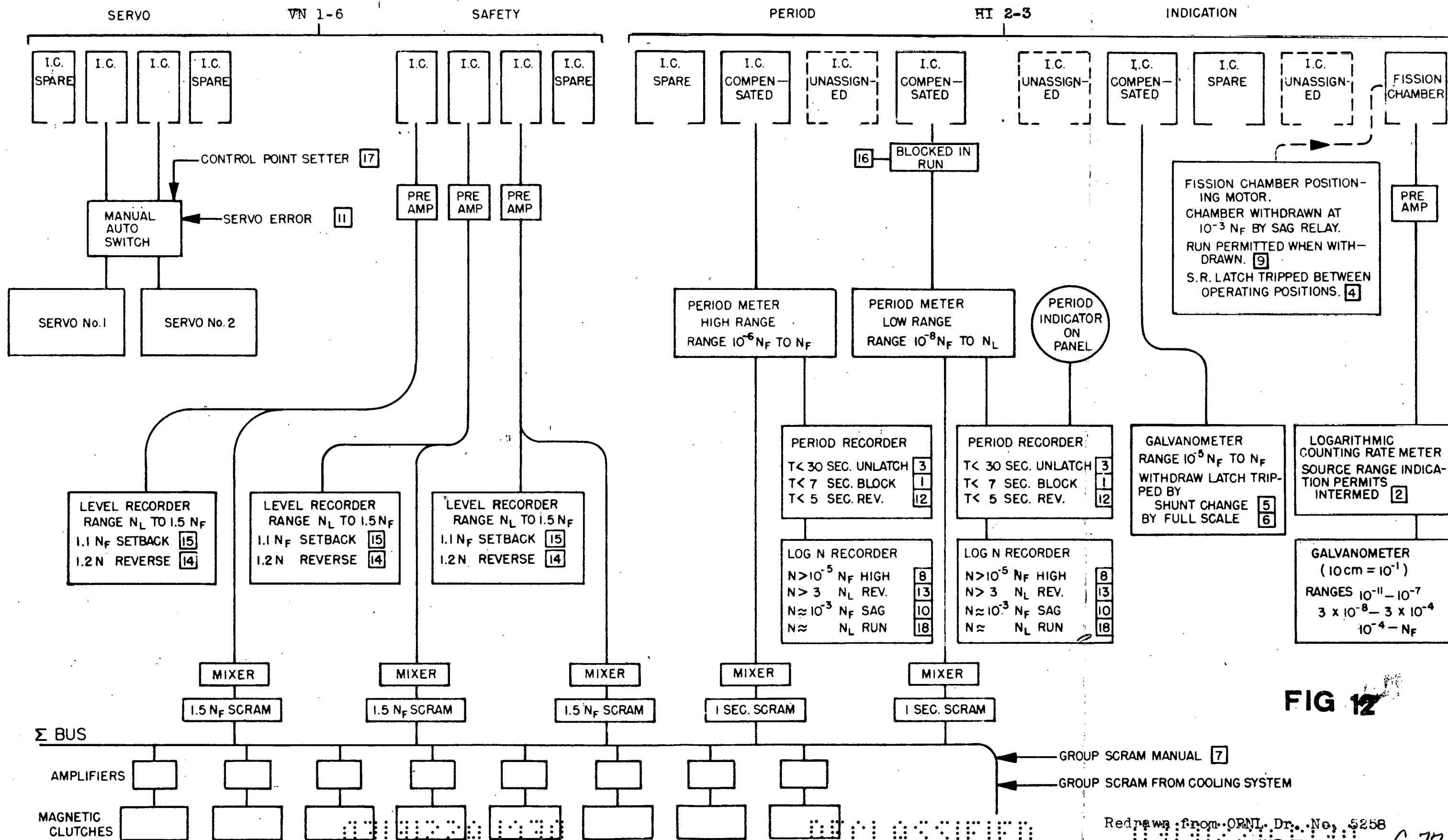


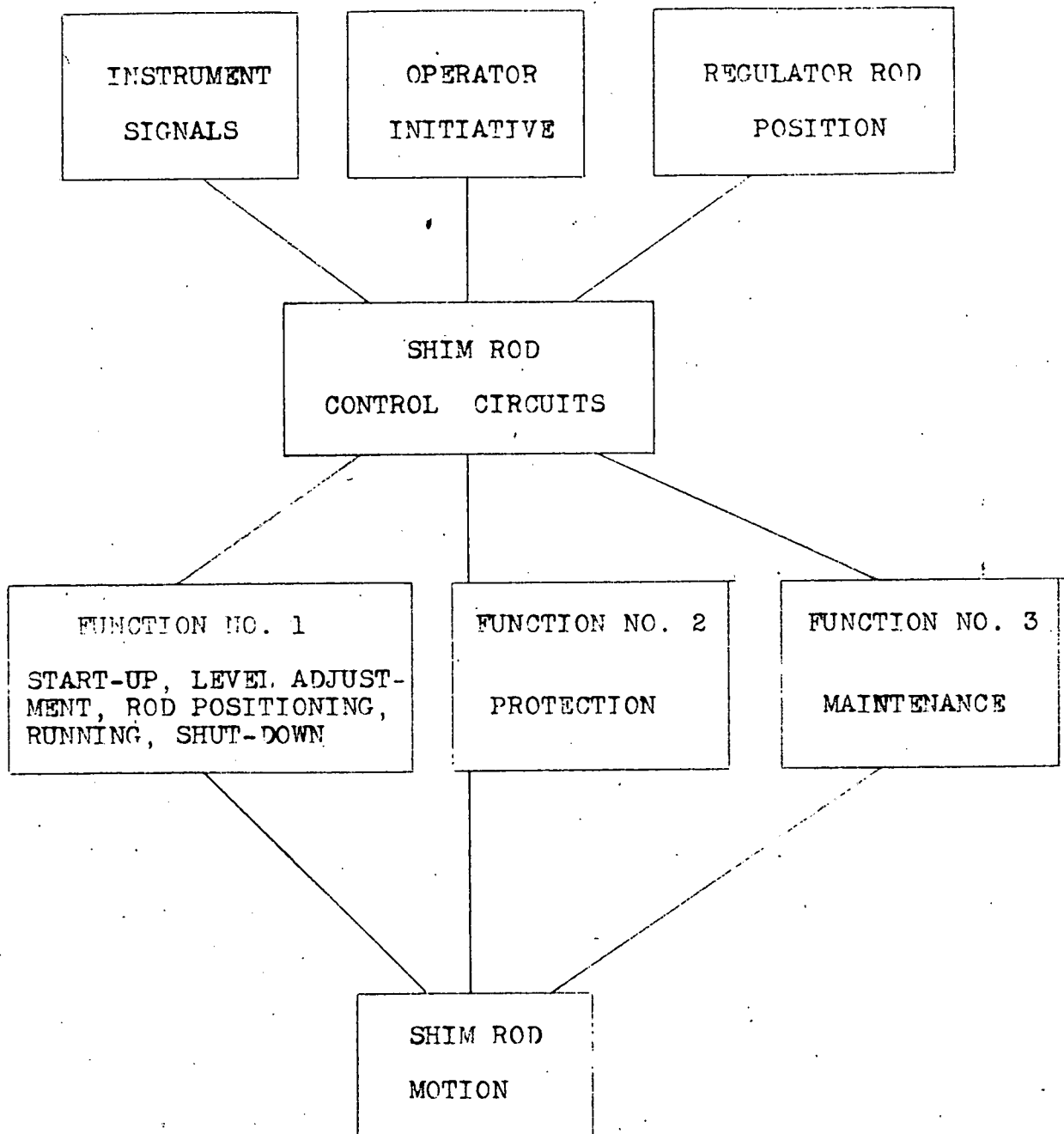
FIG 12

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FUNCTIONAL DIAGRAM OF SHIM ROD CONTROL

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FIG. 13

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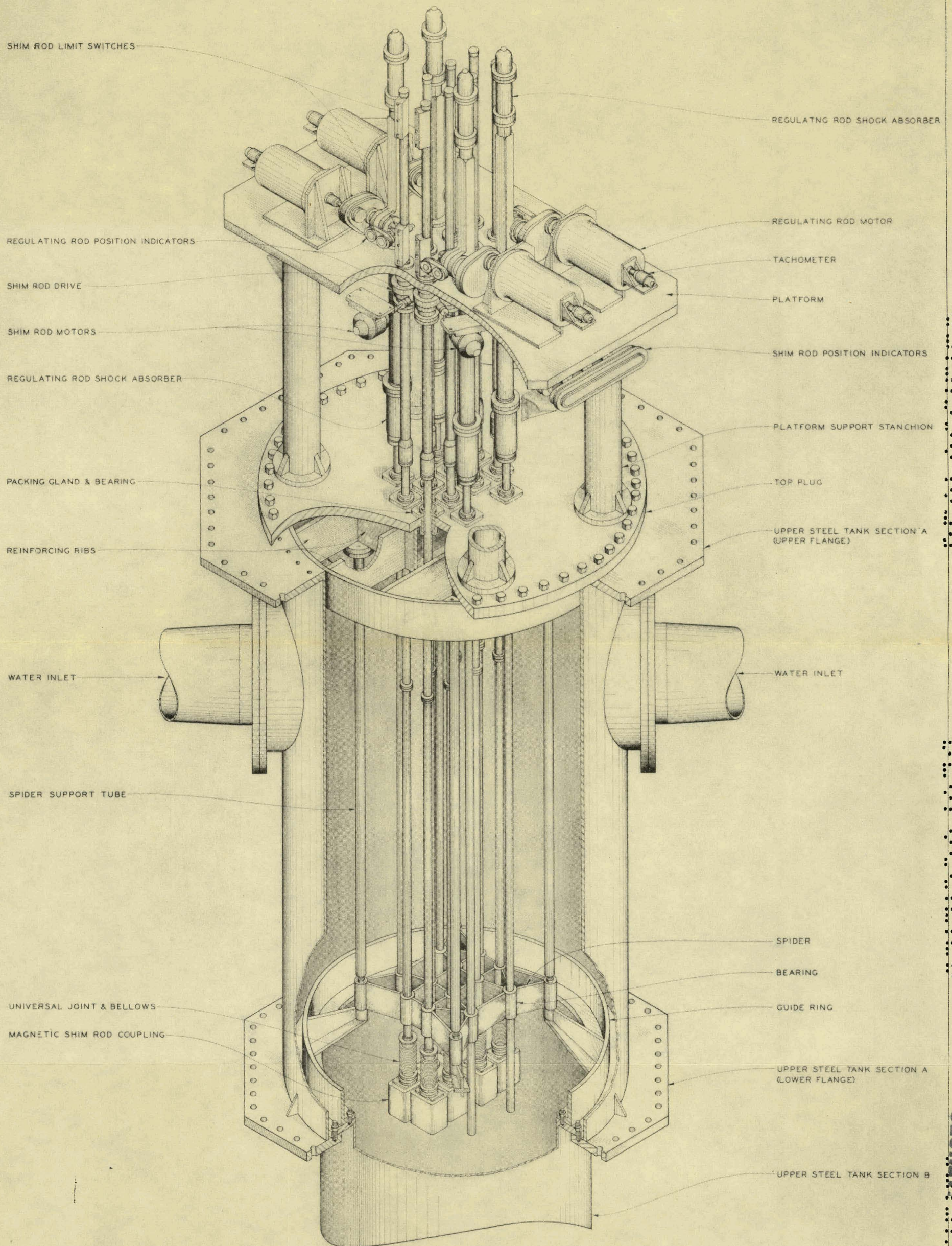


FIG.
DEVELOPMENT REACTOR
CONTROL ROD ARRANGEMENT

FIG 14

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SHIM ROD CONTROL - BLOCK DIAGRAM

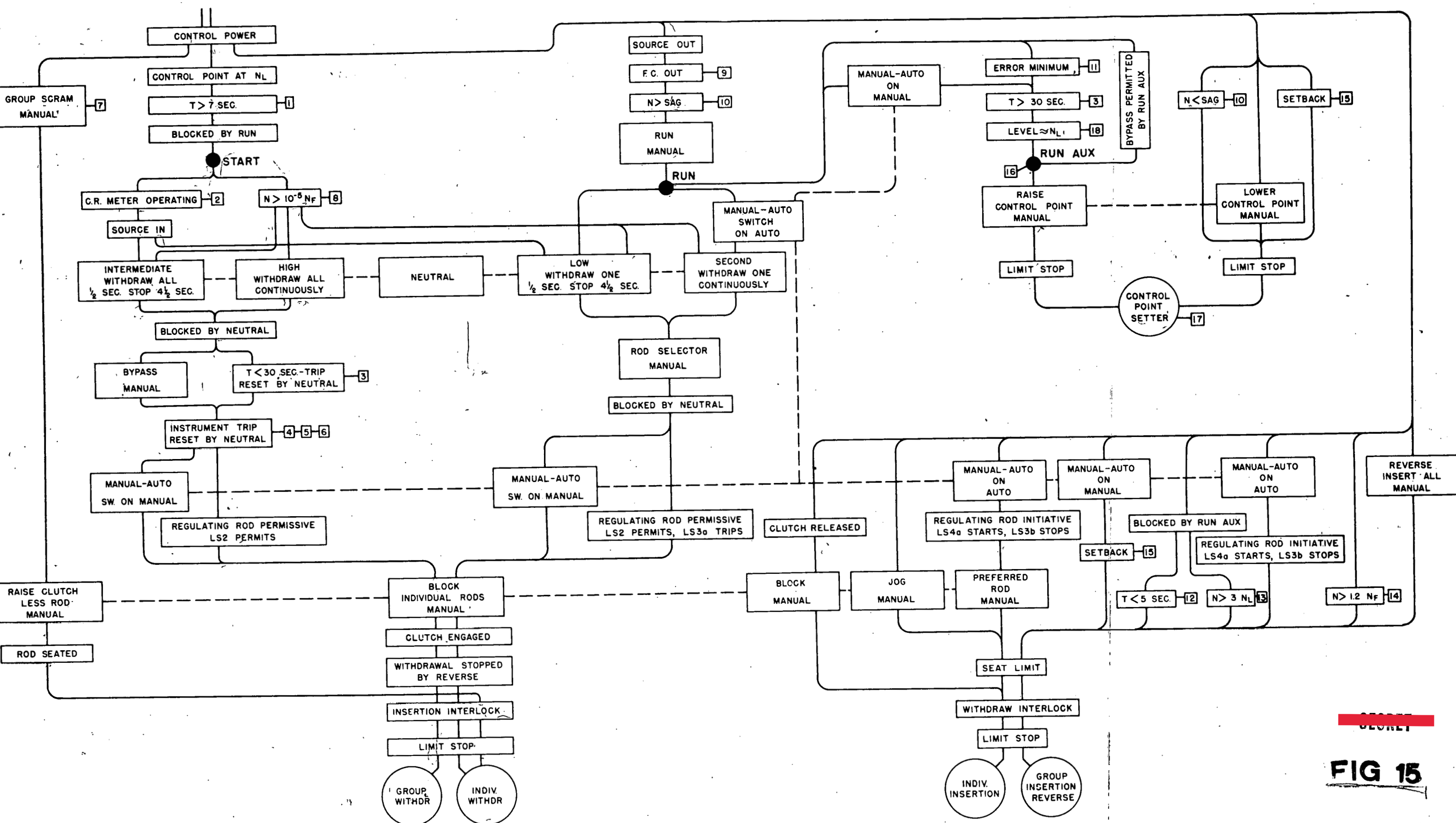


FIG 15

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circuits. The solid lines show the "flow" of control power which eventually results in shim-safety rod motion. Rectangles represent conditions to be satisfied in order for power to reach the objectives represented by the circles. The black dots denote the two main regimes, START and RUN, of withdrawal operation. Dotted lines mark points of inter-connection with instrument signals, mostly from the ion chamber circuits, see Figure 12.

It should be noted that the objective of raising or lowering the control point setting for the servo is not, strictly speaking, part of shim-safety rod control. It is convenient to include it in Figures 15 and 16 because of the close and complicated relation between it and shim-safety rod control. The bulk of Fig. 15 is concerned with functions 1 and 2 of Figure 13. The third function, providing for maintenance, appears in the path at the extreme left of Figure 15. The purpose here is to provide direct means for manual withdrawal of drive rods. Since this conflicts with safety if the drive actually has a rod moving with it, the manual raise switch is put in series with a group scram manual switch so that raising will result only if the magnets are de-energized. Need for withdrawal of the drive rods applies to dummy rods during runs, and to all rods during interrun servicing.

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Consideration of Fig. 15 shows that the objectives of rod insertion can be reached by any of several simple parallel paths, while the objectives of rod withdrawal can be reached only via compliance with more numerous conditions. This is, of course, a reflection of the fact that rod insertion is a safe operation, whereas withdrawal might be dangerous in particular cases.

The paths leading to withdrawal are organized according to the speed to be made available. Since a-c current induction motors are preferred because they are not subject to accidental overspeeding, the most convenient means of getting a wide range of withdrawal speeds is by various degrees of intermittency. This leads to the four speeds previously mentioned -- low, second, intermediate, and high -- which, together with the neutral position, may be selected by a single speed shifter switch handle, analogous to an automobile gear shift. Actually, withdrawal of any individual rod or rods may be manually blocked, which gives the operator effectively an additional range of speeds between low and intermediate, and between second and high.

The permission for high speed withdrawal is withheld after the level has reached one per cent of the full operating level, N_f . This leads to the START--RUN distinction, where START applies to operation at levels up to N_L and RUN to operation from N_L to N_f .

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N_L is about one per cent of N_f . Much of the complexity of the withdrawal paths is due to the transition from START to RUN during start-up, and the reversion to START from RUN when the pile is accidentally or intentionally allowed to "sag" well below N_L .

Fig. 15 shows that START is available subject to proper control point setting, to sufficiently long pile period and to blocking by RUN (so that one is not simultaneously in both START and RUN). The RUN condition is obtained by satisfying conditions of the neutron source, position of the fission chamber, neutron level, and operator manual initiative. The condition on neutron level is governed by a "sag relay", which operates at $0.1N_L$ or $10^{-3} N_f$.

Intermediate and low speeds are available at arbitrarily low levels, subject only to the neutron source being in position* and the operation (i.e., stable visible reading) of the counting rate meter (CRM) and galvanometer, which are the only indicators effective at very low levels. High speed becomes available only at the higher levels. This means that, in the start-up of an unpoisoned

* An antimony source will be installed for the initial start-up. After the reactor is operated for an appreciable time the γ, n reaction in the beryllium will maintain the neutron level at $10^{-8}N_f$ minimum which will give reliable indication on the counting rate meter and galvanometer.

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pile, high will play only a small part, but in start-up from a shut-down immediately following prolonged operation, high speed is needed to win the race against poison growth, and in this case the level would not have dropped below $10^{-5}N_p$.

If the pile rises with a period less than 30 seconds, the withdrawal will be interrupted, and the operator will have to switch back to neutral before resuming it. This interruption can be by-passed, however, by a spring return manual switch. This means that the by-pass cannot be used unless the operator is at the controls with his hand on the by-pass switch.

Another switch is the manual-automatic switch, provided for changing from one servo regulating rod combination to the other and for manual operation in case both servos fail. Thus, this switch by-passes the permissive function of the regulating rod limit switch LS2, which requires that, while operation is servo-governed, withdrawal of shim-safety rods is permitted only if the regulating rod is practically fully withdrawn.

In RUN the use of low and second is subject to a similar regulating rod action, involving in addition to LS2 the limit switch LS3a so that shim-safety rod withdrawal permission is obtained at LS2 and is retained until LS3a is reached.

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Of the remaining conditions on withdrawal, it is perhaps only necessary to explain that "withdrawal stopped by REVERSE" indicates the overriding, interrupting action of a REVERSE signal; while "insertion interlock" merely indicates that any given rod is not to be simultaneously subject to withdrawal and insertion.

Eight parallel paths leading to shim rod insertion are shown in the lower right corner of Fig. 15. Of these, only the one farthest left is free from the condition that the rod must not be seated (i.e., fully inserted); this path provides for automatic run-in of the drive rod after a rod has been dropped, unless the insertion is blocked by the manual switch (as it might be during inter-run or dummy rod maneuvering).

All the other paths are effective only if the rod is off the seat limit switch. Otherwise, jog (the insertion of individual rods at will) is always available. So is group insertion via the manual REVERSE or via the safety circuits on a period of less than 5 seconds, or a level greater than $3 N_L$.

With the manual-automatic switch on automatic, i.e., with servo-governed operation, insertion of a single preferred rod is initiated by limit switch LS4A on the regulating rod. This means that the regulating rod is nearing full insertion and in order to keep in range of control is "calling for" shim-safety rod insertion. As a consequence of shim-safety rod insertion, the regulating rod

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then withdraws until it operates LS3b, stopping the shim-safety rod insertion. A similar group insertion is initiated by LS4b, placed somewhat farther in on the regulating rod's travel. Thus LS4b backs up LS4a in cases where insertion of a single shim rod does not satisfy the demands of the regulating rod.

The usual function of a setback signal is to lower the servo control point. However, if the manual-automatic switch is on manual, setback initiates a group insertion, since in this case its usual function would be without significance. Setback is a mild safety function, set to operate at about 1.1 N_f or on emergency signals from the cooling system.

The control point setter is a motor-operated rheostat which determines the control setting of the servo. There are three ways a lowering action can be initiated: manually, by the sag relay, and by the setback relay. The latter two are complete, i.e., the rheostat keeps running down until it reaches the limit stop.

Raising of the control point is subject to achieving the RUN AUXILIARY condition. This requires an "error minimum" condition showing good servo operation (unless the manual-automatic switch is on manual), a period greater than 30 seconds and a level near N_L . These conditions are not permanent, however, as they are by-passed once RUN AUXILIARY is established. Manual raising of the control point is then permitted. The general scheme set forth in Fig. 15

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is carried out in terms of relays and switches in the elementary diagram Fig. 16.

As described previously, sensitivity of control by the shim-safety rods is selected at will by the operator in accordance with the power level needs. However, no mention has been made of the variation of sensitivity with position of the shim-safety rods in the active lattice.

The effectiveness of the shim-safety rods in controlling the reactor is proportional to the neutron density of the volume of the pile in which the rod is located.

The neutron density in the active lattice varies from the top to the bottom approximately as a cosine curve. As a result the shim-safety rods exercise minimum control when they are almost out of the active lattice and maximum control when they move into the midsection of the active lattice.

Under certain conditions with the shim-safety rods entirely removed from the active lattice or with one shim-safety rod completely inserted and the others removed, a condition of minimum control or zero control sensitivity might occur. However, any critical condition which might occur because of minimum or zero control sensitivity is effectively eliminated by proper location of limit switches and controls. A sufficient number of shim-safety

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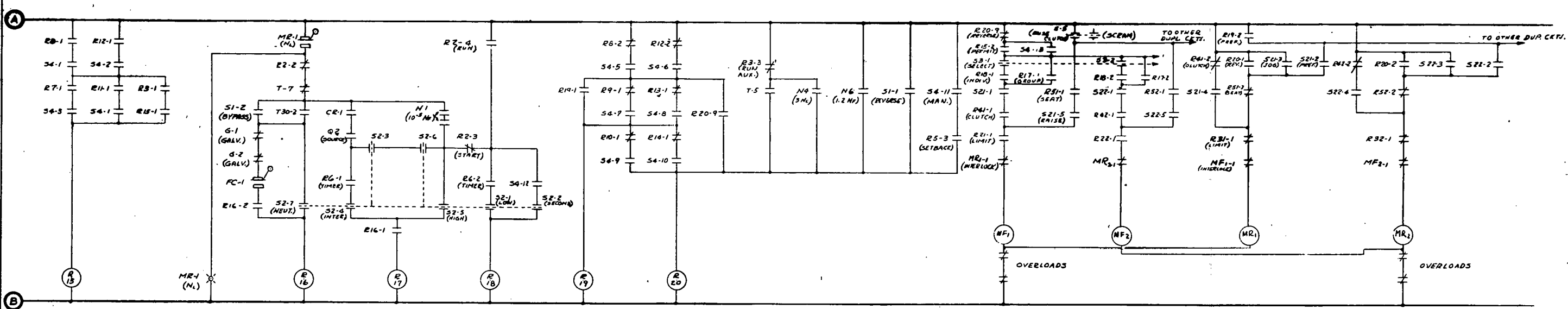
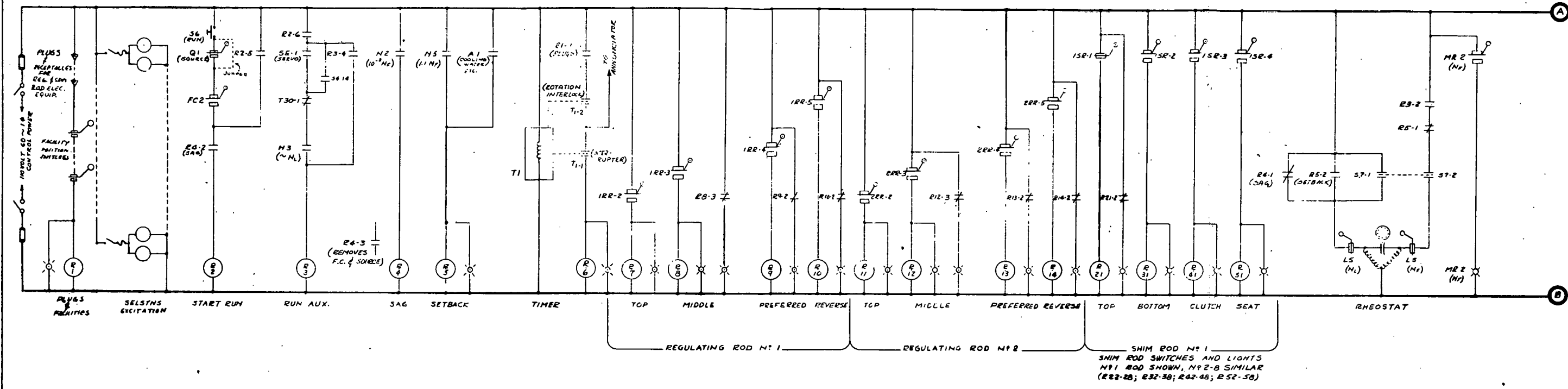
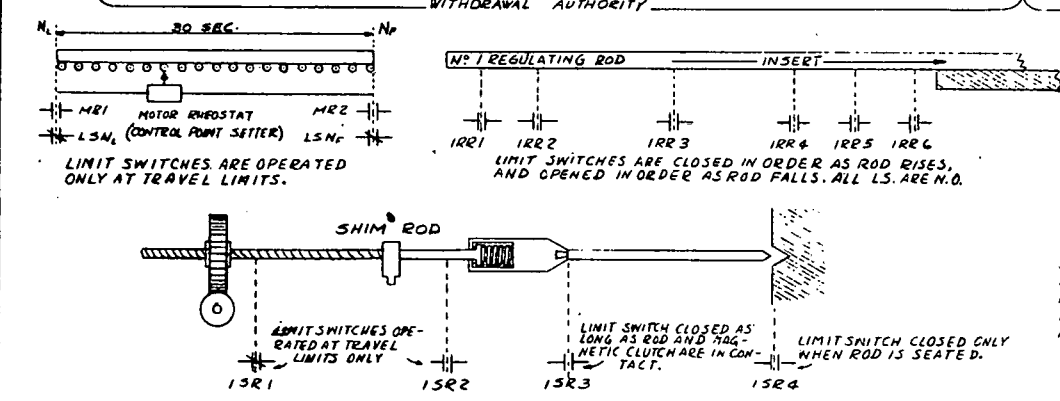


FIG 16



5-1 REVERSE AND BY-PASS

CONTACT	POSITIONS
1	REV. NORM. STAY
2	REV. NORM. STAY
3	REV. NORM. STAY

5-2 SPEED SHIFTER TWO SWITCHES BACK TO BACK

CONTACT	POSITIONS
1	REV. NORM. STAY
2	REV. NORM. STAY
3	REV. NORM. STAY

5-3 MANUAL - AUTO - MATIC

CONTACT	POSITIONS
1	REV. NORM. STAY
2	REV. NORM. STAY
3	REV. NORM. STAY

5-4 INDIVIDUAL ROD SWITCH

CONTACT	POSITIONS
1	REV. NORM. STAY
2	REV. NORM. STAY
3	REV. NORM. STAY

MASTER SWITCH CONTACTS ARE CLOSED ONLY IN POSITION MARKED 'X'. RELAY CONTACT POSITIONS SHOWN ARE FOR DE ENERGIZED RELAY COILS. LIMIT SWITCHES SHOWN IN NORMAL POSITION (NOT OPERATED).

NOTICE: DRAWING BASED ON PRESENTLY AVAILABLE DATA. APPLICABLE IMPROVEMENTS OR OPERATIONAL CHANGES WILL BE MADE AS DEVELOPED.

DATE	BY	DATE
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CHEMICAL PLANTS DIVISION
BLAW-KNOX CONSTRUCTION COMPANY
PITTSBURGH, PA.

FOR
UNITED STATES ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS
MATERIALS TESTING REACTOR
ELEMENTARY DIAGRAM
SHIM ROD CONTROL

CONTRACT NO. BKC-3150 3150-603-1

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rods are kept in the active lattice at all times to assure prompt control of neutron production in the pile.

3. Magnet Release Safety Circuits.

The philosophy upon which this system is based is that the most carefully made system will eventually fail. For this reason the circuit includes a considerable number of relays whose sole function is to provide a warning of abnormal operation. These relays provide a warning light on the unit containing them, and the warnings are collected into a single pair of warning lights on the main control panel of the reactor. The safety circuit is expected to trip all seven of the safety magnets from any one of three classes of input signals (seven or more total input channels, see Fig. 13). These classes of input signals are: (1) High neutron intensity level (triple), (2) high rate of increase of reactor power (short period), (3) scram warning from temperature, water flow, or other auxiliary source. Fig. 17 shows a block diagram of the system.

No relays, thyratrons or other type of completely off-on controls are used in the course of the signal from ion chambers to magnet current. The reason for this is that off-on devices do not provide any warning of failure. With straight electronic control throughout, a continuous report of the condition of the circuit is provided by under and over current or voltage relays which do not take part in the normal operation of the circuit. Alternatively, a monitor or pilot signal

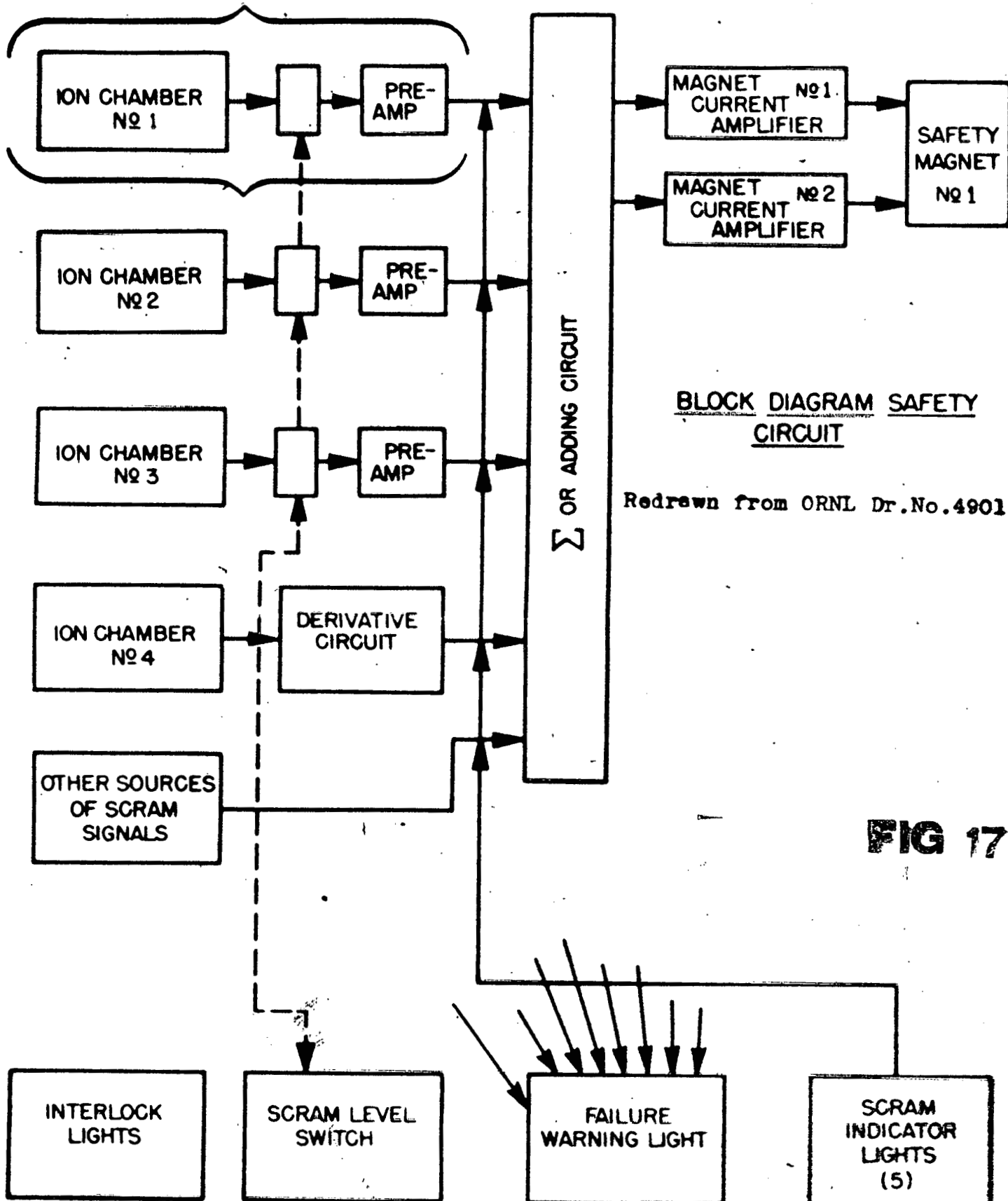
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Fig. 14

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CHANNEL No 1



BLOCK DIAGRAM SAFETY
CIRCUIT

Redrawn from ORNL Dr.No.4901

FIG 17

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may be sent through the system and change or failure of this signal reported. These systems do not make failure less probable, but they do report it when it occurs so that it may be corrected before operation is required.

Fig. 18 shows a simplified diagram of channel #1. The leads for warning lights operated by relays, test points for protective maintenance and interlock switches are omitted. The adding circuit is a necessity if any one of the signal channels is to operate all of the magnets. This is the most vulnerable point of the system and has been reduced to a single wire joining the units. Any short to ground or open in this lead provides a warning signal. Shorts in the tubes or wiring of one or more of the magnet current amplifiers will not prevent the application of a scram signal to other magnet current amplifiers. There are two magnet current amplifiers complete with power supplies for each magnet. This permits the removal of one amplifier for replacement or servicing without dropping a rod.

4. Operation

The operation of the reactor is described in Section V,B. of this report.

G. Effluent Control

There are three classes of radioactive effluent to be considered at the Materials Testing Reactor site. These are: process water purged from the reactor primary cooling water and experimental

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Redrawn from CRNL Dr 4902

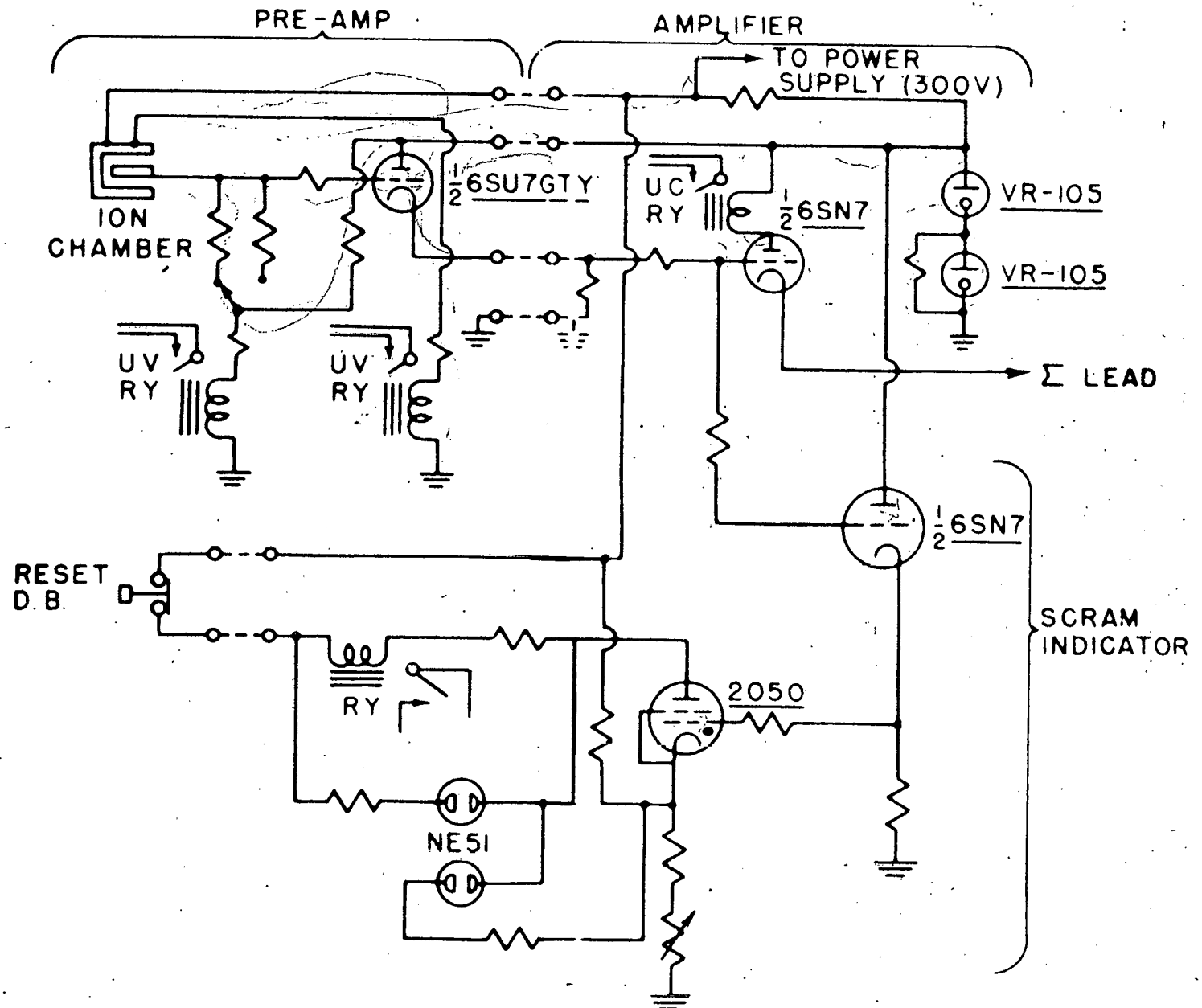


FIG. 18

CHANNEL 1

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plug cooling water system; drainage from floor drains, and "warm" drains from laboratory sinks; and "hot" chemicals from the laboratories.

1. Process Water

The process water will be purged at 50 gpm by adding fresh demineralized water with overflow from the sump tank to the retention basin. The water will be held from 72 to 96 hours in the retention basin before discharge to the leaching beds.

The water will have an estimated activity of .5 microcuries per liter of Na^{24} at the time of discharge to the leaching beds. While the Health Physics Conference at Chalk River in the Fall of 1949, agreed on a permissible level of 5 microcuries per liter of Na^{24} at the time of discharge, the Argonne Health Physics Division favor the lower value and this has been used in design.

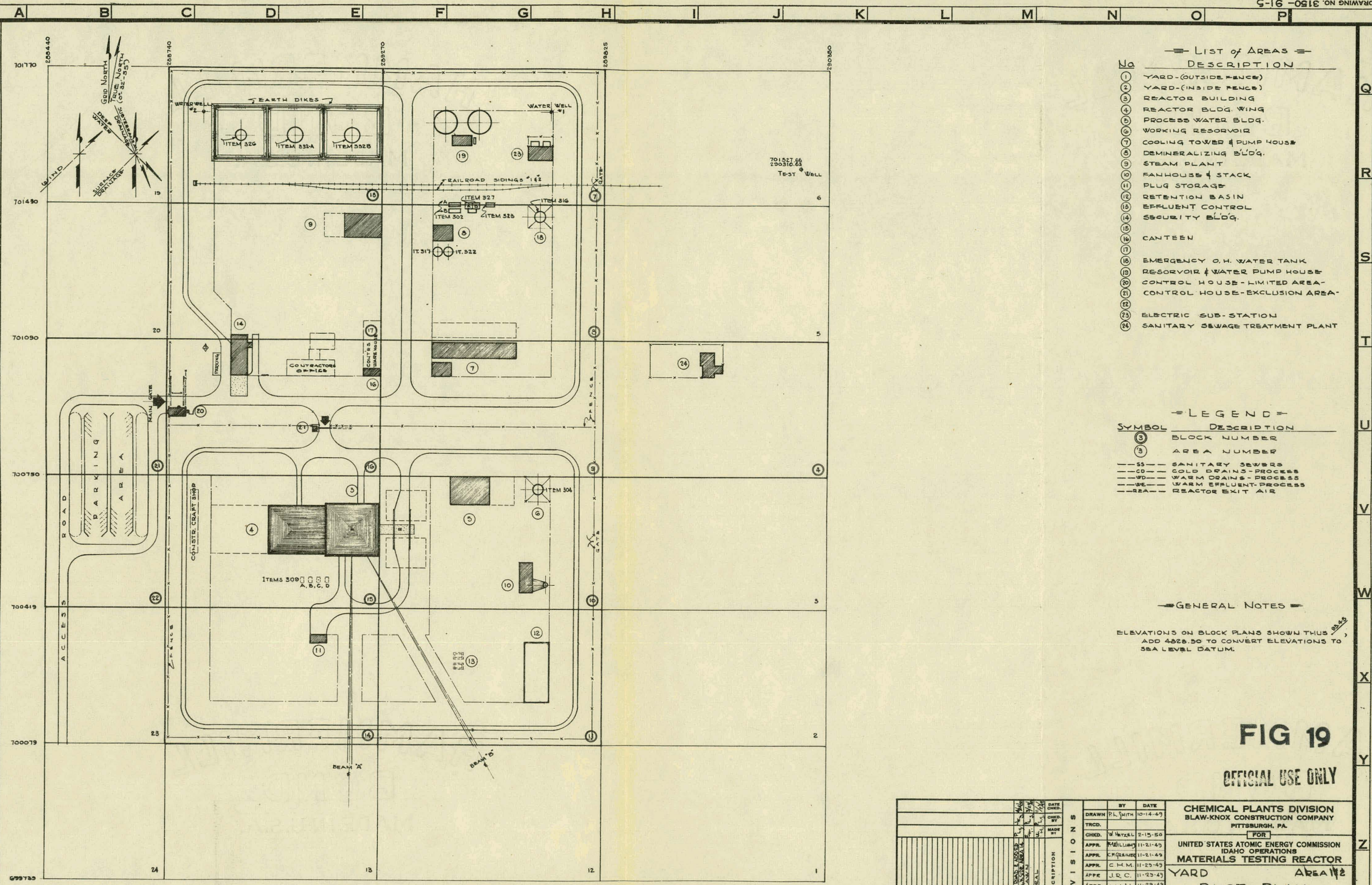
The water at time of discharge will have an estimated activity of 1×10^{-4} microcuries per cc of P^{32} compared to the permissible activity for P^{32} of 2×10^{-4} microcuries per cc. Since the half-life is 14.1 days, the amount of decay during the 72 hour holdup is negligible.

Figure 19, BK-3150-91-5, is a plot plan of the reactor area. This plan shows the relative positions of the retention basin and leaching beds.

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- LIST OF AREAS —
- | No | DESCRIPTION |
|----|---------------------------------|
| 1 | YARD-(OUTSIDE FENCE) |
| 2 | YARD-(INSIDE FENCE) |
| 3 | REACTOR BUILDING |
| 4 | REACTOR BLDG. WING |
| 5 | PROCESS WATER BLDG. |
| 6 | WORKING RESERVOIR |
| 7 | COOLING TOWER & PUMP HOUSE |
| 8 | DEMINERALIZING BLDG. |
| 9 | STEAM PLANT |
| 10 | FANHOUSE & STACK |
| 11 | PLUG STORAGE |
| 12 | RETENTION BASIN |
| 13 | EFFLUENT CONTROL |
| 14 | SECURITY BLDG. |
| 15 | CANTEEN |
| 16 | EMERGENCY O.H. WATER TANK |
| 17 | RESERVOIR & WATER PUMP HOUSE |
| 18 | CONTROL HOUSE - LIMITED AREA |
| 19 | CONTROL HOUSE - EXCLUSION AREA |
| 20 | ELECTRIC SUB-STATION |
| 21 | SANITARY SEWAGE TREATMENT PLANT |

- LEGEND —
- | SYMBOL | DESCRIPTION |
|--------|-------------------------|
| ③ | BLOCK NUMBER |
| ③ | AREA NUMBER |
| SS | SANITARY SEWERS |
| CD | COLD DRAINS - PROCESS |
| WD | WARM DRAINS - PROCESS |
| WE | WARM EFFLUENT - PROCESS |
| REA | REACTOR EXIT AIR |

— GENERAL NOTES —

ELEVATIONS ON BLOCK PLANS SHOWN THUS ^{92.5},
ADD 4828.50 TO CONVERT ELEVATIONS TO
SEA LEVEL DATUM.

FIG 19
OFFICIAL USE ONLY

DRAWN P.L. SMITH 10-14-49		BY DATE		CHEMICAL PLANTS DIVISION	
TRCD.		10-14-49		BLAW-KNOX CONSTRUCTION COMPANY	
CHKD. W. HETZEL 2-15-50		10-14-49		PITTSBURGH, PA.	
APPR. J. WILLIAMS 11-21-49		11-21-49		FOR	
APPR. C.F. QUINER 11-21-49		11-21-49		UNITED STATES ATOMIC ENERGY COMMISSION	
APPR. C.H.M. 11-25-49		11-25-49		IDAHO OPERATIONS	
APPR. J.R.C. 11-25-49		11-25-49		MATERIALS TESTING REACTOR	
APPR. W.W.J. 11-29-49		11-29-49		YARD AREA 12	
APPR. G.E.K. 12-18-49		12-18-49		PLOT PLAN	
APPR. CUSTOMER 12-18-49		12-18-49		CONTRACT NO.	
SCALE 1"=100'		12-18-49		BKC-3150 3150-91-5	

2. Warm Drains

Drainage from floor drains in the reactor building, the laboratory warm drains, etc., will drain to 1500 gallon glass-lined catch tanks. These tanks will be equipped with mixing pumps and sampling outlets. At intervals, dependent on the rate of effluent accumulation, samples will be taken from these tanks and monitored to determine the activity. When this effluent activity is found to be of other than short-lived isotopes, the contents will be pumped to one of two 10,000 gallon underground storage tanks. Effluent having short-lived activity will be pumped to the retention basin. A laboratory log system will be initiated to facilitate and augment the counting procedure, i.e., the log system will tell the monitor personnel what to look for in "counting" the sample. However, the laboratory log alone will not be depended upon. A complete counting routine will be followed to assure the absence of long-lived isotopes before the effluent in a catch tank is pumped to the retention basin. The effluent from the vapor scrubber, installed in the exhaust vent line from "warm" hoods, will go to catch tanks and the disposal procedure will be the same as that just mentioned.

The activity in each of the 10,000 gallon storage tanks will be limited to 10 curies. More tanks will be installed as needed or the liquids stored in the tanks will be shipped to the chemical waste processing plant for concentration and storage. The catch tanks and the 10,000 gallon storage tanks will be buried over concrete pads that will drain to sample pits so that leaks may be detected.

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3. Hot Wastes

Highly radioactive liquid wastes from the laboratories will be poured into glass-lined 9 liter shielded cans. The cans containing only short-lived activity will be stored and then monitored at a time when the activity should be below tolerance. If it checks out below tolerance, the solutions will be dumped just as any other nonactive waste. Those cans containing long-lived activity will be emptied into one of the 10,000 gallon underground storage tanks or shipped to some other area for storage.

4. Solid Radioactive Wastes

Solid radioactive wastes will be buried in a fenced in area removed from the general work areas but still within the restricted area fence, see Figure 19, Plot Plan drawing BKC-3150-91-5.

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V. PLAN OF REACTOR OPERATION

The Materials Testing Reactor is essentially a research and development tool. Both the design and method of operation are based on the premise that the facility will be made available to carry out a very varied program of research and development problems. Therefore, the operational procedures will of necessity be subject to alteration from time to time to fit the experimental program.

This section of the report gives the general method of operation of the reactor without regard to the experimental work.

A. Handling of Fuel Assemblies

The fuel assemblies will be delivered to the Materials Testing Reactor ready for use. On receipt the assemblies will be stored in a vault in the Reactor Building. Before use, the assemblies will be inspected to make certain they are within the drawing specifications. Suitable jigs and gauges are being developed for this purpose.

The following description of the unloading and loading of the reactor assumes that the reactor has been in operation and is to be recharged. The operation of the reactor itself will be discussed later.

Normally during shutdown 1000 gpm of process water is circulated through the reactor to remove the heat of radioactive decay

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of the fission products and the reactor components, and about 350 lbs/min of air is drawn through the graphite. At shutdown, the shim rods are all the way in with the absorbing sections of the rods in the active lattice of the reactor.

1. The first step in the recharging operations is to purge the water system within the reactor with non-radioactive water. This is done by closing the process water line completely and running 1000 gpm of demineralized water into the reactor. Within twenty minutes the water will have been completely purged and the top plug of the reactor can be removed. During this purging operation and continuously while the top plug is off, demineralized water at 1000 gpm is run through the reactor to the seal tank, with overflow to the retention basin. The seal tank weir will maintain the level near the top of the reactor. When the entire system has been purged, the water will be recirculated. After the top plug has been removed the flow rate will be adjusted to a rate that will not cause turbulence and yet will be high enough to remove the radioactive decay heat.

2. The second step is the removal of the top plug. This is done by removing the nuts from the bolts holding the plug in place, disconnecting the shim rods by de-energizing the shim-rod magnets, raising the shim-rod magnets, disconnecting the electrical connections to the top plug, and lifting the plug out by the crane. The plug is stored in a hole in the main floor of the reactor building. The

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magnets hang through to the basement and can be inspected and repaired if necessary. Since the magnets will have an activity of about two curies, they will be either shielded or heavy screens will be placed at an adequate distance from the magnets to protect personnel.

3. With the top plug removed the water has sufficient depth, 19' 6", over the top of the active section of the reactor to permit handling of the fuel assemblies. The third step is the removal of the top grid from off the fuel assemblies. This top grid is stored within the reactor just above the reflector while the fuel assemblies are being changed. After removal of the grids the beryllium section over the discharge chute is removed and stored above the reflector. A plug in the discharge chute below the beryllium section is also removed and stored. The grids, beryllium section, discharge chute plug and fuel assemblies are all handled by light tools moved vertically by a 1000 pound capacity crane above the top of the reactor.

The removal of the plug from the discharge chute exposes the remainder of the discharge chute. This chute is a six inch ID pipe leading to the canal through the sub-pile room. In the sub-pile room there will be a hydraulic operated valve in this chute.

The canal will contain a receiver mechanism consisting of a rotating hydraulic cylinder which will seal the bottom of the discharge chute and at the same time receive the discharged fuel assemblies. Operation of this cylinder is controlled from the basement.

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When this receiver is in contact with the discharge chute, fresh demineralized water is forced up the tube into the reactor to purge the discharge chute.

4. A fuel assembly is discharged through the following steps:
 - a. The hydraulic cylinder in the canal is rotated to the vertical position and sealed to the discharge chute.
 - b. The valve in the discharge chute is opened hydraulically and fresh water forced up the discharge chute.
 - c. The discharge tool is connected and locked to the top of the fuel assembly to be discharged; the fuel assembly is lifted out of position and lowered down the discharge chute; when the assembly is in the bottom of the rotating hydraulic cylinder of the receiver in the canal, the tool is disconnected from the fuel assembly and removed.
 - d. The valve in the discharge chute is closed.
 - e. The rotating hydraulic cylinder in the basement is disconnected from the discharge chute, rotated to a horizontal position and the fuel assembly ejected hydraulically. The fuel assembly is then handled and placed in storage by hand tools.
 - f. Meanwhile a fresh fuel assembly is placed in the space vacated by the burned out fuel assembly by lowering the new assembly from the top of the reactor.
 - g. The shim-rods and removable beryllium sections are handled in identical manner with the fuel assemblies.

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5. After all the fuel has been renewed, the stopper and beryllium plug are replaced in the discharge chute and the grids replaced.

6. The top plug is replaced, the connections made and checked, and the shim rods connected by energizing the magnets.

7. The fresh water is shut off and the regular process water circulated through the reactor.

B. Operation of the Reactor

1. Start-up

The initial step in operating the reactor is to start the auxiliary facilities and bring them into operation. This is done by shutting off the circulation of the 1,000 gpm water pump and starting the pumps that will deliver 20,000 gpm to the process water storage tank. At the same time the vacuum jets on the vacuum evaporators for cooling the process water are started. The air system blowers are started and 1850 lbs/min of air is drawn through the graphite in the reactor. Water is run through the experimental plugs to cool them.

When all the above utilities are working satisfactorily, the reactor itself is started. In the Oak Ridge National Laboratory Physics Division Quarterly Report CNL-35, Dr. H. W. Newson has given an excellent description of the startup and operation of the reactor.

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The following paragraphs are excerpts from his description.

"Startup must be divided into two cases depending on whether the intensity is in the counter range or in the period range. Taking up the former case first, when the reactor is first started or when all of the active lattice has been removed for reprocessing, and replaced with fresh assemblies, the neutron intensity will probably be too low for operation of the period circuits. This will make high speed unavailable. However, a proper source will be present and a definite reading will be obtained on the logarithmic counting rate meter (CRM). This intensity will make intermediate speed possible. The operator should first test the circuit by moving the fission chamber and observing the change in the reading. He then selects intermediate speed. The rods will move out intermittently, one-half second movement followed by a four and one-half second pause. The logarithmic scale will show an increase of only a few millimeters during the first minute of rod motion. However, when the pile is approaching critical 20 or 30 minutes later and at about 1000 times the initial intensity, the galvanometer spot will show definite movements during the motion of the rod and much slower movements during the dead time. Just below critical the quick movement of the galvanometer spot will be 5-10 mm. Just beyond critical the spot will show about an equal displacement during the motionless period of the shim rods. This will be the operator's signal that he is slightly above critical. At this point he should shift to low speed.

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When the displacement per time interval becomes 1-2 cm the period meter should show a period of 10-15 seconds. Having verified the reading of the period meters, he should increase k until the reactor is rising with a stable period of about 7 seconds. This must be done by successive approximations. The $4\frac{1}{2}$ seconds dead time of the rods in low and intermediate is about long enough to allow the transient part of the apparent period to die out between successive additions of k. Ordinarily the period meter will make its presence felt by tripping the latch on intermediate speed at about the time the pile passes critical. In unlikely cases, the reactor may go over critical at an intensity too low to operate the period meter. Until the period meter shows signs of life, the operator should proceed cautiously. However, if the operator misreads the signs and puts the reactor on too short a period, nothing worse will happen than a reversal of the shim-safety rods for a fraction of a second. By observing the size of the quick jumps below critical the operator should be able to determine the k of the reactor fairly accurately after he has had a little practice. In general, it may be said that, in this kind of startup, the period meter will come into play to correct an operator's mistake although perhaps not soon enough to warn him in advance.

"If the active lattice in the reactor has been operated at full power for a day or more the intensity is not likely to fall below the period control range even through the reactor has been shut down for

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several days. In this case the intensity will be sufficient to permit the use of high speed. The interlocks will require that the logarithmic amplifiers of the period meters are operating and indicate a minimum power in the neighborhood of $10^{-5}N_F^*$. (A considerably lower level may be permissible after a trial period). The operator will put his speed selector into high and note carefully the reading at the time the rods start moving. The reactor will become critical about a factor of 100 above this intensity. The exact factor will be determined from experience. When the reactor has risen a factor high enough to be nearly critical, the operator shifts from high to intermediate and observes the small jumps of the galvanometer spot. He may then shift back to high if his observation indicates that the pile is still decidedly subcritical or remain in intermediate if the pile is approximately critical. If critical is approached in high, a period of 30 seconds will be reached just below critical and the operator will be warned by the tripping of the latch on the shim rod motors. The action of this trip should be anticipated by the operator who should shift into intermediate before it is actuated. When the reactor is definitely over critical the operator should shift into low and adjust the period to 7 seconds as before. A skillful operator should achieve a startup in such a way that the 7 seconds opening of the shim rod motor circuit and the 5 seconds reversal will never be actuated.

* N_F is the neutron intensity at full power.

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Frequent dependence on these automatic aids will be evidence of careless manipulation. It is felt that this combination of manual initiative under the supervision of the period circuits gives considerably greater safety since either the operator or the circuits alone should be sufficient to prevent a start-up accident.

"After the reactor has reached a stable 7 second period there should be no further motion of the shim rods. When the pile reaches $1\% N_F$ the servo should bring it to a constant power automatically. If the servo is inoperative, or has been switched off by the operator, the leveling will be performed more crudely by the reversal of the shim rod motors."

2. Run

After the reactor is leveled the operator should check his instruments, particularly for evidence that the ion chambers connected to the safety circuits are showing proper readings and also to be sure that the counting rate meter has been withdrawn within the shield. At this point the operator will shift into "Run" and set the servo control point at the desired power of operation. Final determination of N_F is made by checking the actual power output of the system by means of the temperature rise of the water.

If before startup the reactor was not seriously poisoned by Xenon, the reactor will become critical with the ^{bottoms?} tops of the absorbing sections of the rods somewhere near the middle of the pile.

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During operation Cd rods should be withdrawn to compensate for Xenon growth and the Cd should be further withdrawn and Th inserted to set up a rod configuration of greatest neutron economy. This process will be limited by special limit switches which prevent withdrawal or insertion of rods to zero sensitivity. This is discussed in the section on interlocks.

3. Shutdown

The reactor is shut down by simply turning the selector switch at the control console to "Reverse." This runs all shim-safety rods into the active section to the full absorption position. Auxiliaries are shut down at will except for the 1,000 gpm water pump and the 350 lbs/min air blower which are operated continuously or at least until residual gamma heat is dissipated to a very low level.

4. Scram

Scram or automatic dropping of the shim rods with immediate shutdown occurs whenever the operator pushes the scram signal or whenever any of the various interlocks to the scram circuit operate.

On scram the magnets holding the shim rods are de-energized and the rods fall in less than 30 mil-sec.

C. Operation of Utilities

The process water system, the air cooling system, and the effluent control will be run in accordance with normal operating

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practice for ordinary commercial plants. At least one operator will be present at all times in the process water building, steam plant, and demineralized water plant; and roving operators will inspect the water wells, fan house, and all parts of the effluent control system at frequent intervals.

D. Initial Operation of the Reactor

The members of the Materials Testing Reactor Steering Committee will remain with the project during initial startup and until all parts of the project operate satisfactorily. The design engineers and physicists will be available for consultation and aid to the project as long as their help is needed.

The initial startup should be greatly aided by the experience gained from operation of the mockup critical experiment. The fact that the mock-up has operated satisfactorily as a critical experiment gives us confidence that only minor difficulties will be encountered in startup and operation of the reactor itself.


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VI. EXPERIMENTAL WORK

As noted in the introduction to the last section, the very nature of the Materials Testing Reactor as a research and development tool makes it impossible to outline the experimental work that will be conducted with the aid of the reactor. Rather it is hoped and expected that the reactor will be used for an extremely varied series of research and development experiments. If the reactor is not so used, it will not have fulfilled its basic function.

The experiments and research programs for the reactor normally will be initiated by scientists and engineers who are working in the National Laboratories, universities, and research institutions. Each experiment suggested will presumably have been approved by the originating laboratory. While the organization has not been fixed, we believe all experimental work conducted with the reactor will be under the supervision of a technical committee. This committee will probably be made up of representatives of the Nation Laboratories as well as one or more representatives from the MTR staff. This committee or some other group will review all suggested programs, examine each experiment from the viewpoint of safety and feasibility, and schedule the experiment on a priority basis. As soon as an experiment is approved, a liaison engineer or scientist will work with the experimentalist in designing, setting up and conducting the experiment. It is expected in most cases that the experimentalist


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will design and construct all equipment at his home laboratory. The actual carrying out of the experiment will be under the joint supervision of the experimentalist and the reactor technical personnel.

The operating staff will be expected to conduct any experiment approved by the technical committee that the operators consider safe. Likewise, the operating people will not be asked to run any experiment that they do not consider safe.

In case of disagreement the laboratory directors or their representatives will be asked to express their opinion on any experiment.

A. Experimental Facilities

The chart shown in Figure 20 is a summary of the experimental facilities which will be built into the reactor. The Figures 4, 5, and 8 presented in the first part of this report show the location of most of the main experimental holes and Figure 7 shows a cross sectional view of one of the main beam hole plugs. The main beam hole plugs can only be removed with the reactor shut down and the lead doors must be closed whenever the plugs are being moved away from the holes in their coffins.

The cooling water and experimental facilities will be installed one foot inside the outside of the biological shield. A one foot

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SUMMARY OF EXPERIMENTAL FACILITIES

TYPE OF HOLE	FACILITY DESIGNATION	SIZE, IN.	APPROACH	ESTIMATED FLUX (MAX.)		REMARKS
				THERMAL	FAST*	
Through	VT2	2 3/4 x 2 3/4	Vertical thru shim rod hole	2.5×10^{14}	1×10^{14}	An attempt is being made to provide a return loop for slurry experiments.
Through	HT1	4 7/8 x 4 7/8	Horizontal thru reactor	1.8×10^{14}	3×10^{13}	Next to core, loop thru HG2 for slurry.
Beam	HB1,2,3	6 ID in Be	Horizontal to core edge	5×10^{14}	7×10^{13}	{ One can be 1/4 mile beam, and one for neutron spectrometer not to be installed initially.
Beam	HB4,5,6	6 ID in Be	Horizontal to Be in active sect.	2.8×10^{14}	2×10^{11}	
Beam	HG9	10 x 16	Horizontal to tank wall	8×10^{13}	4×10^{12}	5 x 12 fast neutron channel to core.
Beam	DB1,2,3,4,5,6	6 ID	Inclined down to tank	4×10^{13}	5×10^{10} (DB2 only)	2 ID can channels to core.
Thermal	T2	6 ft. x 6 ft.	Horizontal to Pb window	1×10^{12} Max.		Thermal column.
Thermal	T2H1,2	4 x 4	Horizontal across T2	2×10^{10}		East-west thru structure.
Thermal	T2H3,4,5,6,7,8	4 x 4	Horizontal parallel to T2	6×10^{11}		Within T2, to the Pb window.
Thermal	T2V1,2	12 x 12	Vertical from top of pile	2×10^{11}		Extend to bottom of T2.
Pneumatic Rabbit	HR1,2	1 ID	Horizontal thru reactor	1.6×10^{14}	4×10^{13}	Next to core, air operated.
Pneumatic Rabbit	HR3,4	4 ID	Horizontal to reactor tank	8×10^{13}		Pneumatically operated.
Hydraulic Rabbit	VH1,2	1 ID	Vertical from basement	2.4×10^{14}		In Be, hydraulically operated.
Hydraulic Rabbit	VH3,4	1 1/2 ID	Vertical from top	2.5×10^{14}	10^{13}	In Be, spare regulating rod holes.
Instrument	WN1,2,3,4,5,6	2 1/2 ID	Slant from top	5×10^{13}		In graphite pebble zone.
Through	HI2,3	10 3/4 ID	Horizontal thru reactor	10^{13}		North-south thru graphite.
Through	HG1,2	8 ID	Horizontal thru reactor	10^{13}		East-west in graphite.
Through	HG3,4	4 ID	Horizontal thru reactor	10^{13}		East-west in graphite.
Beam	HG5,6	6 ID	Horizontal to tank wall	8×10^{13}		Elevation mid-height of core.
Isotope	VG1,3,5,6	4 ID	Vertical from top	3×10^{13}		In graphite pebble zone.
Isotope	VG2,4	4 ID	Vertical thru reactor	3×10^{13}		In graphite, discharge to canal.
Access	VG9	10 x 10	Vertical from top	3×10^{13}		Access to HG9.
Isotope	VG7,8,10-20	2 1/2 ID	Slant from top	7×10^{13}		Terminate in graphite pebble zone.
Isotope	VG23-40,42-45 47-63	2 ID	Vertical from top	$10^{13} - 10^{12}$		In graphite zone.
Gamma	GT1,2	6 ID	Vertical from top	500r/hour		Gamma thimble in water.
Fast Flux	~40 Be Ass'ys.	3 x 3	Vertical from top tank**	5×10^{14} Max.	7×10^{13} Max.	Removable reflector pieces.
Fast Flux	~15 Be Ass'ys.	3 x 3	Vertical from top tank**	5×10^{14} Max.	7×10^{13} Max.	Removable Be in core.
Shielding	Facility		Horizontal from North	10^{12}		Special handling equipment and converter plate to be installed when needed.

* Neutrons having energies > 1 mev.

** Charged or discharged only when reactor is shut down and top plug is removed.

Note: These estimated neutron flux values are based on beryllium or graphite (according to material penetrated) in the experimental holes. Actually experimental apparatus and samples will cause local "sinks" and reduce the flux attainable for a particular experiment.

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concrete plug will be placed outside of these facilities for shielding from the cooling water.

Before installation within the reactor each experimental plug will be checked dimensionally and for handling in mock setups.

B. Types of Experiment

A great many experiments have already been suggested by various scientists. Some twenty different types are listed below:

1. Isotope manufacturing, P^{32} , Na^{24} , Ca^{45} , etc.
2. Exposures of tissue and animals.
3. Functional tests of fuel assemblies for future reactors.
4. Shielding materials.
5. Slurry experiments.
6. Effects of irradiation on reactor materials.
Wigner disease.
7. Neutron crystal spectrometer studies.
8. Measurement of fission product poisoning on thermal reactors.
9. Second order capture studies.
10. Military isotopes.
11. Neutron diffusion studies.
12. General knowledge of solid state.
13. Effect of irradiation on creep of metals, tensile strength, impact strength, corrosion.
14. Tests of canning and cladding of reactor materials.
15. Diffusion of fission products

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16. Effect of irradiation of metallic pile coolants.
17. Dissociation of homogeneous pile materials.
18. Studies of irradiation damage of plastics, lubricants.
19. Induced activities of engineering materials.
20. Shielding survey of the reactor itself.

C. Limitations on Experiments

Very few limitations will be placed on the types of experiments that may be run in the reactor. However, every effort will be made by the technical committee and operating personnel to assure themselves that every experiment is safe in every respect. The real assurance is that senior scientists and engineers will be asked to approve experiments in respect to safety. This means that the emphasis will be placed on the examination of every experiment and the manner in which it is to be conducted rather than the exclusion of various types of experiments on an a priori basis. Each experiment is a part of operations and so it must be closely supervised.

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VII. CHEMICAL PROCESSING AND DISPOSAL OF REACTOR PRODUCTS

The reactor will be recharged every 15 days. About 1% of the U²³⁵ is burned daily. At each recharge the entire core or all the fuel and shim rod assemblies will be removed and replaced by fresh assemblies. The burned out fuel assemblies will be stored in the canal for at least 21 days as they require water cooling for at least this time to prevent melting of the fuel plates. After 21 days air cooling will be adequate, and the assemblies may be shipped at any time in coffins to the 25 Separations and Recovery Plant. Before shipment the end boxes will be cut off the fuel assemblies by sawing under water in the canal. The coffins will be filled with water to permit cooling by water siphoning.

The fuel assemblies are stored at the chemical plant for a total of 145 days after removal from the reactor before processing. In the processing, the fuel elements are dissolved and the uranium recovered and decontaminated. The recovered uranium is then reduced, formed into aluminum alloy, and reformed into fuel assemblies for use in the reactor. The U²³⁵ is used until its concentration has fallen to about 60% or for eight passes through the reactor.

The reactor products will be disposed of by the chemical plants by releasing the gaseous products to the atmosphere through scrubbers and filters and by concentrating and storing the liquid products.

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It is expected that the reactor will be used to manufacture a great many isotopes. Among those will be U^{233} in rather sizable amounts. Special processing plants will either have to be built or the irradiated materials shipped to other installations. In the case of the U^{233} , the irradiated thorium may be shipped for a time to Oak Ridge National Laboratory for processing in the semi-works already built for studies of the U^{233} separation process. The fuel assemblies will be manufactured at Oak Ridge in their pilot production plant until the fuel assembly production plant at the Reactor Testing Station is completed. A separate report covering the hazards involved in the chemical plants will presumably be submitted to the Commission. The MTR Steering Committee has no responsibility for these plants and, therefore, this report does not cover the plants.

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VIII. SAFETY MECHANISMS OF THE REACTOR

A great many safety mechanisms have been designed into the reactor and the reactor control circuits. Many of the safety devices that have been described previously in this report will only be mentioned in this section. Others will be described in more detail. In general, the safety devices deal with the reactor proper, the reactor control system, and the auxiliary facilities. Much of the development and engineering tests conducted on the reactor components is discussed in this section since, in a way, these tests have proved that the components are satisfactory in respect to design.

A. Safety Features of the Reactor Control System

The operation of the reactor control system covered in the general description and discussion sections, is described in this section with regard to the safety provisions in instrumentation and interlocking devices.

Precautions, insuring the reliability of instrumentation control, include fail-safe design and the operation of multiple instruments. In general, the fail-safe circuit designs of the instruments include reliable modulating arrangements that provide warning of malfunctioning prior to failure. Multiple instruments or complete instrument control circuits are provided for Period, Safety, and Control Rod Magnet Release Circuits.

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The reactor is controlled at all power levels by interlocking devices limiting the removal of safety control rods or effecting the insertion of these rods at different speeds. The rapidity of control rod insertion or reduction of reactor power level varies from the most rapid or complete insertion by "scram" action to controlled insertion to $0.01 N_f$ by fast or slow "setback" requiring 1.5 minutes or 8 minutes, respectively.

The interlock controls, summarized in the following under process operations and reactor operations, in accord with their supervisory functions, automatically check incorrect manipulations and error during operation of the reactor.

1. Process Operations

Since the removal of heat from the reactor is dependent on the function of process operations, process interlocks associated therewith are designed to effect "scram", fast setback or slow setback. These are covered in detail in ORNL CF 50-4-80, and Blaw-Knox Contract 3150, Conference Notes of May 22-23, 1950. They are reviewed briefly below.

Scram operations will take place with the reactor operating at $0.01 N_f$ when the main flow of cooling water decreases below five per cent of normal; and with the reactor operating at N_f when the flow in any one monitor tube is below 95 per cent of normal; when the supply

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of water in the working reservoir decreases below a safe minimum level and when an interruption to the normal electrical power is sustained for 30 seconds.

Fast setback occurs when the main flow of cooling water decreases below 80 per cent of normal and when the flow of cooling water to experimental plugs decreases.

Slow setback occurs when the temperature rise of the cooling water passing through the reactor reaches 105 per cent of normal; when the supply of water in the working reservoir decreases below the working level, or when the flow of air in the cooling system decreases.

The entire system of supporting processes for the MTR project from the principal water and air cooling systems of the reactor to the incident systems of supply and effluent wastes are provided with an adequate number of instruments. These instruments are associated with local and/or remote indicators, recorders, controllers and alarms as required.

2. Reactor Operations

The start-up and run of the reactor is accomplished by a combination of manual and automatic controls under the supervision of automatic permissive and safety interlocks associated with the reactor neutron instruments as indicated in Figures 15 and 16.

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The permissive interlocks, effected by the neutron instruments and control elements associated with the experimental facilities, the position devices of the regulating rod and fission chamber, and the control rod clutch engagements, permit start, and control the rate of withdrawal of the control rods.

The safety interlocks, associated with period, safety and power level instruments operate to limit, reduce or drop the power level of the reactor in accord with the degree of deviation from normal operating conditions.

B. Mock-up Tests

A mock-up of the reactor including fuel assemblies and reflector was made for hydraulic tests. This mock-up was full size and consisted of the main aluminum tank, grids and grid supports for the fuel assemblies and reflector. Natural uranium fuel assemblies and aluminum pieces in place of the beryllium reflector were used. Pumps and suitable piping were provided for circulating water through the lattice. Studies were made of the pressure drops, cross flows, etc.

After completion of the hydraulic tests, the mock-up was rearranged by installing enriched uranium fuel assemblies, beryllium blocks for part of the reflector, graphite balls outside of one-half of the aluminum tank, block graphite was placed outside the graphite

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balls, and concrete block shielding was placed outside of this graphite. Complete regulating and shim rods were installed along with sufficient of the reactor control instruments for a complete operating mock-up of the reactor at low fluxes. A view of the outside of the mock-up is presented in Fig 21.

The mock-up has been operated for several months at a power level of 100 watts. The reactor control instrumentation including the servo has functioned very satisfactorily. When the water is turned on, there is a very slight increase in the exercise or movement of the servo. This is to be expected since no means of degasifying the mock-up water was provided and there are always many air bubbles in the water.

The mock-up has proven that the design of the reactor is safe as far as its technical features are concerned. The expansions expected are well within the design limit.

C. Water System

The normal operation of the water system has been described. The cooling of the reactor after shutdown and in case of emergency is described in detail in the next section.

D. Fuel Assemblies

While the fuel assemblies are not a part of the Safety Mechanisms of the Reactor, the safety and success of the reactor is

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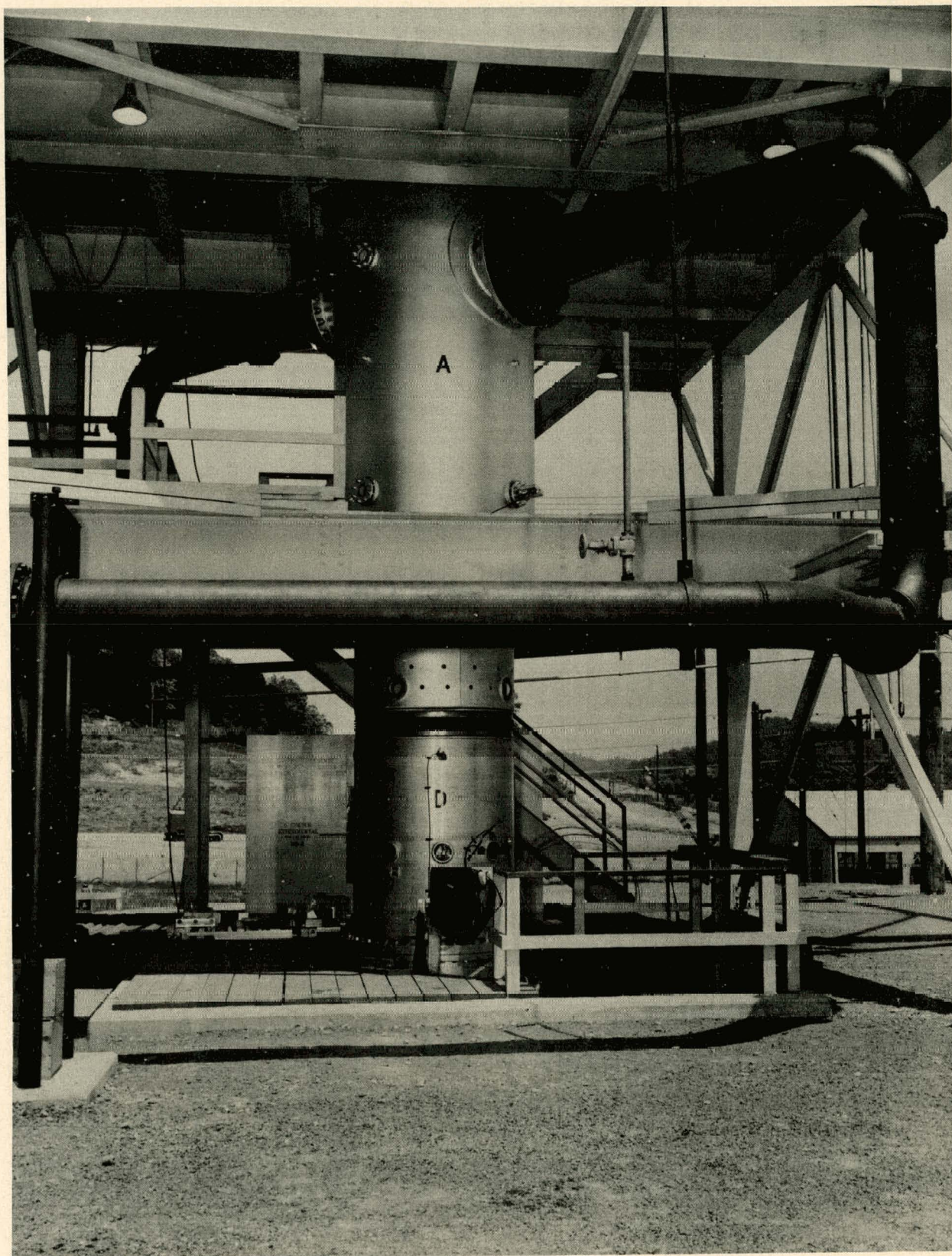


FIG 21

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directly related to the reliability of these assemblies. The discussion of the tests which have been made on the fuel assemblies is, therefore, placed in this section.

The design of the fuel assemblies has been developed to provide a means of transferring the very large amount of heat to the cooling medium and obtain an assembly with sufficient rigidity to stand up under the various forces due to temperature differences and the flow of the cooling water. At the same time, the design of the fuel assembly has been fixed within narrow limits by the fact that the cooling water also acts as the moderator.

1. Vibration

The water flow rate through the active lattice of the fuel assemblies is 30 ft per sec with a pressure drop of 28.0 psi overall. Vibration tests have been run with water velocities up to 40 ft per sec over extended periods without signs of failure of the fuel assemblies. Endurance tests at normal operating conditions have been run for 58 days, reference ORNL 50.

2. Heat Transfer

Four different types of heat transfer tests have been carried out. The first of these was the heating of single passage heat transfer units with the same internal dimensions as each passage in the fuel assembly. The purpose of this test was to determine if

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the actual heat transfer could be conducted without bubble formation in the film. Heat transfer rates up to fluxes of 740,000 Btu/hr sq ft were experimentally produced, reference MonT-433, page 20.

The second type of experiment was the heating of the edges of the active sections only by means of Calrods to determine if the temperature difference between the fuel section of the sandwich and the edge of the picture frame would cause cracking. No evidence of damage occurred. Mathematical analysis indicated that the maximum stress at any point in the fuel assemblies due to temperature differences is 28000 psi while the yield strength is well over 4000⁴⁰⁰⁰⁰ psi, reference ORNL 113.

A test was conducted with multiple passage units to determine if melting of the fuel plates would occur if one passage became plugged for any reason. No evidence of high temperatures or melting occurred. Analysis of this problem also indicated no trouble should be expected. A passageway with only 50% of the normal cross section would cause increases of only 20°F in the temperatures of the adjacent fuel plates. This work has demonstrated that all the heat from a single fuel plate can be transferred to the cooling medium by transfer on one side of the fuel plate only; or in other words, the factor of safety is at least two.

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The fourth test was a very simple one. A tube of aluminum was closed at one end, filled partially full of water, and an electrical current run through the tube that would result in a static heat transfer rate many times that of normal operation. In this case, the water immediately shot from the tube. The tube was not damaged and there was no evidence of incipient melting. This test will be referred to in the next section when the hazards of the reactor are discussed.

The following discussion refers to slow rates of change in k . Conditions present when there are very rapid rates of change in k are discussed in the next section. Under normal operation the film temperatures at the edge of the lattice will be 212°F or over; so not much additional heating would be required to flash some of the water and cause loss of moderator. This temperature is estimated as 242°F , since the pressure is about 45 psig or 60 psia at the center line of the reactor. The maximum metal temperature is about 245°F under normal operation or about 325°F when the steam flashes. These temperatures are far below the point at which any damage to the aluminum plates would occur since the melting point of the cover plates is about 1220°F . However, the fuel plate would continue to heat due to the fission heat, and would melt if the steam did not condense or be forced out and admit additional cooling water within a few seconds.

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It is believed that flashing would occur at the hottest points in the reactor and would result in a vaporization of water followed by condensation and cooling in localized regions without the entire lattice becoming filled with steam at one time. The flow rate and pressure of the water, 30 ft/sec, is high enough to force steam downwards and out of the lattice.

3. Irradiation Tests

Irradiation Tests of uranium aluminum alloys have been carried out at Hanford. Samples made from 93 isotopic % U^{235} and containing varying percentages of uranium in aluminum were irradiated for 1, 3, and 6 months. Small unclad samples decreased 3% in density with approximately 1% increase in each dimension under irradiation for three months. Small clad plates with 1 mm diameter fuel and $\frac{1}{2}$ mm diameter cover plates had slight density decreases but no dimensional changes in the lengths or widths. The data obtained to date are presented in the following tables:

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<u>U Initial</u>		<u>45 Day Sample</u>		<u>157 Day Sample</u>		<u>U atoms fissioned per total atoms present Initially</u>
<u>% wt.</u>	<u>%235</u>	<u>Thickness Change</u>	<u>Density Change</u>	<u>Thickness Change</u>	<u>Density Change</u>	
13.8	93.3	1.2	- .32	.23	- .40	1/500
29.6	28.0	1.2	- .30	.40	- .40	1/470

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Irradiation tests of a full scale fuel assembly in the NRX Reactor are planned but not yet under way.

4. Corrosion Tests

Very extensive corrosion tests have been run extending to many times the life of the fuel assemblies. The rate of corrosion in simulated reactor cooling water is negligible.

E. Beryllium

The use of beryllium in this reactor will be perhaps the first time so much of this metal has been used in one mechanism. Analysis has been carried out to determine if there is any danger of failure of the beryllium pieces of the reflector due to temperature differences. This work has indicated that the stresses can be kept low. Rods of beryllium have been heated and quenched with temperature differences several times those to be expected in the reactor with no damage to the beryllium. In addition, very extensive corrosion tests extending over many months with full scale assemblies have indicated that rate of corrosion of beryllium of good quality in simulated reactor water is very low. In fact, corrosion only takes place rapidly with high chlorides in the water. The water specification for the demineralized water for the reactor is .25 ppm chloride, far below the danger point.

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No danger is anticipated from beryllium poisoning since the beryllium will almost never be handled. The greatest danger will come during the production of the metal and in machining. Since the production and machining will be carried out under the direction of AEC Health Physics Groups, and in an AEC shop, this danger has been reduced to a minimum.

F. Shielding

Very extensive calculations have demonstrated that the shielding is adequate in all respects. A large factor of safety has been used in the shielding design since the amount of very high energy radiations is not accurately known. A careful test of the shielding will be made during the startup of the reactor. For this reason a very large number of test holes have been designed into the reactor in the permanent graphite and biological shield regions. The biological shield will be made with barytes aggregate giving a minimum density of 217 lbs/cu ft.

G. Control Rods

Extensive tests both in the laboratory and in the mock-up have been made on both the regulating rods and the shim-safety rods. Tests made by picking up shim rods with the magnets in flowing water at 30 ft/sec and dropping by de-energizing the magnets, have been made up to 1,000 cycles without damage to the rods.

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IX. SPECIAL HAZARDS

The greatest hazard of the reactor is, of course, the estimated 35,000,000 curies of fission products present after long operation. About one-fourth or about 9,000,000 curies of these fission products are gaseous and might be released in case of melting of the fuel plates. The Reactor Safeguard Committee considered this possibility at its meeting at Oak Ridge in February, 1948. Other hazards and the means taken in the design of the reactor to nullify them are discussed in the paragraphs below.

A. Cooling after Shutdown

During ordinary operation 20,000 gpm of process cooling water is run through the reactor. As described earlier in this report, this water is pumped from a sump tank to an overhead process water storage tank which delivers the water to the reactor under constant head. This tank will hold 150,000 gallons and thus could be used for 7.5 minutes in case of power failure.

The reactor process water system has been designed so that, in case of power failure of the main power supply, the reactor will continue to operate for thirty seconds without change. At the end of the thirty seconds the reactor will automatically go into "reverse" or the shim rods will be automatically started in, thus shutting the reactor down. At the same time, the main water valve in the line from the overhead tank to the reactor will start to close. This

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valve will require about 60 seconds to close to avoid danger of water hammer in the system. An eight inch bypass line around the main valve will not close, but will remain open holding the flow from the overhead tank to the reactor at about 1,000 gpm. This flow is more than sufficient to remove the heat of decay of the fission products and reactor materials. The purpose of the thirty second delay is to keep the reactor in operation through very short power outages. Power outages over thirty seconds may be relatively frequent since power will be supplied by a single high line to the reactor site.

When power outages over about five seconds occur, delay relays will automatically start a Diesel electric standby unit that will reach full power in less than one minute. As soon as the emergency unit is up to power, the emergency standby process water pump will start automatically. This pump will deliver 1000 gpm at low head directly to the reactor through pipe lines that bypass the overhead tank. Figure 9 illustrates the flow systems for the reactor water. No cooling of this water by evaporation or otherwise will be carried out as the temperature rise in the system will be only 17°F in the first 24 hours.

In case of failure of both the main water system and the standby water system, demineralized water can be pumped directly from the demineralized water storage tank to the reactor inlet pipes. The demineralized water plant will not operate on emergency power but if needed,

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filtered water can be run to the reactor by installation of a spool piece in the filtered water line. As a last ditch, a fire hose can be used to keep water over the fuel plates provided the top plug is removed. With all these water lines it is hardly conceivable that the reactor could not be supplied with some water.

As noted under the description of the reactor in the first part of this report, the inlet water lines enter the top of the reactor tank and the exit lines after leaving the bottom of the tank are turned upwards above the top of the lattice and then downwards and out the bottom of the shield. Syphon break lines run from the inlet to the tops of these exit loops. These exit water lines are joined into a single line below the basement of the reactor building. The lines are welded throughout including the connection to the seal tank weir box, in which the water level is always above the lattice in the reactor. Only an extremely severe earthquake could break this line. The danger of sabotage of this line is considered low since the line will be buried in a tunnel several feet underground. Simply breaking the line would probably not permit sufficient water to leak out to create immediate danger. Great care is being exercised in the design of these lines to make certain that the thermal stresses and the mechanical stresses due to water hammer are kept very low.

Further, danger of breakage of the main reactor tank is minimized by the thermal shield acting somewhat as a tank. The graphite balls

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on the outside of the tank and the beryllium on the inside of the tank would cut down water leakage in case of tank failure.

Subsequent to the February, 1948 meeting of the Committee in Oak Ridge, theoretical and experimental work was carried out to determine the minimum water cooling necessary to keep the fuel plates from melting after shutdown. The results of experimental tests at Oak Ridge were reported in ORNL-120, "Heat Dissipation from Fuel Assemblies of the High Flux Reactor After Shutdown," by W. K. Stromquist dated November, 1948. The conclusions from these tests are:

1. "With outside recirculation of water around the mock fuel assembly channel by natural convection, the heat flux could be maintained at greater than 27,400 Btu/hr sq ft, the capacity of the experimental equipment.

2. "With recirculation restricted to the mock channel itself the maximum heat flux that could be maintained without vapor binding was 17,800 Btu/hr sq ft when the water in the tank above the channel was at the boiling point, and 19,650 Btu/hr sq ft when the water in the tank was at 30°C."

A check was run at Argonne National Laboratory in which a fuel assembly (no uranium in the plates) was immersed in a column of water and heated directly with a high electric current passing through the aluminum of the assembly. By boiling the water with no forced

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convection 220,000 Btu/hr sq ft heat flux was maintained for 20 minutes without damage to the assembly.

According to an analysis by Dr. J. R. Huffman, see memorandum "Cooling After Shutdown," dated June 7, 1948, ORNL-CF 48-6-155, a copy of which is attached as Appendix B, heat production in the reactor after shutdown from 30,000 kw operation will be approximately the values shown in the following table:

FALL OFF OF HEAT FLUX AFTER SHUTDOWN

<u>Time</u>	<u>Production, kw</u>	<u>Heat Flux, Btu/hr sq ft</u>
Operation	30,000	322,000 average
4 sec	1,950	20,000
5 sec	1,300	13,000
30 sec	877	9,000
1 min	715	7,300
12 min	416	4,200
1 hr	269	2,800
1 day	90	900
4 day	45	400

These values indicate that as long as the active lattice is submerged in water, there is no danger of melting and releasing gaseous fission products. Thus, the problem reduces to one of designing the water system so that the fuel assemblies and the surrounding beryllium is covered with water at all times. This has been done by the arrangement of piping within the reactor described previously.

In case of breakage of the discharge chute in the sub-pile room, it is estimated that 35 gpm would flow out of the reactor

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tank when the level is 6 inches above the upper fuel assembly grid. This estimate is based on the size of the cooling passages in the beryllium that is installed in the chute during operation, and the water circulation holes in the side of the chute below this beryllium piece.

Realistically this seems to be excessive design because of the rapid drop off in the heat generation after shutdown. For example, one minute after shutdown after operation at 30,000 kw and assuming 200°F water leaving the reactor, only 51 gpm are required for the cooling of the reactor as far as the fuel plates are concerned. One hour after shutdown only 20 gpm are required. If the reactor water is allowed to boil, only 376 cu ft would be evaporated during the first 24 hours neglecting all conduction. The volume of the reactor tank above the lattice and below the inlet water line is 305 cu ft. Further, there appears to be little danger of loss of much of the water from the reactor tank even through this tank might fail. As noted above the graphite pebbles outside of the tank would limit the expansion of the tank and the beryllium reflector section would limit cross-flow and even then the thermal shield would retain most of the water.

In case all efforts fail and the fuel plates melt, most of the gaseous fission products could be held in the reactor system by shutting off the main water cooling system or simply going on "emergency"

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cooling. In the normal course of operation, all gases removed from the process water are removed in the flash evaporators and exhausted to the atmosphere at the top of the process water building. Floating heads are present on the seal tank, sump tank and process water reservoir. In other words, the process water system is completely closed except for the vents from the evaporators. The seal box weir is, of course, vented during shutdown but not during operation.

Those gases which might escape into the reactor building could be trapped there by shutting off all air circulation. No windows will be present in the reactor building. All personnel entrances will have two doors but single doors will be used on the truck entrances. The construction will be as tight as practicable, and the air exhaust will be through self closing shutters. Of course, some gases would diffuse out but these should not prove serious and only the immediate reactor area would have to be evacuated for any length of time.

B. Fission Product Loss from Fuel Plates

Theoretical and experimental work has demonstrated that the diffusion of fission products through pure aluminum is negligible. While the 10 year Kr^{85} does diffuse slightly at $250^{\circ}C$, it is considered that negligible amounts will diffuse out at the temperature of the fuel plates. However, loss of fission products due to

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mechanical injuries to the fuel plates, to pitting or corrosion of the aluminum cover plates or to the incomplete covering of the fuel sandwich part of the fuel plates are possibilities.

Each fuel plate will be inspected visually and by x-ray photographing before assembly. Each fuel plate will be given a heat test to determine if the cover plates were welded to the fuel alloy at all points. It was been found that by heating to 1130°F for one hour, that incompletely welded spots are raised in bubble-like areas. All fuel plates with any questionable welds or mechanical deficiencies will be rejected.

After brazing, each fuel assembly is inspected visually and by "Go-No-Go gages" to make certain that the distances between fuel plates are correct.

Corrosion of aluminum in simulated reactor water has been found to be negligible. However, localized pitting may possibly occur. In order to make certain that fission products do not get into the process water in large amounts, there will be a monitor tube directly below each fuel assembly which will measure the temperature, the water velocity and continuously sample exit water from each fuel assembly. The water samples will be automatically scanned by counters to determine the activity of the water. An alarm signal will be given to the operators in the control room in case the activity below any

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fuel assembly is found more than a few per cent greater than normal. Some activity will always be present due to the Na^{24} and Nl^6 in the water. In addition, skin plates of .005 in. 72S alloy are rolled hot to the outsides of the fuel plates for preferential corrosion to prohibit corrosion pitting of the 2S alloy plates.

C. Disintegration of Fuel Plates

No evidence of disintegration of fuel plates due to internal pressure of the gaseous fission products has been found to date. However, in use of 93.5% isotopic U^{235} , disintegration or bursting of the fuel plate might occur. It is known that this phenomenon will not occur with the alloys which are expected to be used and with the burnup times now planned. As soon as the reactor is placed in operation, samples will be placed in the reactor of the highest available isotopic concentration and the highest weight percentages that can be made to determine if bursting occurs with extremely high burnups. Meanwhile, tests are being continued at Hanford so that as much data as possible will be available when the reactor is placed in operation.

D. Neutron Beams

Locally, danger within the reactor building will be present in that neutron beams might escape from the beam holes if the beam hole plugs are not properly designed or operated. Fission counters will be placed on the walls opposite the six main beam holes and will ring an alarm in case neutron beams do escape.

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E. Explosion of Reactor

The Reactor Safeguard Committee at its meeting at Oak Ridge in February, 1948 considered the hazard involved in explosion of this reactor. A report of that meeting is quoted as follows:

"There is no direct danger from a violent nuclear explosion in a thermal neutron reactor like the one under consideration. The essential reason is the much longer nuclear multiplication period. A nuclear explosion can actually not produce as much as 1 Kg. TNT equivalent energy and this is not even sufficient to vaporize the water present in this reactor. On the other hand, if no water is in the reactor, the nuclear reaction cannot proceed. For this reason, consideration need not be given to dangers of destruction of the pile or the structures housing the pile by a severe explosion which might disperse in the air, the radioactive materials present in the pile."

In view of the above statements, no attempt has been made to repeat the calculations on the danger of a nuclear explosion as such. Rather, because of the conclusions obtained by North American Aviation in their study of reactor hazards, as reported in the report "A Study of Reactor Hazards", by M. M. Mills, dated December 7, 1949, NAA-SR-31, that all reactors having large amounts of excess activity are inherently unsafe, two other studies were undertaken. The first considered the inherent control of a rise in reactivity which would be possible by the most rapid continued withdrawal of the shim rods. The second

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study examined what would happen if there was an instantaneous rise in reactivity. Studies were undertaken of various reactivities up to and including 2% increase in both cases.

Some of the preliminary results of calculations that will appear in a report which is now being written, are given in the following paragraphs. This additional report on the general subject of the danger of melting of the lattice of the Materials Testing Reactor will be prepared and presented at a later date.

1. Change in Density of the Moderator

Dr. Bernard I. Spinrad of Argonne National Laboratory has calculated the decrease in weight of moderator necessary to reduce the reactivity of the reactor by 2%. He found a density decrease of 6.6% of the water or an equivalent increase in volume of the aluminum fuel plates due to temperature rise would decrease the reactivity by the 2%.

Aluminum expands approximately 7.1% between the steady state operating temperature and the melting point. Since the ratio of aluminum to water in the core is .7 to 1, the apparent decrease in the density of the moderator due to the volume change in aluminum would make a change of 5% if all the aluminum was heated uniformly thus appreciably decreasing the necessary change in water density

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to reduce the reactivity to safe limits. However, the flux in the reactor is not uniform as the highest flux is 1.8 times the average flux. In addition, the control rods and the experiments present will result in lowered flux levels at various points. Another consideration is the fact that the fuel plates lengthen as they are heated. Since the temperature is not constant over the length of the fuel assemblies, this is not considered further here.

The actual change in density of the water is very difficult to estimate due to the greatly subcooled condition of the bulk of the cooling water and the fact that it is in turbulent flow at 30 feet/sec. Due to the very short time interval involved, the bulk temperature will not be raised to the saturation temperature even though surface boiling is taking place at the interface with the fuel plates. It is difficult to predict analytically whether steam bubbles will have a life great enough to cause appreciable decrease in moderator density before vapor binding at the surface with resultant melting of the fuel plates.

Since the maximum speed of removal of the shim rods gives an increase in $\delta k/k$ of .22% per second when the rods pass the center plane, the control system will readily control the reactivity since it is designed to have adequate response to periods as low as .1 seconds.

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2. Instantaneous Changes in k.

Using the very conservative values of 30 milliseconds required dropping time and one "g" acceleration, the shim-safety rod system will safely handle an instantaneous increase of 1% $\delta k/k$. This is without consideration of the inherent safety characteristics mentioned previously, i.e., volume expansion of the aluminum, surface boiling in the moderator, and the negative temperature coefficient for thermal fission. The 5% moderator density decrease (relative to the core volume) caused by the expansion of the heated aluminum plus the other two effects should easily reach the 6.6% moderator density decrease that has been mentioned as required to be inherently safe for an instantaneous increase of 2% $\delta k/k$ without the help of the safeties. Indications are that between 2 and 3% instantaneous increase in reactivity can be tolerated in an emergency without melting of the fuel plates. Also, when aluminum goes through the change from the solid to the liquid phase it increases in volume 6.3%, and the local melting will change the geometry as it progresses to further decrease $\delta k/k$.

It is believed that in the worst theoretical accident with the Materials Testing Reactor, roughly 20% of the fuel plates would melt and shut the reactor down. The melting of 20% of the fuel plates will release as much as 2,000,000 c of gaseous fission products into

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the reactor process water system. It is doubtful if 500,000 curies would escape from the water system. No accurate estimate of this value has been attempted.

Calculations are being continued on this phase of the work using the REAC at Argonne. As noted previously, an additional report will be presented when the work is complete.

3. Expulsion of Coolant

Experimental work conducted at Argonne National Laboratory is covered briefly in Appendix E, a report by Mr. S. Untermyer, entitled "Reduction in Reactivity of Heterogeneous Reactor by Expulsion of Coolant," ANL-WPB-60. The experimental work indicates that steam is generated in a stagnant tube within ten milliseconds and this steam fills approximately 10% of 1 cm cooling passage.

The conditions in the MTR differ widely from these test conditions in that we have a dynamic system with water flow at 30 ft per second under approximately 45 psig at the center of the lattice. The volume of steam formed and present at any given instant probably would be much less than 10%.

4. Oxidation of Reactor Materials

The report by Mr. M. M. Mills, presented in NAA-SR-31, included the heat evolved in the formation of uranium oxide and

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aluminum oxide from the uranium and aluminum present in the reactor as contributing to disaster conditions. Experimental work conducted at Argonne, reference Appendix F, entitled "Reactor Hazards" by J. J. Katz and H. R. Hoekstra, dated Jan. 27, 1950, ANL-WDM-596, indicated that very little heat would be contributed due to oxidation of the aluminum and uranium. Further work recently carried out at Argonne confirms this result.

F. Sabotage

Some consideration has been given to sabotage in the design of this reactor. In this respect, it appears the most easily sabotaged item is the process water tower; but a bypass and other pipe lines will be installed which can be used to keep the reactor flooded in case of sabotage of this tower. Wrecking of this tower would cause considerable delays for repairs but no danger of explosion or melting the fuel plates.

The reactor itself is not easily susceptible to sabotage because of the very heavy shielding surrounding the entire reactor. Shaped charges would seem to be the most dangerous type of sabotage explosive and these would not do much damage to the reactor lattice unless they could be placed in experimental holes. This is unlikely as the plugs would not then fit without considerable alteration.

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All electrical lines within the reactor area will be underground and thus relatively free from danger of sabotage. Also key pumps will be not only emergency power driven but will be equipped with their own individual and local prime movers. Thus several devices would have to be sabotaged at once to cause severe damage.

Explosives placed in the control room in such a way that the instruments would be damaged would cause several days loss of operation but could hardly cause permanent damage. This same statement applies to the motor drives on the top plug.

The bottom plug is designed of very heavy steel and probably several hundred pounds of explosive would be required in the sub-pile room to seriously damage this plug.

Cutting of all lines between the reactor and the control room would de-energize the magnets on the shim-rods permitting them to drop, shutting down the reactor.

G. Bombing

The same considerations apply to bombing as applied to sabotage as discussed above. The thickness of the biological shield and its great mass means that ordinary bombing would do little harm to the reactor unless there was a direct hit and the bomb, by chance, penetrated the top plug. This is so improbable as to be negligible.

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In addition, to penetrate the top plug would require an armour piercing bomb. A bomb exploded in the reactor building would cause severe damage to the regulating and shim control rods on the top plug and to the control room instrumentation but could hardly result in release of the gaseous fission products.

The physical sizes and location of the reactor building and auxiliary buildings will make them very difficult to camouflage. The reactor building particularly is almost an ideal target for low level or skip bombing. Also, it would appear to be nearly an ideal target for a bomb or directed missile with homing devices. The dispersion of the various projects within the Reactor Testing Station make them very poor targets for area bombing. Also the dispersion of the MTR buildings within the area reduce the probability of hits due to high level bombing with conventional bombs.

The walls and roof of the reactor building will offer no real resistance to skip bombing or high altitude bombs with homing devices. The walls will cut down the damage due to fragments from near misses.

H. Third Safety Device

Some consideration has been given to building a third safety device. The present safety devices are:

1. The regulating rods.
2. The shim control rods.

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However, to date, no satisfactory third safety device has been suggested. The introduction of neutron absorbing gas or a powder has not been considered practicable. The use of spring loaded rods has been considered and a very slight amount of consideration has been given to a gun or pressure chamber which would fire cadmium or boron containing chains sufficiently fine to pass into the individual cooling water channels within the fuel assemblies until mechanically removed. All these devices have been considered impracticable.

I. Isotopes Present

The report of the Oak Ridge Meeting of the Safeguard Committee, see Appendix A, presented the following table of the maximum inventory of radioactive isotopes emitting alpha rays:

<u>Isotope</u>	<u>Half Life</u>	<u>Maximum Inventory</u>
U ²³⁶	1,000,000 yrs	.40 gm
U ²³³	160,000 "	700 "
Pu ²³⁹	24,800 "	.2 "
Pu ²³⁸	115 "	.005 "

In addition, isotopes may be present for irradiation purposes as part of the isotope production or research and development programs. No real estimate of the amounts of these isotopes can be made.

J. Storage of Fuel Assemblies

Fresh fuel assemblies will be stored in a vault until used. Used assemblies will be stored in the canal for at least

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21 days after discharge. The storage will be in 28 assembly units, cadmium poisoned, and several feet from one another. These units will be individually locked. During the initial months of operation until the chemical plants are completed and operating, the canal will be used to store up to a maximum possible number of 270 spent fuel assemblies.


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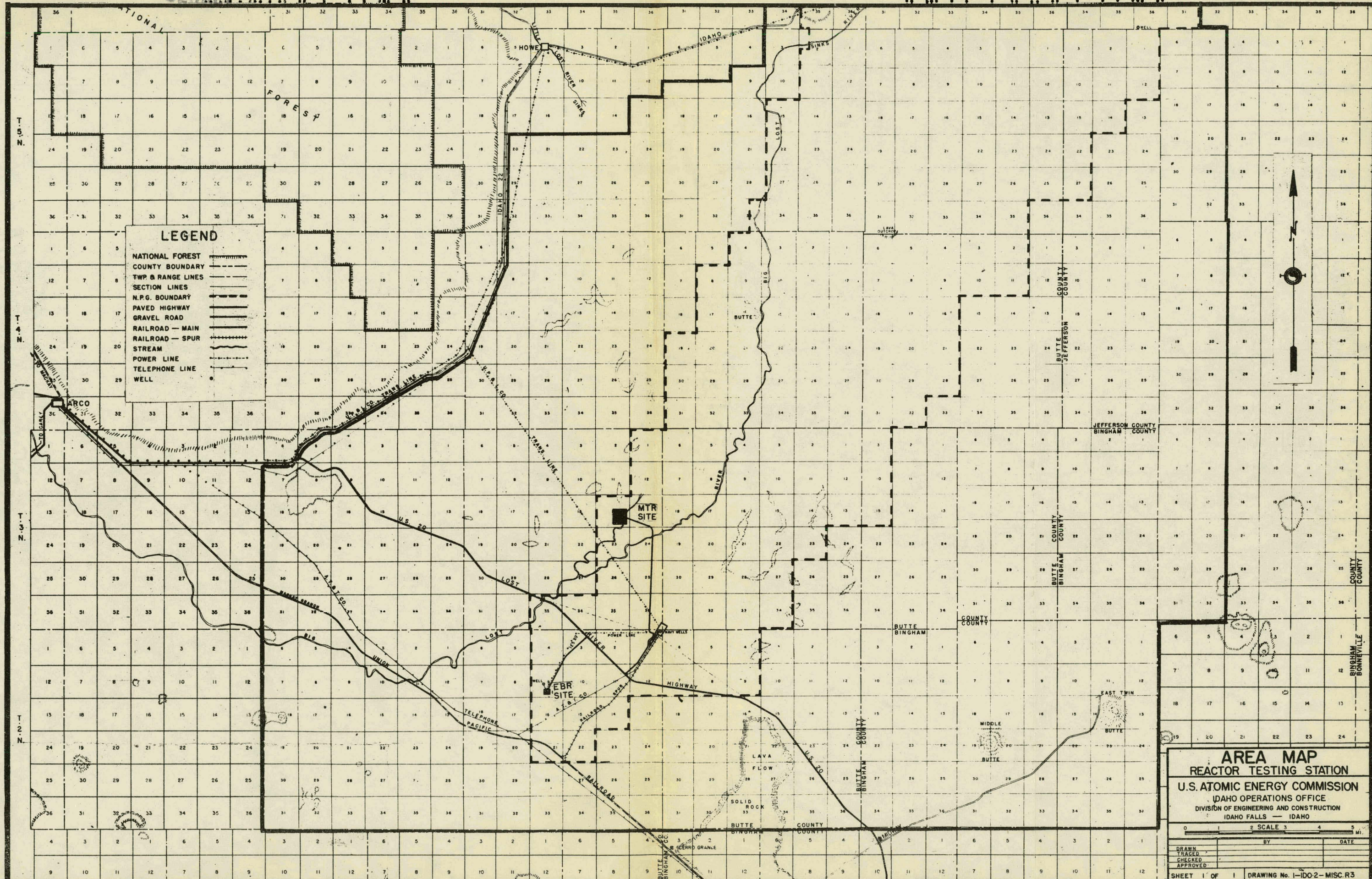
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X. POPULATION NEAR SITE

The actual site is shown on Figure 22 and the general site location is shown as Figures I and II of Appendix G. The reactor will be built about four miles Northwest of the Naval Gun Station and about 48 miles west of Idaho Falls. The actual site is located in Section 14 of township 3 north, range 30 East in Butte County, Idaho.

The towns and cities near the site with their population and distances from the actual reactor location are tabulated below. The populations of the Idaho towns and cities were taken from the 1948 Official Highway Map of Idaho.

<u>Town or City</u>	<u>Miles from Site</u>	<u>Population</u>	<u>Direction</u>
Howe	9.7	100	NW
Arco	11.3	550	W
Moore	16	300	W
Terreton	30	260	NE
Roberts	42	319	E
Mackay	43	776	NW
Blackfoot	44	3,680	SE
Shelley	45	1,751	E
Aberdeen	48	1,016	S
Idaho Falls	48	19,000	E
Lewisville	49	371	E
Menan	49	432	E
Ucon	50	449	E
Rigby	52	1,978	E
American Falls	53	1,439	S
Pocatello	57	25,000	SE
Rexburg	61	3,437	E
St. Anthony	65	2,719	NE
Rupert	82	3,167	SW
Burley	90	6,500	SW
Gooding	107	2,568	SW
Butte (Montana)	169	37,081	N
Salt Lake City (Utah)	204	140,267	S



AREA MAP
REACTOR TESTING STATION
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
DIVISION OF ENGINEERING AND CONSTRUCTION
IDAHO FALLS — IDAHO

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The Smith Hinchman and Grylls report gives the population of the counties near the site as follows:

<u>County</u>	<u>Population</u>
Bingham	20,100
Blaine	4,500
Bonneville	22,800
Butte	2,000
Clark	800
Jefferson	9,300
Madison	7,500

The Snake River Plains are practically uninhabited except in the north and east where there is some irrigated ground. During the summer the plains are pastured by occasional sheep herders. Some horses and antelope are occasionally seen but there is very little other wild life.


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XI. VITAL INSTALLATIONS

There are no vital industrial or war installations near the proposed site except those connected with the Reactor Testing Station itself. Within the hundred mile radius there are several important installations. There are several plants at Pocatello making phosphates and elemental phosphorous. The Naval Gun Factory with its machine shops and the Union Pacific Railroad Shops at Pocatello are important installations. The other communities are essentially distributing centers to the rural areas.

The dam at American Falls and to a minor extent the partial dam and power plant at Idaho Falls are important sources of local power.

Within the two hundred mile radius there are the Butte and Anaconda, Montana, copper mining and smelting plants, and just outside the two hundred mile radius are the copper mining and smelting and other manufacturing plants of the Salt Lake City area.

U. S. Route 20 crosses the site and in fact runs as close as six miles to the actual reactor site. While this is an important road, alternative routes are available. The railroad which serves the area is a branch of the Union Pacific serving Arco and Mackay with no through connections. The Pocatello to Butte branch of the

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Union Pacific is always 43 or more miles from the site while the main line of the Union Pacific through Pocatello to the west lies more than 50 miles from the site at all points.

The irrigated areas along the Snake River and in the Lost River Valley are important potato, sugar beet, and feed raising lands. A little marginal dry farming, mainly wheat, is carried out in the areas within the twenty-five mile radius.


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XII. WINDS AND PRECIPITATION

The information presented in this section is taken almost entirely from the report by Smith, Hinchman and Grylls, Inc., entitled, "Survey Report - Fort Peck, Montana - Pocatello, Idaho Sites," March 26, 1949, pages 40 to 50, and from "Meteorology of the Reactor Test Site, Arco, Idaho," prepared by the Weather Bureau, Sept. 16, 1949. A copy of the first section of this last report but omitting the tabular data is attached to this report as Appendix E.

A. Wind.

The surface wind is generally from the Southwest with variations dependent on the local temperature and weather conditions. Very low winds from the Northeast occur, see Figure 2 of Appendix G. The actual site of the Materials Testing Reactor is not far from the exit of the Little Lost River valley. At times local winds blow up this valley during the day and down at night. This flow of air tends to change the wind direction to the south in the daytime and probably more westerly at night although no data are available on night air flow. The average surface wind speed is 8-10 mph with stronger winds occurring in winter than in summer. The following frequency may be expected:

0 - 3 mph	10 - 20%
4 - 15	60 - 70
16 - 31	10 - 20
32 - 47	1 - 2
over - 47	less than 1

Very little reliable data are available on the upper air. Apparently up to 900 feet or so the winds are similar to the surface winds and from 3,000 to 25,000 feet the winds are westerly with average speeds of 15-25 mph. Practically no data except pictures of the smoke from the one explosion noted in Appendix G are available on the frequency of stable and turbulent wind conditions. In general, it is believed the winds are comparatively uniform. Inversions are discussed in Section XIII.

There is considerable mixing of the air during the hot clear days in the summer time. Frequent dust devils occur with mixing of the air up to perhaps 3,000 feet as a maximum.

B. Precipitation.

The average yearly precipitation in the area is about 10 inches. The greatest precipitation occurs in the winter months. The greatest precipitation in any one month was 2.6 inches indicating that floods are relatively unknown. Twenty-five to thirty inches of snow per season is to be expected. Fifteen inches of snow has fallen in a single day. Some precipitation may occur in any month.

C. Correlation of Precipitation and Wind Direction.

No data are available on this subject.

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D. Fog.

Fog is considered to be mild. During the winter months clouds may extend over the area especially at night. Very rarely this cloud or fog may drop to the ground. This cloud formation occurs most frequently from late November until late February and from midnight until noon. Apparently this cloud remains about 1000 or more feet from the ground. Mild winds usually accompany the cloud formation.

E. Temperature, Humidity and Dust

The average monthly temperatures at Arco vary from 14 to 66°F with the maximum recorded temperatures varying from -41 to +101°F. The humidity varies from 54 to 80% in the morning, 27 to 71% at noon and from 24 to 73% in the evening at Pocatello. The highest wet bulb temperature at the site has been estimated at 65°F. Due to desert type climate, frequent dust devils occur, but no very large dust storms have been recorded.

The design of the reactor area and the outside piping has been based on the assumption that five feet of frost is possible.

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XIII. INVERSIONS

The only information concerning inversions is presented in paragraph 2B, page 7 of Appendix G. Frequent inversions can be expected, but little is definitely known concerning the heights to which they extend.

Since low night winds from the northeast frequently occur, the most trouble with argon might be expected in early mornings after continuous discharge under stable air conditions with low wind rates. When the stable conditions are broken up by rising hot air, some contamination of the area may occur. This condition may be expected to be aggravated by the fact that the stack is located east of the reactor building.

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XIV. STACK GASES

There are three sources of contaminated air in the Materials Testing Reactor Area. These are:

- A. Reactor cooling air
- B. Laboratory hoods
- C. Vents of effluent control tanks

A. Reactor Cooling Air

A total of 1850 lb/min. of filtered cooling air is drawn through the reactor to cool the graphite, thermal shield and experimental plugs. The air in passing over the thermal shield and graphite may pick up particles of dust and graphite. In addition, the air will carry with it an estimated 1500 curies per day of argon 41 activity.

The air volume is estimated as about 34,000 cfm, at the estimated temperature and pressure existing at the top of the stack. Due to cooling, this air volume will contract to about 29,000 cfm; and the activity is based on this volume. The argon activity is then about 3.4×10^{-5} curies/ft³ or 1.3×10^{-3} microcuries/cc.

A dilution of 1300 is required to reduce the activity to the value of 10^{-6} microcuries/cc agreed to at the Chalk River Conference of the Radiation Protection Committees of the United States, Great Britain, and Canada, September 29 - 30, 1949.

The intensity of gamma radiation at the stack outlet was calculated from the concentration on the basis of the energy of the gammas emitted as 1.37 mev, as 1.0×10^8 ev/sec. cc.


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Calculations on the activity of the air are presented in "Argon Activity in Reactor Cooling Air," by J. A. Lane, dated November 17, 1949, ORNL Central File No. 49-11-180 attached as Appendix H of this report.

After leaving the reactor, the air passes through a round concrete duct to blowers and to a stack. Space has been provided for the construction of an air filter building if filters are required by operational tests.

B. Laboratory Hood Air

Laboratory hoods will be designed with two exhaust exits. The first of these will be the normal hood air which will be exhausted from the individual hoods by fans directly above the laboratories and without filtering.

The second air system is expected to carry away all the contaminated air from the hoods. This contaminated air system will consist of 2-inch diameter pipes with 2-inch openings in each hood and cave in the laboratories. These 2-inch tubes or pipes will join into the 6-inch header which will lead to a scrubber in the basement. All the air in the system will be passed through the scrubber and will be scrubbed with caustic solution to remove gaseous acid materials. A 12-inch pipe will run from the scrubber to the fan house and then to the stack. If filters are needed at a later date, they will be installed.

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C. Vents from Effluent Control Tanks

Due to the small amount of activity expected, the vents from the effluent control tanks, particularly the water system, will be exhausted at the point at which the tanks are located.

Hydrogen and Oxygen formed by the decomposition of the cooling water in the reactor will be removed from the water and compressed to atmospheric pressure by the deaerators and steam jet compressors of the cooling water system. This hydrogen and oxygen will be vented at the top of the process water building. The amount of radioactive gases mixed with the oxygen and hydrogen will be negligible. A liquid vent seal will be installed just under the roof to prevent flashbacks and explosions in case of possible ignition of these gases.

D. Stack

The stack will be 250 feet high. Calculations are presented in Appendix I on the stack height and activity to be expected at ground level.

With a 250-foot stack, activity at ground level will very seldom, if ever, attain tolerance levels.

The diameter of the stack at the top will be five feet with an exit velocity of 1740 feet per minute. It was desired to have the diameter such that the velocity will be nearly 2000 feet per minute to make certain that the exit gases are ejected well above the stack.

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E. Correlation of Reactor Operation with Meteorological Conditions.

The Idaho Operations Office plans to maintain a continuous survey of the meteorological conditions and of the activity at several points in the Reactor Testing Station Area. Reports will be correlated with operations of the reactor at all times.

The only hazard expected is that an inversion may occur under such conditions that the stack gases flow downwards immediately around the stack and spread out at the foot of the stack. Such a condition might occur without sufficient dilution to bring the activity below tolerance. Oak Ridge National Laboratory has never experienced this phenomenon, and Hanford has reported a similar condition only once. In case of the MTR, the elevated temperature with its heating of the stack would tend to force upward currents around the stack. The exit velocity of the air of 1740 ft/min with its temperature of 150°F or more makes this type of inversion very doubtful.

The almost continuous winds in the reactor area and the Little Lost River valley will cause a very wide distribution of the radioactive argon with very little danger within the reactor area. The location of the stack in the normal downwind direction from the reactor area will aid also. There are no people in the normal downwind direction for twenty or more miles from the reactor.

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XV. DISPOSAL OF LIQUID WASTES

The disposal of the liquid wastes has been largely described previously. Three classes of liquid wastes will be present -- the cooling water, warm wastes and hot wastes.

The disposal of the purge from the process water system is handled by means of storage until the activity has decayed to a low level. The activity in the process water is mainly due to Na^{24} which has a half-life of 14.8 hours. The activity on discharge is expected to be 1.7×10^{-3} curies per cu ft. After storage for 72 hours in the retention basin the activity will be 3×10^{-5} curies per cu ft. This is one tenth the value agreed to at the Chalk River Conference last fall.

The retention basin will consist of two 360,000 gallon tanks. One of these will be used for the storage of the process water, while the second will be held as a spare in case a fission break should occur. In this case, the entire contents of the process water system could be dumped into the basin at one time. No provision has been made for evaporating or otherwise disposing of this water provided it could not be eventually sent to the leaching bed. This has been taken as a calculated risk since the danger of a major accident of this kind is so remote and disposal facilities could be improvised if ever needed.

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The second class of liquid wastes are the warm liquid wastes from floor drains, laboratory sinks, etc. As noted previously, these will be caught in 1500 gallon sampling tanks, monitored and the contents sent to the retention basin or 10,000 gallon storage tanks. Later when the Chemical Plants are in operation, it is planned to ship these liquid wastes to the chemical plants for concentration and permanent storage.

The third class of wastes is the hot wastes from the laboratories. These will be collected in nine-liter shielded cans and stored until shipped to the Chemical Plants for disposal.

The philosophy of operation at the site is that the minimum laboratory work will be done there. As many of the items as possible will be returned after irradiation to the National Laboratories and other institutions for detailed experiments. However, some work will be done at the MTR site, and the wastes must be adequately handled. Long distance shipment of the wastes for disposal has been given no consideration.

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XVI. HYDROLOGICAL DATAA. Ground water

The source of the ground water in the area is mainly from the Little Lost and Big Lost River sinks. Additional water is from the Snake River and irrigated areas to the Northeast. The water flow is generally from Northeast to Southwest, although nothing is known concerning the flow in the immediate area of the reactor.

The surface down to about 45 feet is generally gravel with a little fine topsoil. Lava rock extends below 45 feet and downward to the bottom of the test well or a depth of at least 588 feet. The lava rock is honeycombed with openings of about 1/8" diameter. Frequently large openings occur, and these range upwards to the size of tunnels, tubes and caves. The water level is 453 feet below the surface. Estimates of the total water flow through the lava plain run as high as 6,000 second feet. Pumping tests on the first production well were carried out, 800 gpm being pumped with .35 foot pulldown. What little surface drainage there is, is toward the Northeast, opposite to the main body of water flow. Normally surface drainage is small due to the high porosity of the gravel overburden.

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B. Drainage

The path of water flow from the surface to the ground water level is unknown. In case of a major accident with loss of a large volume of liquid wastes it is expected that the drainage through the gravel overburden would be quite rapid. The drainage pattern through the lava would be less rapid but still very high as compared to flow through sands or clays. It is expected that the flow would be around rather than through any clay beds.

The results of permeability tests made by the Twin City Testing and Engineering Laboratory were given in their letters of February 6, 1950, and April 7, 1950, to the Blaw-Knox Construction Company as follows: "The permeability of the soil is quite low. The permeability of the sand and gravel soil is in proportion to the amount of clay and silt present (material passing #270 sieve). Those soils high in this fine material are the lowest in permeability. The gravel appears to be permeable, but because of the good gradation of the material from fine to coarse the void space is very small. The clay is almost entirely impervious." The actual permeability measured varied from 0 to .31 feet per day.

The rate of flow of the main body of water through the lava is estimated as one-half mile per year. On this basis the contaminated water would reach the Snake River Canyon springs and enter the Snake River in about 140 years, since the distance to the springs is about 70 miles.

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No experiments have been made on dilution as a function of rate of flow. However, the flow certainly would be viscous or laminar and not turbulent. Some dilution would take place as well as diffusion.

During the 140 year flow, absorption, adsorption, fission decay, along with the dilution and diffusion would reduce the radioactivity to a very small level. Due to dilution the radioactivity would reach the Snake River Canyon springs over a period of many years, thus resulting in still further dilution by the main stream of water.

Considerable more hazard is involved when we consider the other installations at the Reactor Testing Station. Since these , will be monitored constantly, means can be taken to obtain water from uncontaminated sources if necessary.

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XVII. EARTHQUAKES

"The Pacific Coast Uniform Building Code," 1949, designates the Testing Station region as Zone 2 as given by the "Seismic Probability Map of the United States", published by the U. S. Coast and Geodetic Survey. All buildings and structures in the Materials Testing Reactor Area will be designed and built in accordance with Zone 2 specifications as given in the above code. The pertinent section of the code is attached to this report as Appendix J, and a report on "Earthquake Information of Possible Application at Arco, Idaho," by J. Stewart Williams is attached as Appendix K.

The Smith, Hinchman and Grylls report gives information concerning 16 earthquakes that occurred in Idaho and adjacent areas from 1884 to 1945 with an intensity of V or more on the Rossi-Forel Scale. This report includes another report by J. Stewart Williams. His conclusions are as follows:

"Earthquake risk at this site is appreciable, but not great. Since isoseismal maps for principal earthquakes have been drawn, beginning in 1925, the isoseismals of only one earthquake reach Cerro Grande. Prior to this time several earthquakes recorded for surrounding areas may have been felt at Cerro Grande. There is no record of a major earthquake originating close to Cerro Grande.

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"However, Cerro Grande is surrounded by areas of comparatively high seismic activity. Furthermore, it lies in a region of geologically young faults, any one of which must be considered potentially active. For these reasons earthquake risk at this site should not be dismissed from consideration in planning an industrial plant to be built at the site.

"Cerro Grande is situated within one hundred and fifty miles of several areas of pronounced earthquake activity. Any one of these might produce a shock stronger than it has yet produced with a correspondingly greater intensity at Cerro Grande. The earthquake history of one hundred years for this area is very short, from the geological point of view. An earthquake might occur any day that would alter substantially our ideas of the distribution of seismic activity in the area about the Snake River Plains.

"Earthquake risk in any area is relative to the type of structure to be built. Reinforced concrete buildings, well constructed in every way, with high factors of safety and incorporating features recommended by engineers acquainted with earthquake-proof design, stand less risk of being damaged. Such buildings, set on the lava bedrock at Cerro Grande, certainly would be reasonably safe from earthquake damage.

"Considering the earthquake history of the surrounding area, and its geologic structure, any industrial plant involving large capital in

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capital investment or any government plant important to the general welfare should be constructed to resist intensities of eight or nine. The cost of meeting this requirement would not be great, and it would offer practically complete assurance against earthquake damage.

"Viewed from another angle, certain types of structures are of such design as to be essentially earthquake-proof. For such structures, earthquake risk in any area would not be great. For such structures at Cerro Grande it would be practically nil.

"No traces of recent faults are known by the writer to cross the Snake River Plains. The chances, then, of displacement in the ground that would cut water supplies are small, small enough to be eliminated from consideration."

In spite of the fact that a Zone 3 area exists both North and South of the Arco area, the distances are so great that Zone 2 has been considered completely safe.

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XVIII. FIRE

The probability of an ordinary fire having any effect on the reactor is very remote. All buildings in the reactor area are designed as completely fireproof. All partitions will be steel, and all furniture will be steel. In addition, there will be no combustible material except work paper in use and electrical insulation which, with a few exceptions, will be carried in conduits.

Fire extinguishers and portable hose will be available. No fire trucks or station will be maintained at the reactor area but will be available from the central facility station about four miles away.

The greatest danger of fire will exist at the fuel oil storage area of the steam plant, but such a fire could have no affect on the reactor. The fuel oil storage area will be equipped with a Foamite system for its protection.

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XIX. HAZARDS TO SURROUNDING POPULATION

The greatest hazard to the surrounding population is, of course, the remote possibility of melting of the fuel plates with release of the gaseous fission products. The only people within possible danger within the 25 mile radius are those in the region of Howe and Arco. Since the population of this area is only a few hundred, temporary removal of all of these from their homes could be carried out without excessive difficulty. Since Howe and Arco are the nearest towns to the area, it can be expected that their population will rapidly increase due to influx of people connected with the Reactor Testing Station. It is hardly conceivable that the population could ever become over a few thousand in any case.

It is believed that the danger to the surrounding population with a reactor designed in the manner in which the Materials Testing Reactor has been designed is very much less than many of the chemical plants and mercury boilers which exist in Metropolitan Areas.

It is difficult to conceive of any other danger or risk to the surrounding contry. This includes the meteorological, geological, and hydrological conditions.



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XX. REFERENCES AND ACKNOWLEDGMENT

Most of the material for this report was taken from departmental memorandums and reports issued by Oak Ridge National Laboratory and Argonne National Laboratory. Almost all the experimental work mentioned was done by Oak Ridge National Laboratory. The following is a list of references that give more details on the subjects covered in this report:

- MonT-433. "Feasibility Report on Clinton High Flux Pile" by Huffman, Leverett, Newson, and Weinberg.
- MonP-272. "Physics of the High Flux Pile" by Greuling, Soodak and Weinberg.
- ANL-4263 "Development Reactor Project Quarterly Report for Period Ending March 1, 1949."
- ANL-4304 "Materials Testing Reactor Project Quarterly Report for Period Ending Sept. 1, 1949.
- ANL-EKF-11 "Retention Basin" by D. H. Lennox.
- ORNL-120 "Heat Dissipation from Fuel Assemblies of the High Flux Reactor After Shut Down" by W. K. Stromquist.

References on the Control System are listed on Page 22.



Stuart McLain, Chairman
MTR Steering Committee

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APPENDIX A

"Report of the Atomic Energy Commission Reactor Safeguard
Committee on the Proposed High Neutron Flux, Water Cooled
Thermal Reactor," by Edward Teller dated February 10, 1948,
less its Addendum I.

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February 10, 1948

REPORT OF THE ATOMIC ENERGY COMMISSION REACTOR SAFEGUARD
COMMITTEE ON THE PROPOSED HIGH NEUTRON FLUX, WATER
COOLED THERMAL REACTORIntroduction

The members of the Reactor Safeguard Committee (Manson Benedict, B. G. Holzman, J. Kennedy, A. Wolman, and E. Teller, Chairman) together with G. Weil as representative of the Atomic Energy Commission and E. P. Wigner, as consultant to the Committee, met in Oak Ridge on February 8 and 9, 1948, to consider possible hazards of the proposed high flux, water-cooled, thermal neutron reactor expressly in relation to the possibility of its location at the Du Page site of the Argonne National Laboratory. As a basis for its studies the Committee had the Feasibility Report, MON T-433, together with a very complete explanation of the design and operation of the pile by M. C. Leverett, J. R. Huffman, Henry Newson, A. M. Weinberg, K. Z. Morgan, and other members of the Clinton Laboratories' staff. During the discussions we also had the benefit of advice from W. H. Zinn of the Argonne National Laboratory.

The purpose of the high flux pile is to provide the greatest practical intensity of thermal and resonance neutrons for purposes of nuclear theory and technology. In addition, this pile could be used as a converter, producing materials by neutron absorption. In particular, it is planned to use the pile for production of U²³³, and the possibility was mentioned that in case of emergency polonium may be made from bismuth in the high flux pile.

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This pile differs in many important respects from the one recently considered for construction at the DuPage site of the Argonne National Laboratory and we may profitably compare the properties of these two reactors. The fact that this is a thermal neutron reactor and that it is water cooled makes it more similar to reactors with which experience has been obtained. Moreover, the characteristically longer multiplication times under this arrangement make the pile much less susceptible to severe nuclear explosion, as will be discussed in detail in a subsequent paragraph. The high fission rate of this reactor, corresponding to a design energy level of 30,000 kw., introduces hazards in greater magnitude than were associated with the 1,000 kw. fast neutron reactor. In terms of hazards which would arise from the dispersion of the pile materials in air over a wide area, this increase in power level we find very serious as regards the radioactive fission product content. Fortunately, the design of this pile provides no higher level of alpha particle emitting radioactive substances than the 1,000 kw. fast reactor, including the alpha particle radioactivity of the Pu^{239} , the Pu^{238} , U^{233} , and U^{236} . It is to be emphasized that this consideration specifically excludes the production of Po from Bi. The rule of thumb formulas which were used in discussing the location of the 1,000 kw. reactor as applied to this thirty-fold larger quantity of fission products would provide a strong argument against location of this reactor at the DuPage site if there is any significant possibility of dispersing these fission products in free air.

As with the Chicago pile, the Committee has been impressed by the careful design of the pile and its great potential value as a research and engineering tool. It is evident that the pile was originally designed for a location other than the DuPage site.

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Enumeration of Hazards

The Committee has considered two classes of hazards: Those associated with the normal operation of the pile at its design power level and those which would arise in the event of accident or other unplanned mode of operation. The first class includes:

- 1) Radioactive content of the pile cooling water
- 2) The radioactive argon content of the stack gases
- 3) The content of radioactive substances emitting alpha rays in the pile structure
- 4) The fission product content of the spent assemblies discharged from the pile
- 5) The fission product content of the chemical processing plant which will be required to reclaim U²³⁵ from the spent assemblies
- 6) The fission product content of waste solutions from the pile

The second class includes:

- 1) Nuclear explosion
- 2) Interruption of flow of cooling water or loss of cooling water level

The radioactive content of the cooling water will, in the opinion of the Committee, give rise to no significant health hazard even in the immediate vicinity of the pile, provided the precautions described to the Committee are followed. These include shielding of the cooling lines and provisions of retention basins of sufficient capacity to hold the entire contents of the cooling water system in the event of rupture of the active elements of the pile and contamination of the cooling water by fission products.

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The Committee was advised that a maximum of 2500 c. of radioactive argon would be present in the stack gases conveying cooling air from the pile building. The information available to the Committee did not permit it to evaluate the magnitude of this hazard and it is felt that a further study should be made of it, taking into account all pertinent meteorological and geographical conditions.

The maximum inventory of radioactive isotopes emitting alpha rays in the pile structure was estimated to be

<u>Isotope</u>	<u>Half Life</u>	<u>Maximum Inventory</u>
U-236	>1,000,000 yrs.	.400 gm.
U-233	160,000 yrs.	700 gm.
Pu-239	24,800 yrs.	0.2 gm.
Pu-238	115 yrs.	0.005 gm.

Of these materials, the greatest hazard arises from U²³³. The expected maximum U²³³ content is no more serious a hazard than the Pu content of the fast reactor. When this is coupled with the much lower probability of vaporizing the materials of the high flux pile, it can be seen that the alpha activity hazard of this pile is less than the fast reactor previously considered.

The fission product radioactivity of the high flux pile is approximately 30 times as great as that of the fast reactor. So long as this material remains in the pile tank and a flow of cooling water adequate to remove the heat liberated by fission and the radioactive decay of the fission product is maintained, there is no likelihood of hazards arising from the presence of this material in the pile. However, as is discussed later, accidents which interrupt the flow of cooling water may, through melting and partial

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vaporization, cause dissemination of these fission products into the air surrounding the pile and constitute one of the most serious hazards associated with it. The same consideration applies to the fission product contained in spent assemblies, requiring that they be kept submerged and adequately cooled to prevent melting. In addition, these spent assemblies must be so handled they are not allowed to accumulate a critical reaction mass in any place in the storage channels.

Chemical Processing and Storage of Spent Wastes

A chemical and metallurgical processing plant to recover the U^{235} and prepare it for reuse will be an obvious necessity for a pile of this type. The Committee has not considered in any detail the nature of such a plant, but hazards would certainly exist in that large quantities of radioactive fission products would be present and in solution and perhaps subject to more ready vaporization than in the pile or spent assemblies. Moreover, the opportunity for loss of the radioactive material into nearby surface water is probably much greater in the chemical processing plant and storage tanks for spent wastes than in the pile itself or the storage area for spent assemblies. The chemical processing plant itself might well contain several million curies of radioactive fission products, and the storage tanks for spent waste might contain even an order of magnitude more fission products of rather long lifetime. It is thus apparent that the leakage of this activity in the surface water could constitute a very major hazard. Lacking better information as to the nature of surface water flow and the pattern of water supplies down stream, we have guessed that cities perhaps 50 miles or more away might be endangered. The Committee is forced to conclude that the storage facilities for these radioactive wastes must be much better located with respect to the water supplies pattern of neighborhood

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areas than they could be at the DuPage site. Partly because of the inherent hazard of the chemical processing plant itself and partly because of the difficulty of transporting liquid wastes from the chemical processing plant to remote storage, it will likely prove necessary to locate the chemical processing plant also at a remote place and adjacent to the waste storage facilities.

Hazards of Nuclear Explosions

There is no direct danger from a violent nuclear explosion in a thermal neutron reactor like the one under consideration. The essential reason is the much longer nuclear multiplication period. A nuclear explosion can actually not produce as much as 1 Kg. TNT equivalent energy and this is not even sufficient to vaporize the water present in this reactor. On the other hand, if no water is in the reactor, the nuclear reaction cannot proceed. For this reason, consideration need not be given to dangers of destruction of the pile or the structures housing the pile by a severe explosion which might disperse in the air the radioactive materials present in the pile. However, this is not meant to imply that fission heat or heat from fission products cannot melt the pile, liberating fission products should the water supply be stopped.

Hazards arising from Failure of Water Cooling Systems

Accidents which interrupt the flow of cooling water through the pile or cause loss of water level over the active elements of the pile constitute its most serious accidental hazard. Should the water flow be interrupted, it is anticipated that the pile will be shut down immediately; even in this event it is not clear how much damage might result. In this connection experiments on the cooling of the assembly structure by stagnant water might provide some

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reassurance. If the accident which interrupted flow of cooling water also made it impossible to shut down the pile, there is a possibility that evaporation of water from the active elements would stop nuclear fission, but the extent to which the pile elements might be melted or vaporized is very difficult to predict with certainty. A break in the cooling system below the level of the active elements in the pile might drop the water level below them and leave them without even stagnant water for cooling. Every possible precaution to prevent this occurrence should be taken, since the fission product heat from recent high level operation would surely melt the assembly elements and release gaseous fission products. An estimate of the maximum amount of radioactivity which might be released at this time in this way can be obtained from the fact that the total radioactivity of fission products present in the pile one hour after shut down is still about 35,000,000 c. Out of this approximately 9,000,000 c. might be considered gaseous and thus liable to escape into the air. Under any disposition of this pile consideration might well be given to methods of confining these gaseous fission products and preventing their widespread dissemination. It should be remembered in connection with this type of hazard that the safety which can be built into the water supply system is dependent to a considerable extent upon local conditions at the site selected. Factors which must be taken into account would include the location and production of various reservoirs, pipe lines, and the possibility of the pile remaining submerged even though a major break should occur in the cooling system.

General Conclusions

Although the possibilities of dangerous contamination of the air or surface water of the Chicago region with radioactive fission products from this pile can be made very small by proper design and careful operation, the Committee is not convinced that they can be so far reduced as to make the DuPage County site a proper location for this pile. We have been lead to this opinion

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by the large content of fission products resident in the pile, which, if dispersed in the air or ground water by accident, no matter how improbable, would constitute a very serious health hazard to the entire Chicago area. The factor is aggravated by the location of the DuPage County site relative to Chicago in the direction of the prevailing winds. Any further consideration of the location of the high flux pile on this site or other sites in this general vicinity should by all means take into account details of the meteorological conditions and the pattern of surface water movement. A supplement to this report dealing with the meteorological conditions of this vicinity will be prepared by Colonel Holzman. Preliminary consideration of this factor suggests that the prevailing winds make sites to the Southwest of Chicago particularly undesirable for a pile operating at this power level.

This Committee in all of its meetings to date has of necessity considered the safety aspects of each pile as an isolated unit independent of its relation to existing reactors, proposed reactors, and reactors likely to be proposed in the future. The Committee's analysis of the suitability of sites proposed for individual reactors would be more effective if it could be made by taking more and more into account the outlines of the reactor program as a whole. Also, provision for chemical processing facilities and radioactive waste disposal might more effectively be considered on a broader basis than for each individual reactor.

/s/ Edward Teller

CONCURRED IN BY:

Manson Benedict
Colonel Benjamin Holzman
J. W. Kennedy
Abel Wolman

ADDENDUM I -

"Climatic Annex" by Holzman - 3/19/48

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APPENDIX B

"Cooling After Shutdown," by J. R. Huffman

dated June 7, 1948 ORNL CF# 48-6-155

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Oak Ridge National Laboratory
Central Files Number
48-6-155

Date June 7, 1948

File _____

Subject Cooling After Shutdown

Those eligible to
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From: J. R. Huffman

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COOLING AFTER SHUTDOWN

SUMMARY

Exploratory calculations have been made to show that the high flux reactor will cool itself sufficiently after shutdown even though there is no forced convection of the water remaining in the pile tank.

Ten seconds after shutdown natural convection of the water in the pile tank will provide sufficient cooling without boiling if a down flow is available to complete a thermal syphon. Failing this less than 5% boiling will provide sufficient circulation for the same purpose.

More exact reduction factors in heat flux, etc, after shutdown have been developed and further calculations made to determine the total water requirements and flow rates assuming that forced convection is still available after shutdown. This is valuable for sizing parts of the cooling water system.

The conclusion is that failure of the cooling water system will not be a factor contributing to possible melting of the pile after shutdown and release of gaseous fission products as a hazard. This work has been confirmed recently by similar calculations and experiments at Argonne. Experiments now under way at Oak Ridge also strengthen these arguments.

INTRODUCTION

The Reactor Safeguard Committee, in its review of the high flux pile, stated that an important danger in the pile is the failure of cooling after shutdown. The danger is that the pile will melt and release the gaseous fission products into the atmosphere. Three recommendations resulted:

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Cooling After Shutdown

1. Provisions to insure cooling water supply after shutdown. This included possibility of flooding the pile building.
2. Provisions to maintain the atmosphere in the building or in the system.
3. Limit the initial operating power to 20,000 kw.

This memo discusses the first of these recommendations. This is not a report, but rather a compilation of exploratory calculations. Hence, no apologies are made for the presentation of the results.

Possible Accidents

Due to the momentum of the cooling water flow during operation, it is assumed that full flow will continue for ten seconds after the control rods have stopped the neutron reaction. Accidents which could effect the cooling water flow are listed as follows:

a. Stoppage of Main Pumps. This has already been taken care of, in a sense, in a design of a cooling water system as follows:

- 1) The 200,000 gallons available in the working and safety reservoirs, which is equivalent to 133 minutes at 1,500 gpm.
- 2) Provisions of a 1,500 gpm line from the demineralized water system.
- 3) It has been tentatively planned to provide a 1,500 gpm line from the filtered water supply at X-10. There is a reservoir in the filtered water system which contains 3,000,000 gallons and will be equivalent to 2,000 minutes at 1,500 gpm, not counting the filter plant supply rate of 4,000 gpm.

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Cooling After Shutdown

- 4) After 15 to 30 minutes it is reasonable to consider that the Fire Department could provide cooling by using the regular fire trucks.

b. Complete Rupture of Feed Water Lines From the Working Reservoirs. This might be caused by direct sabotage as can any of these accidents. Under such circumstances, Items 2, 3 and 4 under a. would still be available for cooling.

c. Complete Rupture of All Feed Lines or of the Tank Above Pile Level. This might be guarded against by (1) cooling by thermal syphon; (2) cooling by permitting water to boil; (3) assistance of Fire Department after 15 to 30 minutes.

d. Complete Rupture of Tank or Pipes Below Pile Level. As will be apparent subsequently in this memo, this seems to be the worst accident. After 15 to 30 minutes it is conceivable that again assistance could be obtained from the Fire Department provided water by some means could be maintained above the pile level in order to serve as shielding.

During operation the pressure in the tank above the pile is 45 lbs per square inch and below the pile 10 lbs per square inch, giving a pressure drop across the pile of 35 lbs per square inch. With complete rupture of the tank or water outlet pipes below the pile, the pressure drop would be 45 lbs per square inch. Assuming that the pressure drop across the pile is proportional to the 1.8 power of the velocity, a flow of 18,000 gpm would result. The working and safety reservoirs would provide cooling for 11 minutes and the 3,000,000 gallon reservoir in the filtered water system would provide 167 minutes. This would require a large diameter pipeline from the reservoir.

Power Levels

The heat to be removed after pile shutdown will be due to (1) gamma and beta rays from fission product decay, and (2) delayed and gamma induced neutron effects. For the purpose of this memo only the first of these is considered. It is estimated that the heat from the second cause amounts to less than a 10% addition.

After shutdown the total heat generated is given by

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Cooling After Shutdown

$$P = BF \left[t^{-0.2} - (t + T)^{-0.2} \right] \text{ Mev/Sec} \quad (1)$$

F = fissions/sec kw
 t = seconds after shutdown
 T = seconds of pile operation

This is taken from MonP-303, Pile Technology Lecture No. 16, and covers the range 10 seconds to 100 days. During operation the power level is given by

$$P = 187F \text{ Mev/Sec/kw} \quad (2)$$

so that the reduction factor is

$$A \left[t^{-0.2} - (t + T)^{-0.2} \right] \quad (3)$$

From the results of Cannon's measurements (CC 2176), the values of the kilowatts generated per gram of fission products are available from 30 minutes to 1,000 hours after shutdown. These have been converted to reduction factors, extrapolated back to times as low as 0.1 second and plotted. Calculations and curves are given in Table I and Figures I and II. By comparison with the above formula, a good value for the constant A appears to be 0.080 to 0.090 over the range 360 seconds to 4 days for a 20-day irradiation period.

There is another way to evaluate the constant A. During operation the heats generated are:

Total Heat	187 Mev/fission
Fission Product Heat	7 β Mev/fission
	5 γ Mev/fission
Pile Neutron Capture Heat	5 γ Mev/fission
Th, Al, H ₂ O Capture Heat	2.7 γ Mev/fission

During operation the total heat generated is the fraction $\frac{19.7}{187}$ of operating heat; $\frac{7}{187}$ due to β and $\frac{12.7}{187}$ due to γ . After shutdown this heat is $\frac{14.7}{187}$ of operating heat; $\frac{7}{187}$ of β and $\frac{7.7}{187}$ of gamma.

From these considerations, the heat generated in the pile at the instant of shutdown is

$$\frac{14.7}{187} (30,000) = 2,350 \text{ kw} \quad (4)$$

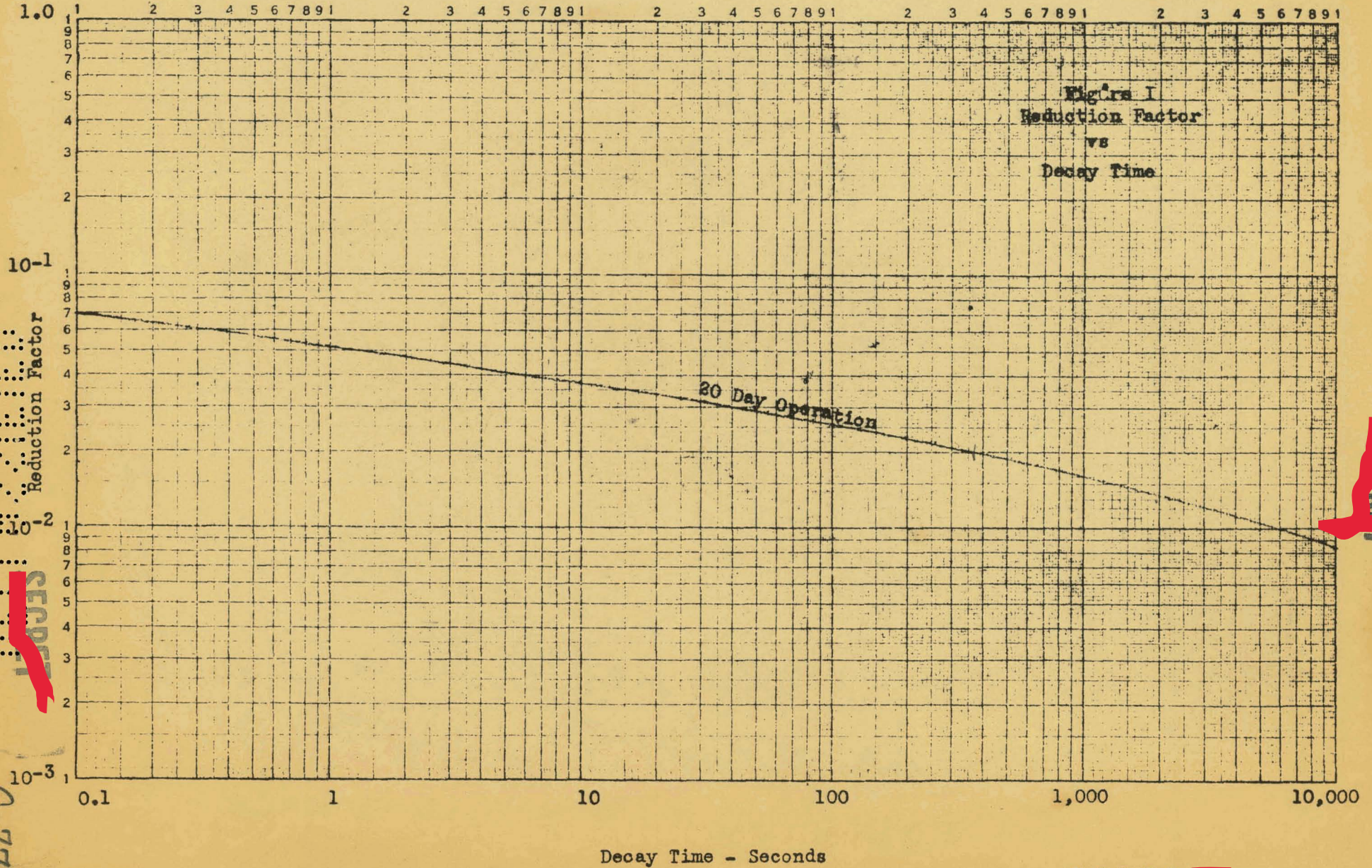
TABLE I

REDUCTION FACTORS AFTER SHUTDOWN

Cooling Time	$t^{-0.2}-(t/T)^{-0.2}$	10-Day Oper.			20-Day Oper.			40-Day Oper.		
		kw/gr	F	A	kw/gr	F	A	kw/gr	F	A
6 min	0.252	2.00	2×10^{-2}	0.0795	1	2×10^{-2}	0.0795	0.52	2.08×10^{-2}	0.0825
12	.213	1.65	1.65×10^{-2}	.0775	0.86	1.72×10^{-2}	.0806	.445	1.78×10^{-2}	.0835
18	.191	1.49	1.49×10^{-2}	.0780	0.78	1.56×10^{-2}	.0815	.405	1.62×10^{-2}	.0848
24	.177	1.38	1.38×10^{-2}	.078	0.73	1.46×10^{-2}	.0824	.378	1.51×10^{-2}	.0853
30	.167	1.32	1.32×10^{-2}	.079	0.70	1.38×10^{-2}	.0826	.360	1.44×10^{-2}	.0863
36	.159	1.25	1.25×10^{-2}	.0786	0.67	1.32×10^{-2}	.0830	.345	1.38×10^{-2}	.0868
42	.153	1.20	1.20×10^{-2}	.0785	0.64	1.28×10^{-2}	.0835	.340	1.36×10^{-2}	.0889
48	.147	1.16	1.16×10^{-2}	.0789	0.62	1.24×10^{-2}	.0842	.330	1.32×10^{-2}	.0898
54	.142	1.12	1.12×10^{-2}	.0789	0.60	1.24×10^{-2}	.0845	.315	1.26×10^{-2}	.0888
1 hr	.138	1.09	1.09×10^{-2}	.0790	0.58	1.16×10^{-2}	.0840	.305	1.22×10^{-2}	.0885
2	.113	0.89	8.9×10^{-3}	.0787	0.48	9.6×10^{-3}	.0850	.260	1.04×10^{-2}	.092
5	.085	0.67	6.7×10^{-3}	.0788	0.37	7.4×10^{-3}	.0871	.200	8×10^{-3}	.0941
10	.066	0.52	5.2×10^{-3}	.0782	0.29	5.8×10^{-3}	.0879	.163	6.5×10^{-3}	.0985
1 day	.046	0.36	3.6×10^{-3}	.0783	0.21	4.2×10^{-3}	.0913	.122	4.88×10^{-3}	.106
2	.0345	0.255	2.5×10^{-3}	.0738	0.155	3.1×10^{-3}	.0898	.092	3.68×10^{-3}	
4	.0232	0.164	1.64×10^{-3}	.0707	0.105	2.1×10^{-3}	.0905	.066	2.64×10^{-3}	
10	.0126	0.075	7.5×10^{-4}	.0555	0.054	1.08×10^{-3}	.0855	.035	1.42×10^{-3}	
20	.0075	0.035	3.5×10^{-4}	.0466	0.0270	5.4×10^{-4}	.072	.019	7.6×10^{-4}	
50	.0032	0.0097	9.7×10^{-5}	.0303	0.0083	1.66×10^{-4}	.052	.0065	2.6×10^{-4}	
100	.0012	0.00305	3.05×10^{-5}	.0254	0.0027	5.4×10^{-5}	.045	.00235	9.4×10^{-5}	

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Drawing # 5766



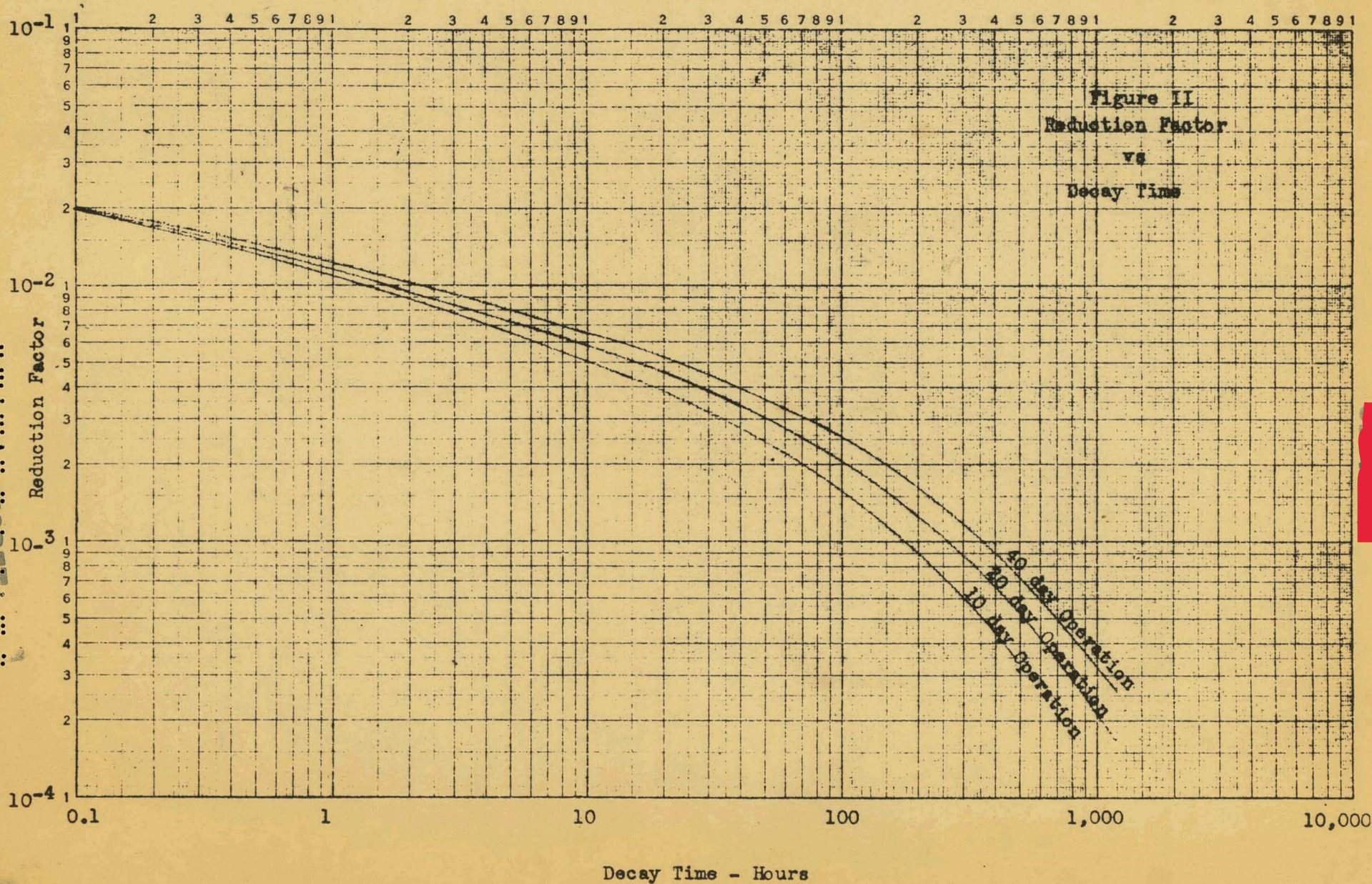
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Cooling After Shutdown

This amounts to a value of 0.0785 for the constant A. It should be noted that this is higher than usually assumed by the 2.7 Mev due to Th, Al, and water capture heat which is included to provide a factor of safety for induced activity decay and delayed neutron effects.

From MonP-272 the heat which escapes the pile and enters the beryllium is 83 watts/sq cm of pile sides. The pile has 11,280 sq. cm of side surface, so that 935 kw escapes into the beryllium. After shutdown only $\frac{5}{12.7}$ of this reaches the beryllium or 400 kw. Hence the heat evolved in the pile after shutdown is

$$2,350 - 400 = 1,950 \text{ kw} \quad (5)$$

This is assumed to come from the metal core although some is evolved directly in the water. For the pile after shutdown therefore, the heat evolved as a function of time is

$$1,950 \left[t^{-0.2} - (t + T)^{-0.2} \right] \text{ kw} \quad (6)$$

Of the gamma heat entering the beryllium from the pile during operation 3 watts/sq cm escapes to the graphite, so that the heat evolved in the beryllium due to pile influence is $(83-3) 11,280 = 905 \text{ kw}$. At shutdown this drops by $\frac{5}{12.7}$ to 350 kw, and follows the following equation

$$350 \left[t^{-0.2} - (t + T)^{-0.2} \right] \quad (7)$$

From MonP-272 the total heat (n and γ) evolved in the beryllium during operation is $(118-4.5)$ watts/sq cm or 1,280 kw.

These calculations are not too exact due to various facts. As mentioned previously, delayed and gamma neutrons will contribute some heat. The heat loss out the top and bottom of the pile have been neglected. Investigations in the Physics Division using high speed rabbits seem to indicate some very high energy short lived betas which would increase the heat release at very short times after shutdown. Work on the critical experiments indicates a possible change in the heat figures given in MonP-272. It is felt that the values are sufficiently accurate for the present survey, certainly at times the order of one minute after shutdown.

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Cooling After Shutdown

Heat Developed as Function of Time

Using the equation for decay given previously, the heat generated in the tank assuming no losses from the tank is given by

$$\int_0^t 30,000(.08) \left[t^{-0.2} - (t + T)^{-0.2} \right] dt$$

$$= 3,000 \left[t^{0.8} + (t + T)^{0.8} - (T)^{0.8} \right] \text{ kw sec} \quad (8)$$

For 20-day irradiation

$$= 3,000 t^{0.8} \text{ kw sec} = 2.42 t^{0.8} \text{ cal/sq cm sec} \quad (9)$$

and the following figures are obtained

TABLE IICUMULATIVE HEAT AFTER SHUTDOWN

<u>t</u>	<u>t^{0.8}</u>	<u>kw sec</u>	<u>Btu</u>	<u>Cal</u>
1 sec	1	3,000	2,840	715,000
2	1.74	5,220	4,950	1,248,000
5	3.62	10,900	10,340	2,600,000
10	6.3	18,900	17,900	4,510,000
30	15.2	45,600	43,300	10,920,000
60	26.5	79,500	75,400	19 x 10 ⁶
5 min	96	288,000	273,000	68.7 x 10 ⁶
10	106	498,000	473,000	119.2 x 10 ⁶
30	400	1,200,000	1,140,000	287 x 10 ⁶
1 hr	700	2,100,000	1,990,000	501 x 10 ⁶
2	1,220	3,670,000	3,480,000	876 x 10 ⁶

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Cooling After Shutdown

The pile is considered to consist of 22 fuel assemblies, 4 thorium shim rods and 4 cadmium shim rods. Compositions are as follows for 60 cm of active length:

TABLE III
PILE COMPOSITION

	<u>Volume</u>						<u>Heat Capacity</u> <u>cal/°C</u>
	<u>H₂O</u>	<u>Al</u>	<u>U</u>	<u>SS</u>	<u>Th</u>	<u>Total</u>	
Fuel Assembly	2,270	1,680	13			3,900	3,290
Thorium Shim Rod	2,180	880			722	3,780	2,974
Cadmium Shim Rod	2,200			1,400		3,600	3,380

Hence in the active section there are 115,100 cc and 97,800 cal/°C. Per liter of pile volume, heat capacity is 850 cal/°C; per liter of assembly volume 844 cal/°C. The total water in the active section is 67,500 cc.

In the beryllium there are 1.36×10^6 cc Be, 0.028×10^6 cc of H₂O and 0.912×10^6 cal/°C equivalent to 670 cal/°C per liter.

If all the water is out of the active assemblies, there is a total heat capacity of 30,310 cal/°C in 115,100 cc = 264 cal/°C/liter. Heat generated during operation is 10 kw per gram, 143 gr in assembly, or 1,430 kw in volume of 3,900 cc. This is equivalent to about 367 kw/liter.

The majority of the heat released in the assemblies is given off over the surface area of the core which amounts to 14,045 sq cm or 15.1 sq ft per assembly. At 30,000 kw this corresponds to 322,000 Btu/hr ft² (24.3 cal/sq cm sec) (Average value). (Actually the total area available for water cooling, including end plates, is 18 sq ft.)

The quantity of water in the lattice (67,500 grams) has a total heat content at 100°C (above 40°C) including boiling of $60 \div 540 = 600$ cal/gr or 4.05×10^6 cal. With no motion, the water will boil in about 10 seconds.

On the other hand if the lattice water can be linked by thermal syphon or other means of circulation to the total water in the pile tank, some 3,500 gallons or 26×10^6 gr are available for storage of heat. Then in two hours the temperature rise is 34°C.

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Cooling After Shutdown

Thermal Syphon

The force causing flow in a thermal syphon is

$$\Delta p = \frac{\Delta \rho}{2} h \quad (10)$$

Where $\Delta \rho$ is the difference in density of the water leaving the heated zone and the water in the reservoir above the pile and h is the height of the pile (60 cm). The factor $1/2$ accounts for the variation in density over h .

$$\text{Now} \quad u d w \rho c_p \Delta T = q h w \quad (11)$$

$$\Delta T = \frac{qh}{du} \quad (12)$$

The value of $\Delta \rho$ is roughly proportional to

$$\Delta t \text{ from } 40^\circ\text{C to } 100^\circ\text{C}$$

$$\Delta \rho = 0.000512 \Delta T \quad (13)$$

$q = \text{cal/sq cm sec}$

$\rho = \text{density}$

$w = \text{breadth of passage}$

$u = \text{cm/sec}$

$d = \text{width of passage}$

$T = \text{temp rise of water}$

$c_p = \text{sp ht}$

At steady state the force opposing flow is the friction in the passage given by the well known formula

$$\Delta p = \frac{2f h \rho u^2}{gd} \quad (14)$$

where

$$f = \frac{16 \mu}{d u \rho} \quad \text{for stream line flow} \quad (15)$$

This neglects the friction in the return passage of the syphon system.

By assuming a ΔT the two forces can be balanced by trial and error. A cold temperature of 40°C and a heat flux of $0.87 \text{ cal/sq cm sec}$ are used (10 seconds after shutdown).

Two types of syphons can be studied. First, the water flows up the heated sides of the assembly annulus and down the middle of the annulus.

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Cooling After Shutdown

For this case d in (3) is equal to $S/4$ and d in (4) and (5) is equal to S where S is the 3 mm width of the annulus; and heat into the water is 0.87 cal/sq cm sec.

For this case, i.e., internal syphon, a 60°C rise is not sufficient to remove the heat at 10 seconds after shutdown; any ΔT greater would result in boiling.

The second case is that in which the syphon water moves upward through the whole annulus and downward by some other path, i.e., an external syphon. For this case d in (3) is equal to S ; d in (4) and (5) is equal to $2S$; and q is replaced by $2q$ in (2).

For this case a temperature rise of 30°C is sufficient. This neglects friction in the external passage for water return. To handle this, it is assumed that on shutdown the beryllium plug for the discharge chute is automatically raised. This provides a path 100 cm long of 15 x 15 cm cross section for down flow. In addition, down flow can go through the beryllium and at the tank edge.

TABLE IV

CROSS SECTION FOR FLOW

30 Assemblies (up flow)	=	1,200 sq cm
Discharge chute (down flow)	=	225
Beryllium (down flow)	=	280
Annulus (1/16") at tank wall, (down flow)	=	67

By trial and error it is discovered that at 35°C ΔT the total up flow in the assemblies is 11,900 cm³/sec (1,200 x 9.94 $\frac{\text{cm}}{\text{sec}}$), requiring a velocity of 53 cm/sec for down flow in the discharge chute. For this case the driving force = 0.52 gr/cm² and the friction force is 0.529 gr/cm² (0.301 up flow and 0.228 down).

Hence an "external" syphon will provide cooling at 10 seconds after shutdown and with a ΔT of the water of 35°C, giving a 75°C temperature in the water leaving an annulus.

Heat Removal by Boiling

The other method of linking the water in the reactor lattice with the bulk water in the pile tank is by boiling with the steam bubbles condensing

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Cooling After Shutdown

in the bulk water above the reactor. In addition, this might be the method required to start the thermal syphon to operating, the steam rise acting as an air lift.

The driving force would be at least the height of water in the discharge chute

$$h\rho = 100 \text{ gr/cm}^2 \quad (16)$$

This is backed by the friction drop in the annulus where a mixture of steam and water is flowing. By assuming percentages of steam in the water, turbulent flow so the friction factor is 0.006, one can calculate the opposing force.

$$\Delta p = \frac{18 \times 30 \times 2 f h \rho (1-x) u^2}{gd} \quad (17)$$

where x is the fraction of steam

$$u = \sqrt{\frac{90.9}{1-x}} \quad \text{cm/sec}$$

TABLE V

HEAT REMOVAL BY BOILING

% Steam	u cm/sec	V cc/sec	Liquid cc/sec	Steam cc/sec	Heat Removal cal/sec
5	9.8	21.8	20.7	1.1	1244
10	10.1	22.4	20.2	2.2	1212
20	10.7	23.7	19	4.7	1142
40	12.3	27.3	16.4	10.9	989
60	15.1	33.5	13.4	20.1	813
80	21.3	47.3	9.5	37.8	572

The required heat removal is 680 cal/sec. It is concluded that even a few percent steam is sufficient. Vapor binding is not likely with 5 to 10% steam. It is interesting to note that the liquid water pumped by a small amount of steam is more effective than more complete vaporization.

Cooling Water Requirements

It is interesting to investigate further the water requirements after shutdown. During operation at 30,000 kw 13,100 gpm of water at 30 ft/sec

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Cooling After Shutdown

inside and 25 ft/sec outside the assemblies are required through the lattice portion including leakage water. In addition a flow of 1,900 gpm at 12 ft/sec is required through the beryllium (arbitrarily, in overall design 1,000 gpm more should be added for thorium cooling).

The rates of flow after shutdown are approximately proportional to the rate of decay of heat release. For the lattice this is

$$13,100 \frac{1,950}{30,000} \left[t^{-0.2} - (t + T)^{-0.2} \right] \quad (18)$$

and for the beryllium

$$1,900 \frac{350}{1,280} \left[t^{-0.2} - (t + T)^{-0.2} \right] \quad (19)$$

Since reduction in beryllium heat is less than in pile heat by a factor of 4.2 although the pile water requirements drops off more at shutdown, total flow is determined by cooling rates dictated by the beryllium. It should be remembered that this water flow is determined by thermal stress rather than by heat capacity considerations. As a result the flows through the pile system as a function of time after shutdown are given in column 8 of the following table, calculated from

$$15,000 \frac{(350)}{(1,280)} \left[t^{-0.2} - (t + T)^{-0.2} \right] \quad (20)$$

These figures should be checked for heat transfer using

$$h_f = 150 (1. + 0.011t) \frac{V^{0.8}}{D_e^{0.2}} = \frac{150 (2.1)}{(0.227)^{0.2}} V^{0.8} \quad (21)$$

V in ft/sec
D_e in inches
h_f = Btu/hr ft² °F

TABLE VI

FALL OFF OF HEAT RATES AND WATER RATES

Time	Reactor Lattice			Beryllium			Reactor Lattice		
	kw	Cal/sq cm sec	gpm	kw	gpm	gpm*	ft/sec*	ft/sec	Active Second
Operation	(30,000)	(24.3)	13,100	1,280	1,900	14,000	30		
0 sec	1,950	1.525	850	350	518	3,570	8.2	1.95	
1	1,840	1.43		330					
5	1,300	1.01		234					
10	1,120	0.87	488	201	298	2,050	4.7	1.12	
30	877	0.68		158					
1 min	715	0.55		128					
2	606	0.47		109					
6	490	0.38		88					
12	416	0.32		74					
30	326	0.25		58					
1 hr	269	0.21	118	48	72	498	1.13	0.27	
2	220	0.17		39					
10	129	0.10	56	23	34	236	0.54	0.09	
1 day	90	0.07		16					
2	67	0.05		12					
4	45	0.03	20	8	12	82	0.19	0.05	

* Based on Beryllium controlling

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Cooling After Shutdown

The bulk temperature is taken as 100°F, the allowable film temperature drop as 100°F, and the scale coefficient as 25,000. The possible heat fluxes are calculated in the following table.

TABLE VII

HEAT TRANSFER RATE vs WATER VELOCITY

<u>ft/sec</u>	<u>h_f</u>	<u>h</u>	<u>Cal/sq cm sec</u>
30	6,450	5,120	38.6
8	2,230	2,050	15.5
5	1,534	1,450	10.9
2	739	716	9.5
1	424	416	3.14
0.5	244	242	1.82
0.2	117	116	0.875

This shows the flow in the lattice dictated by the beryllium flow is very much more than sufficient to cool the lattice (by factors of 10). Actually 1 ft/sec in the lattice is more than sufficient to cool at 10 seconds after shutdown; it takes out 3.14 cal/sq cm sec. The actual heat flux is 0.87 cal/sq cm sec which requires 0.2 ft/sec. (See Table VI)

The heat generation in the beryllium at the median plane in the material cooled by one 1/8" D hole nearest the pile is 7.5 watts/cc. However 5 watts/cc is taken as average. The 1/8" D hole drains heat from a surface of 7 sq cm over 100 cm of length.

So 3,500 watts is removed from a cooling surface of 30 π sq cm or a heat flux of

$$\frac{3,500 \times 0.24}{30} = 9 \text{ cal/sq cm sec} \quad (22)$$

This is during operation. The remainder of the beryllium receives much less heat as shown by the fall off given in MonP-272. As a function of time this heat flux falls off after shutdown as follows

$$(9 \text{ cal/sq cm sec}) \left(\frac{350}{1,280} \right) \left[t^{-0.2} - (t + T)^{-0.2} \right] \quad (23)$$

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Cooling After Shutdown

Recent experiments indicate that there is probably a factor of safety of 4 in the thermal stresses developed in the beryllium. Furthermore, thermal stress is not controlling for the emergency heating after shutdown.

So calculations have been made assuming that the heat fall off rate for the pile can be used for the beryllium flow rate. Column 8 of the Table VIII shows that water would boil in the beryllium after shutdown. For safety it is decided to permit the water in the beryllium to reach the boiling point, i.e., roughly a 60°C temperature rise and the flow rate in the beryllium calculated (see column 9). This rate is about exactly twice the flow rate dictated by pile requirements. It is suggested therefore that the flow rate after shutdown could be graded according to

$$2 \times 15,000 \frac{1,950}{30,000} \left[t^{-0.2} - (t + T)^{-0.2} \right] \text{ gpm} \quad (24)$$

and given in the last column of Table VIII.

The total water requirements after shutdown can now be estimated by

$$\int_0^t \frac{1,950}{60} \left[t^{-0.2} - (t + T)^{-0.2} \right] dt \quad (25)$$

$$= 40.5 t^{0.8} \text{ gallons}$$

with sufficient accuracy.

TABLE IX
TOTAL WATER REQUIREMENTS

<u>Time</u>	<u>t^{0.8}</u>	<u>Gallons</u>
1 sec	1	40.5
2	1.74	71
5	3.62	146
10	6.3	255
30	15.2	617
60	26.5	1,080
5 min	96	3,900
10	166	6,750
30	400	16,200
1 hr	700	28,400
2	1,200	57,000

TABLE VIII

BERYLLIUM HEAT FLUX AND COOLING REQUIREMENTS

Time	Cal/sq cm sec	Pile Velocity	Be Velocity	Be Cal/sec	H ₂ O Be gr/sec	ΔT °C	Flow for ΔT of 60°C gr/sec	Pile ft/sec	Be ft/sec	Total gpm
Operation	9.0	30 ft/sec	12 ft/sec	849	29	29.4				15,000
0 sec	2.46	1.95	0.78	232	1.89	123.0	3.87	3.9	1.56	1,950
1	2.32	1.84	0.736	219	1.78	123		3.68	1.47	1,840
5	1.65	1.31	0.522	156	1.26			2.62	1.04	1,300
10	1.41	1.11	0.447	133	1.08	123	2.22	2.22	0.89	1,120
30	1.11	0.88	0.351	105	0.85			1.76	0.70	877
1 min	0.9	0.71	0.288	84.8	0.79			1.42	0.58	715
2	0.76	0.61	0.243	72.2		123	1.2	1.22	0.45	606
6	0.62	0.49	0.197	58.5	0.48			0.98	0.39	490
12	0.52	0.41	0.166	49	0.40			0.82	0.33	416
30	0.41	0.33	0.13	38.7	0.31			0.66	0.26	326
1 hr	0.34	0.27	0.108	32	0.26		0.533	0.54	0.22	269
2	0.27	0.22	0.088	26.2	0.21			0.44	0.18	220
10	0.16	0.13	0.051	15.1	0.12		0.252	0.26	0.10	129
1 day	0.11	0.09	0.036	10.4	0.08			0.18	0.07	90
2	0.085	0.067	0.027	8.0	0.06			0.13	0.05	67
4	0.057	0.045	0.018	5.4	0.044	123		0.09	0.03	45

Sq cm of heat surface = 30π

$$XS \text{ for flow} = \frac{(2.54)^2}{(8)} \frac{\pi}{4} = 0.0794 \text{ sq cm}$$

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APPENDIX C

"Information for Reactor Safeguard Committee Meeting
September 9, 1948," by John R. Huffman and S. Untermyer
dated August 26, 1948. ORNL CF# 48-8-348.

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Date August 26, 1948

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Subject Information for Reactor Safeguard
Committee Meeting

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By J. R. Huffman

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To George Weil

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INFORMATION FOR REACTOR SAFEGUARD COMMITTEE MEETING

September 9, 1948

(High Flux Research Pile)

John R. Huffman and S. Untermyer

One of the main objections raised to the high flux pile operated at 30,000 kw was the danger after shutdown of the fuel assemblies melting and releasing fission products into the air. This objection, it was stated, could be removed if adequate water flow could be maintained after shutdown to remove the heat liberated by radioactive decay of fission products. Accidents might occur which would cut off all of the three supplies of cooling water. It was expressed that maintaining the pile lattice submerged in water might provide adequate cooling by natural convection or partial boiling.

Calculations and experiments have been performed which show that as long as the pile can be kept submerged there is no danger of melting. By natural convection without boiling, 27,500 Btu/hr sq ft can be removed. By boiling the water as much as 220,000 Btu/hr sq ft can be removed for at least twenty minutes. Even if a complete thermal syphon cannot be provided, i.e., the return passage is not available, the water in the pile tank will remove 19,500 Btu/hr sq ft. At the instant of shutdown after operation at 30,000 kw the heat flux is 20,000 Btu/hr sq ft, falling to 13,000 in five seconds.

The accompanying sketches show changes in the design presented in Mont-433 which will assure submergence at all times, barring a direct bombing or a drastic earthquake.

Sketch No. 1 shows changes in the pile structure. The inlet and outlet air and water ducts are relocated in the top of the pile structure. The outlet water ducts are provided with air bleeds at their topmost point to

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eliminate syphon effects. In this manner the thermal shield would provide a secondary tank to hold water in case the pile tank should rupture. If this is so, these changes are sufficient to assure continual submergence of the pile lattice.

One of two other provisions could also be made. In the first alternative a safety reservoir could be placed on top of the pile building or in the pile shield and directly connected to the pile tank. As a second alternative, the basement could be eliminated, the pile placed slightly below ground level and the canal water level brought above the upper level of the pile lattice. In case of an emergency the water in the canal would flow into the pile tank, through any ruptures into the pile structure and through small leaks around the beam holes into the pile building, eventually filling the building with water to a height of 4-5 feet. In this case sufficient water storage capacity would have to be available in the canal or an attached reservoir. Sketches No. 2 and No. 3 show these two additional provisions.

Calculations indicate that by increasing the cooling gap from 0.3 cm to 1.0 cm, natural convection air cooling is sufficient to prevent melting at operating levels below 3,000 kw.

The committee, in its report of February 10th, suggested that consideration be given to methods of confining any escaping gaseous fission products. If such a provision is necessary, the pile and surrounding experimental space conceivably could be enclosed in a pressure shell with air locks. To handle 15 lbs/sq in pressure, a steel shell 1/2" thick, 60 ft in diameter weighing 100 tons would be necessary.

The attached pages present additional data and calculations to assist the committee in studying the pile at various operating levels up to 30,000 kw.

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AIR
OUTLET ←
WATER INLET →
AIR INLET →

→ WATER
OUTLET

GROUND LEVEL

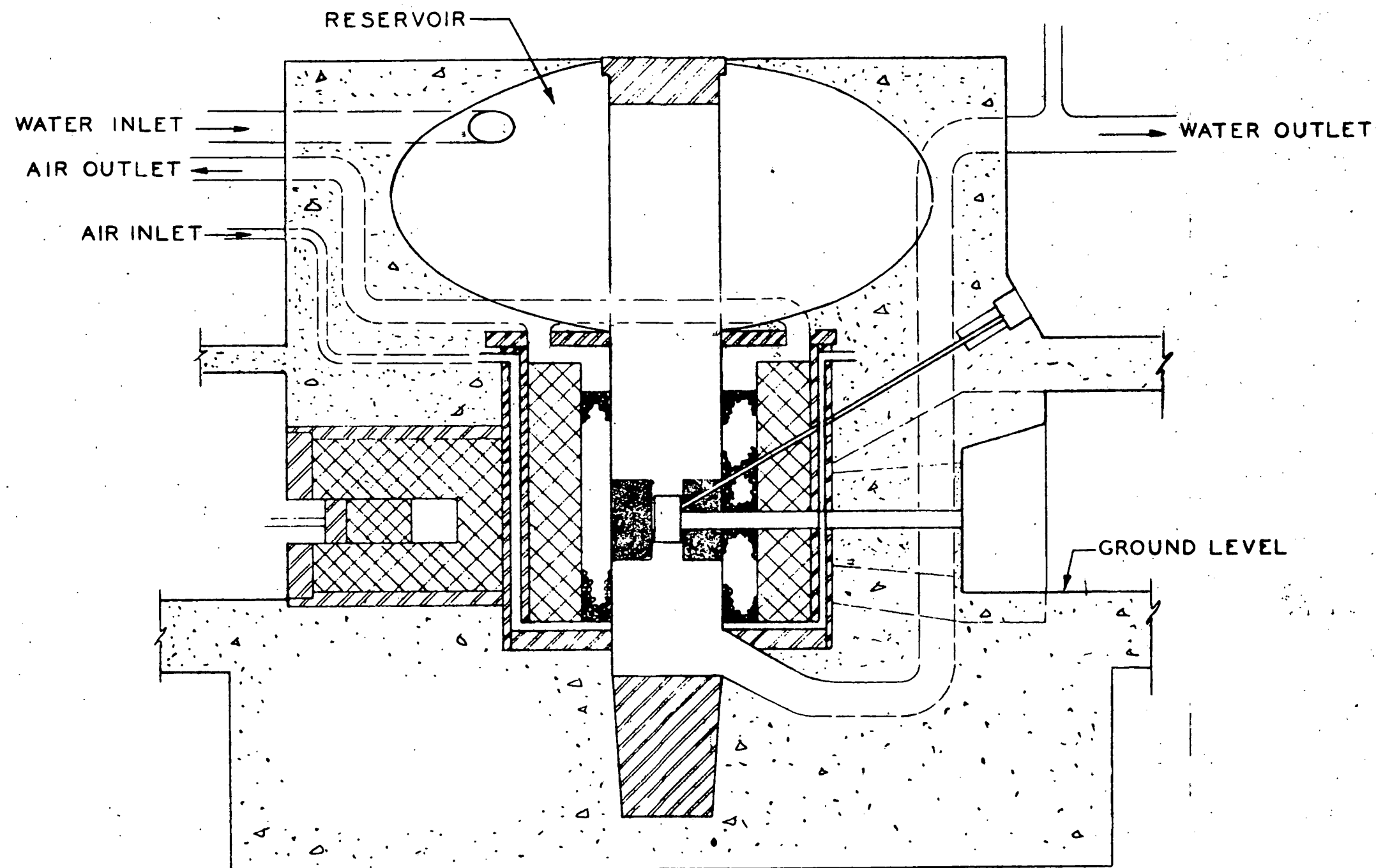
SKETCH NO. 1

1" = 5'

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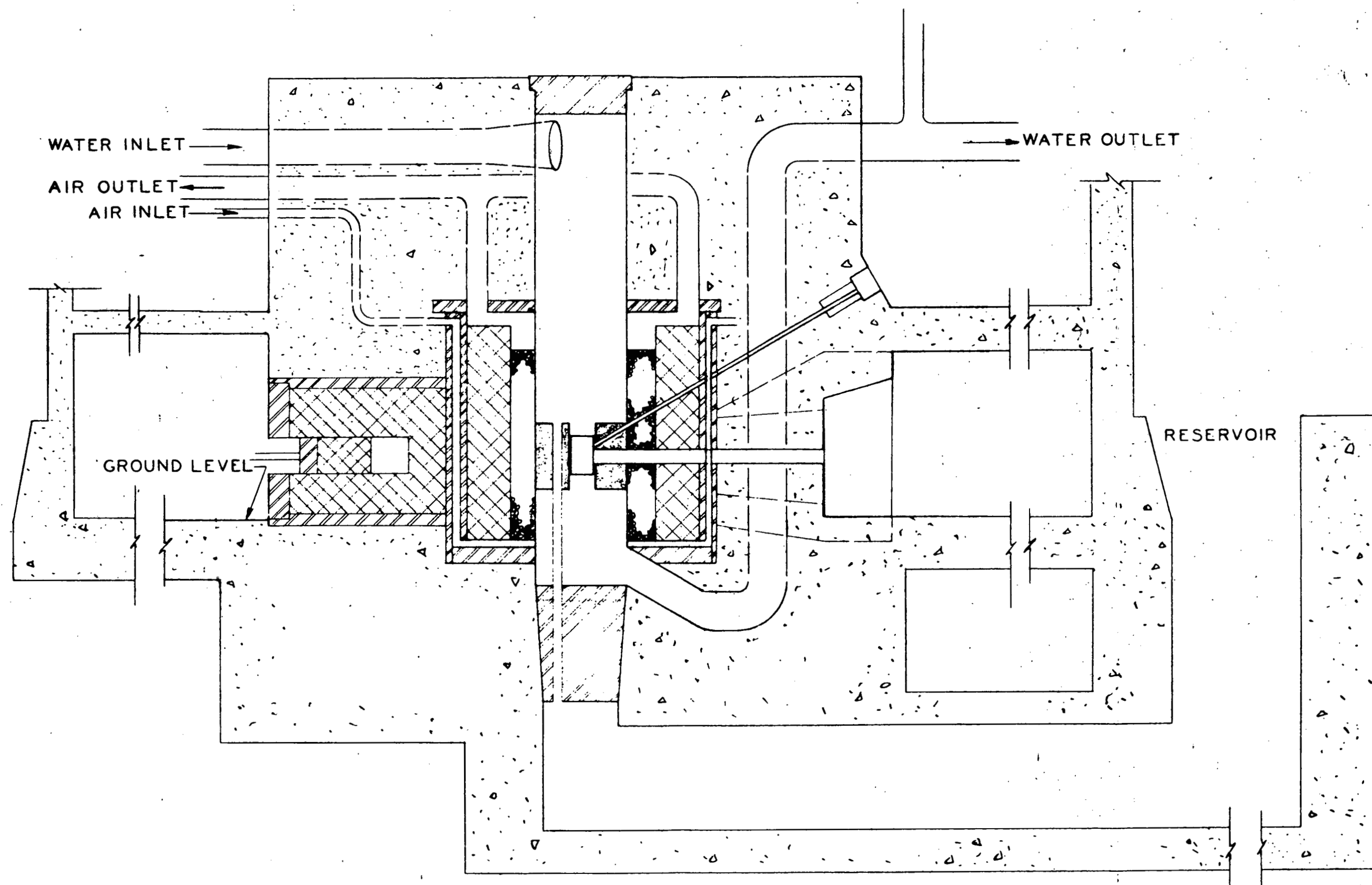
SKETCH NO. 2

1" = 5'

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SKETCH NO. 3

1" = 5'

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Heat Developed After Shutdown

In order to evaluate the heat removal problem and the hazards after shutdown it is necessary to have considerable derived data. For reference, this material is summarized in this section.

TABLE IPile Data

Number of Fuel Assemblies	22
Number of Thorium Shim Rods	4
Number of Cadmium Shim Rods	4
Volume Active Lattice	115.1 liters
Volume of Water in Active Lattice	67.5 liters
Heat Capacity Active Lattice	97,800 cal/°c
Heat Capacity Active Lattice with no water	30,310 cal/°c
Volume of Beryllium (28 liters H ₂ O)	1,360 liters
Heat Capacity of Beryllium	912,000 cal/°c
Heat Surface in Active Assembly	14,045 sq cm
Volume Water in Pile Tank	3,500 gallons
Cross section for Flow in All Assemblies (up flow)	880 sq cm
Cross section of Discharge Chute (down flow)	225 sq cm

The power evolved in the active lattice section, neglecting end losses, is

$$P = 1950 \int t^{-0.2} - (t / T)^{-0.2} \int \text{kw}$$

The total heat evolved in the pile tank is obtained by integration

$$H = 7.2 \times 10^5 t^{0.8} \text{ calories}$$

t = time after shutdown
in seconds

T = time of operation
in seconds

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TABLE II

Heat Flux and Total Heat After Shutdown

(30,000 kw)

<u>Time</u>	<u>$t-0.2 - (t/T)^{0.2}$</u>	<u>$t^{0.8}$</u>	<u>Power kw</u>	<u>Heat Flux cal/sq cm sec</u>	<u>Total Heat Calories</u>
Operation			30,000	(23.3)	
0 Sec	1		1,950	1.525	
1	0.944	1	1,840	1.43	7.2×10^5
5	0.669	3.62	1,300	1.01	
10	0.574	6.3	1,120	0.87	
30	0.45	15.2	877	0.68	
1 min	0.366	26.5	715	0.55	1.9×10^7
2	0.311	46	606	0.47	
6	0.252	111	490	0.38	
12	0.213	194	416	0.32	
30	0.167	400	326	0.25	
1 hr	0.138	700	269	0.21	5×10^8
2	0.113	1,220	220	0.17	
10	0.066	4,400	129	0.10	
1 day	0.046	8,850	90	0.07	6.4×10^9
2	0.0345	15,400	67	0.05	
4 days	0.023	26,900	45	0.03	1.9×10^{10}

One day after shutdown if heat were not removed by other means, the total heat evolved would boil 2,800 gallons of water. Table III shows that it is necessary to link the pile lattice with the water contained in the pile tank to prevent melting.

TABLE III

Temperature Rise After Shutdown

(No Cooling)

<u>Time After Shutdown</u>	<u>Lattice Only No Water</u>	<u>Temperature Rise °C</u>	
		<u>Lattice Only Plus Water</u>	<u>Pile Tank Plus Water</u>
10 sec	149	46	0.2
300	2,270	702	2.6
600		1,220	5.0
3,600		5,110	19.3

Heat Removal by Natural Convection

Exploratory calculations were made to determine the heat flux which could be handled by a thermal syphon. The heated path was considered to be the 60 cm high, 3 mm wide cooling gap in the active portion of the fuel assembly and the downward path to be the 100 cm high 6 inch square discharge chute in the beryllium reflector. These calculations neglect entrance, exit and eddy effects in the pile tank which would lower the heat removal; also the existence of other downward paths (226 sq cm in the beryllium reflector cooling holes and 67 sq cm at the tank wall) and the fact that lighter density water would persist in a column for some distance above the lattice, both of which would favor more heat removal.

$$\text{Driving Force} = \frac{\rho_1 - \rho_0}{2} h$$

$$\text{Pressure Drop in Cooling Slot} = \frac{12 \mu u h}{d^2 g} \quad (\text{McAdams, p.124})$$

$$\text{Pressure Drop in Discharge Chute} = \frac{4f h u_1^2}{2gd_1}$$

$$u_1 = \frac{880}{225} u$$

$$\text{Heat Flux} = q = \frac{3600 d \rho c_p \Delta T u}{2 h}$$

h = height in feet

ρ = density in lbs/cu ft

d = gap thickness in feet

u = velocity in feet/sec

μ = viscosity in lbs/feet sec

g = gravity in feet/sec sec

f = friction factor

q = heat flux in Btu/hr sq ft

c_p = specific heat in Btu/lb OF

TABLE IV

<u>Inlet Temperature</u>	<u>Outlet Temperature</u>	<u>q</u> <u>Btu/hr ft²</u>	<u>q</u> <u>cal/sq cm sec</u>	<u>u</u> <u>ft/sec</u>	<u>ΔP</u> <u>ft H₂O</u>
400C	800C	17,000	1.28	0.42	0.020
40	90	27,300	2.07	0.547	0.027
40	100	41,500	3.13	0.684	0.034

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By comparison with the 1.525 cal/sq cm second heat flux given in Table I, a temperature difference of about 40°C is sufficient to cool the pile at the instant of shutdown without boiling water.

In addition experiments have been performed at ANL and ORNL to confirm the heat removal by thermal syphon with and without boiling. In these experiments the return passage was sufficiently large so as not to produce appreciable additional friction.

The experimental equipment used at ORNL was designed to correspond to one fuel assembly channel and its related reactor structures as nearly as practicable. A mild steel tube with inside cross section $1/8 \times 2-3/4$ inches was mounted vertically and had a heated length of $24-5/8$ inches. An $11-3/8$ inch unheated section extended upward to simulate an upper adapter to a 3-inch glass pipe to simulate the proportional cross section of the reactor tank which, in turn, was connected to a 15-gallon head tank with the water surface 40 inches above the top end of the heated section of the tube. A line of 1 inch and $1/2$ inch pipe connected the bottom of the tank to the rectangular tube below the heated section, and a gate valve in the line was closed when it was desired to prevent outside circulation. The rectangular tube was heated by passing through it a heavy current from a variable transformer, and heat flux was determined from measurements of current and voltage drop in the tube. Thermometers to measure fluid temperatures were placed in the inlet below the heated section and at a position just above the top end of the rectangular tube. Thermocouples placed on the outside of the rectangular tube were used to measure the metal temperatures.

In the tests conducted with outside circulation, the power input was increased successively and readings were taken after steady state conditions were reached for each value of power input. No appreciable fluctuations were observed in any test, and no difficulty was encountered in reaching the capacity of the equipment. Circulation was rapid enough that the metal temperature did not exceed 100°C until a heat flux of 14,000 Btu/(sq ft)(hr) was reached, and boiling was not observed at the top of the $1/8 \times 2-3/4$ inch passage until the heat flux was increased to 25,000 Btu/(sq ft)(hr). With little or no boiling a flux of 27,500 Btu/(sq ft)(hr) was removed.

When the recirculation valve was closed so that fluid could enter and leave the channel only at the top of the tube, a similar procedure was followed except that the tank was drained and refilled each time the water became hot in order to determine the effect of the residual water temperature. Visible boiling and audible bumping occurred when the heat flux was increased to 5,200 Btu/(sq ft)(hr), and all temperatures fluctuated several degrees during the remainder of the tests. The power input was increased successively until a "hot spot" developed, indicating that vapor binding was occurring and that excessive metal temperatures would result. The maximum heat flux

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achieved with hot water in the tank was 17,000 Btu/(sq ft)(hr) and the maximum heat flux with cold water in the tank was 19,750 Btu/(sq ft)(hr). Rough calculations indicated that 5-20% steam formation by volume would remove 20,000 Btu/hr ft².

At the ANL an aluminum section was fabricated to simulate a section of the high flux pile assembly representing the equivalent of three water gaps. This unit was suspended vertically in a 105 gallon water tank. The plates were heated by the passage of low voltage AC current. Power was measured by a wattmeter on the primary of the power supply transformer. Sufficient power was available to run a unit until the plates warped.

The first element tested had been air-cooled at 0.25 cal/cm² second and the plates were slightly warped. This element was used to test equipment and calibrate meters. It was run for about 10 minutes at 44 kw (8.7 cal/cm² sec). It failed immediately at 90 kw (18 cal/cm² sec) and inspection showed that the plates were severely warped.

The second unit was operated for 20 minutes at 48 kw (9.6 cal/cm² sec). At the suggestion of Dr. Teller, the power was increased to 82.5 kw (16.5 cal/cm² sec), and the unit was operated at this power until the water temperature had risen to 96°C. This required about 20 minutes. Boiling was violent at first, but when the water temperature rose to about 90°C the churning sound disappeared and was succeeded by a slight hissing. Current was shut off momentarily when the water reached 96°C. The unit failed when the current was turned on again.

Finally, at ANL, a complete high flux pile fuel assembly (18 plates) was suspended in the 105 gallon tank. The assembly was surrounded by 2-inch thick wood for insulation. It was operated for 30 minutes at 69 kw and then for two minutes at 86 kw without a sign of failure. Output of the transformer was limited by electrical power lines. These power inputs correspond to 1.1 and 1.4 cal/sq cm second respectively. The temperature of the water in the tank was not over the boiling point.

Air Cooling After Shutdown

Calculations were made to determine how soon after shutdown natural convection of air could handle the heat flux.

Combining the proper equations given above and neglecting both friction outside the cooling gap and entrance and exit losses, the following equation can be obtained.

$$q = \frac{3600 \text{ g d}^3 \rho \cancel{\Delta p} c_p \Delta T}{48 \cancel{h}}$$

ΔT = temperature difference in air in $^{\circ}\text{F}$

The driving force is

$$\Delta p = 0.073 \frac{\Delta T}{546 \cancel{f} \Delta T}$$

For the 3 mm gap, 30°C inlet air

$$q = 0.11 \frac{\Delta T^2}{546 \cancel{f} \Delta T}$$

$$u = 0.667 \frac{\Delta T}{546 \cancel{f} \Delta T}$$

$$\Delta T = 6.1 \frac{q}{u}$$

The film drop is found from

$$\frac{h D_e}{k} = \phi \frac{(D_e G c_p) (D_e)}{(k) (L)}$$

(McAdams
p. 199)

TABLE V

Air Cooling After Shutdown

ΔT	q Btu/hr ft ²	q cal/sq cm sec	u ft/sec	t_m
200°C	15.6	0.001	0.265	295°C
600	79	0.006	0.442	903
1,000	152	0.0115	0.512	1,540

If the metal temperature is to be kept at 600°C, i.e., below the melting point of aluminum, 0.0036 cal/sq cm second is the maximum heat flux which can be handled. This is reached at 50 days after shutdown for the pile operating at 30,000 kw. Air cooling after shutdown is definitely not possible for the 3 mm cooling gap and 30,000 kw operation. If operated at 3,000 kw this flux is reached after three days cooling.

The effectiveness of air cooling increases as the cube of the width of the cooling gap. The following table gives the heat flux possible as a function of the gap again maintaining the metal temperature at 600°C.

If a wider gap is defined as

$$d_1 = bd$$

these formula become

$$q = \frac{0.11 \Delta T^2 b^3}{546 \Delta T}$$

$$u = 9.14 b^2 \Delta \rho$$

The film drop is independent of b.

TABLE VI

Air Cooling vs Gap Width

Gap Width	q cal/sq cm sec
3 mm	0.0036
6	0.0290
7.5	
9	0.098
10	
12	0.232
15	0.454

These calculations do not take advantage of the loss of heat by radiation and conduction to the pile beryllium and pile tank walls. For operation at 30,000 kw this might reduce the heat flux by 10%; for operation at 3,000 kw the factor might be 50%.

Using twenty-two assemblies, increasing the plate thickness and decreasing the number of plates per assembly, the heat fluxes existing in the pile at shutdown for various power levels were estimated and are shown in Table VII. Since more gamma energy will escape the pile lattice if no water is present, the power and total heat were taken as follows:

$$P = 0.05 P_0 t^{-0.2} \text{ kw sec}$$

$$H = 4.5 \times 10^5 t^{0.8} \text{ calories}$$

TABLE VII

Effect of Plate Spacing and Power Level
on Air Cooling

Case No.	Plates per Assembly	Plate Thickness	Cooling Gap	Heat Flux Removed by Air cal/sq cm sec	Heat Flux at Shutdown cal/sq cm sec		
					5,000 kw	3,000kw	2,000kw
1	18	0.15	0.3	0.002	0.182	0.11	0.072
2	9	0.20	0.6	0.016	0.36	0.22	0.146
3	5	0.3	1.1	0.098	0.65	0.39	0.26
4	3	0.5	1.8	0.43	1.09	0.65	0.43

In Table VIII the time in which the pile would reach the melting point of aluminum, assuming no other loss of heat, is compared with the time after shutdown necessary to reach establishment of a steady state for air cooling.

TABLE VIII

Time Comparison for Air Cooling

Case No.	Time After Shutdown to Reach 600°C, No Cooling Seconds			Time After Shutdown to Reach Steady State with Air Cooling Seconds		
	5,000kw	3,000kw	2,000kw	5,000kw	3,000kw	2,000kw
1	730	1,400	2,300	-	-	-
2	460	900	1,450	-	-	-
3	400	780	1,250	13,000	1,000	136
4	400	780	1,250	100	7.6	1

From this table a pile operating at 3,000 kw with 3 mm thick plates, and 1.0 cm cooling gaps would be safe with only air cooling after shutdown.

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APPENDIX D

"Materials Testing Reactor Design Data Sheets"

by D. J. Mallon dated October 28, 1949

ORNL CF# 49-10-232

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To: File
From: D. J. Mallon
Date: October 18, 1949
Subject: Materials Testing Reactor
Design Data Sheets

Oak Ridge National Laboratory
Central Files Number
49-10-232

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MATERIALS TESTING REACTOR

Light Water Moderator - Beryllium Reflector

Fifth Edition

October 18, 1949

Physical and Pre-Design

(see MonP-206, 272,357,433:ORNL 79,167)

1.0 ENGINEERING DESIGN

1.1 Pile Structure

Height above floor level	24 ft
Overall cross section of concrete	34 ft x 32 ft 6 in.
Balcony level	11 ft
Height of Basement : in building	13'6"
in sub pile rm,	8'6"
Centerline of Pile above floor level	3 ft-6 in.
Centerline of horizontal beam holes above floor level	3 ft-6 in.
Water depth in discharge canal	17 ft
Clearance to crane hook	28 ft
Thickness of Aluminum tank	3/4 in.
Design cooling water flow	19,500 gpm
Purge of cooling water	.3%

1.2 Reactor

1.21 General Data

Operating output	30,000 kw
Normal pile loading (23 fuel assemblies no shim-rod contribution)	3.3 kg 235
Max. possible loading (37 fuel assem- blies)	5.3 kg 235
Number of lattice spaces	45
Number of shim-safety rod spaces	8
Heat fluxes	
Average	294,000 BTU/hr.sq.ft.
Center of Pile	367,500 BTU
Maximum	533,000 BTU

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Heat Transfer Coefficients Film (30 ft/sec) 6,300 BTU/hr.sq.ft.°F
 Scale 25,000 "
 Overall 5,100 "

Temperatures: July average wet bulb(ARCO) 65°F
 Pile cooling water inlet 100°F
 Pile cooling water outlet 110.2°F
 Max Temp drop across film 97°F
 Max Aluminum metal temperature 223°F

Water velocity in active portions 30 ft/sec
 Pressure drop across pile 35 psi
 Cooling air require'ts @ (70°F & 25.5 cm. Hg) Atmos. cond. 25,000 CFM
 Temperature of graphite pebbles (max) 410°F @ 30,000 kw
 570°F @ 45000 kw
 155°F @ 30000 kw

Exit air temperature 30-40 ins. H₂O
 Pressure drop (thru pebbles) 20.3 mg/sq cm

1.22 Fuel Assembly: 25 per unit area
 % composition starting "25"-93.4%
 28 - 5.5
 24 - 1.1
 % composition rejection 25 -60.3%
 26 -23.7%
 24 & 28 -16

Per cent 25 Consumed before rejection 73.9%
 Maximum Uranium in alloy 22 % by weight
 Ratio: VA1/VH₂O (Active lattice) 0.70/1.00
 25 per active assembly 140. gms

Total heat transfer area per assembly 14,045 sq cm
 Number of active plates 18
 Water gap between active plates .117

1.23 Reflector (Be)

"A" Piece cross section 3.018 x 3.018 in.
 cooling water passages 1/8 Dia holes-1 in. centers
 Ratio V_{H₂O}/ V_{Be} 0.02

Removable lattice pieces-cross section 3.015 x 3.150 in.
 cooling water passages 1/8 Dia holes-1 in.center
 Ratio V_{H₂O}/ V_{Be} 0.02

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1.3 External Reflector

Graphite pebble bed	E-W & N-S(outside)	55-3/4 inside dia
Permanent graphite	E-W & N-S(inside)	7 ft-4in. x 7 ft-4in.
	E-W & N-S(outside)	7 ft-4 in x 7 ft-4 in.
		12 ft x 14 ft
	Height	9 ft - 4 in

1.4 Shielding1.41 Thermal Shield

Composition	Iron
Thickness	8", 2-4" layers with 3" air gap
Inside dimensions	12 ft-1" x 14 ft 1"
Height	12 ft - 10 in.
Distance from centerline of pile	6 ft-0 $\frac{1}{2}$ " and 7 ft -0 $\frac{1}{2}$ in.

1.42 Biological Shield

Composition	Concrete
Thickness (Minimum of Portland Mix)	9 ft

1.43 Top Shielding

Composition (separate) Water	19 ft-10 in. H ₂ O above pile centerline
Plug: 3 $\frac{1}{2}$ " SS	
7" Lead	10 $\frac{1}{2}$ in (total)
Composition at shutdown	18 ft-10 in. H ₂ O above CL

1.44 Bottom Shielding

Composition (separate) Water	8 ft below pile centerline
Plug: 1" SS	
13" Lead	
2" SS	16 in (Total composition)

2.0 Nuclear Data2.1 Neutron Fluxes (nv)

Maximum Virgin Flux ($\lambda_t = 5$ cm)	1.2 x 10 ¹⁴ n/sq.cm.sec.
Maximum Epicadmium Flux	6 x 10 ¹⁴ "
Average Slow Flux(Active Portion, no Thorium)	2 x 10 ¹⁴ "
Maximum Slow Flux(In Be Reflector, no Thorium)	5.2 x 10 ¹⁴ "
Slow Flux at Center of Slab Pile	2.5 x 10 ¹⁴ "

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Nuclear Data (cont'd)

Max. Slow flux at Al - Graphite Ball Interface 34 cm Be 5.2×10^{13}
 50 cm Be 3.5×10^{13}
 Slow Flux at Entrance to Thermal Column 9×10^{11}
 Slow Flux at Face of Thermal Column (with Bore Shield) 6×10^7
 Slow Fluxes in Thorium Production Rods 5×10^{13}
 Neutron Fluxes above and below Pile measured from
 Lattice Surface

Distance Cm	Total Incident N/sq cm/sec	Epithermal N/sq cm/sec
100	1×10^9	6×10^8
150	5×10^7	2×10^7
200	3×10^6	1×10^6
300	3×10^4	1×10^4
400	7×10^2	3.5×10^2

k of cold, clean pile 1.614
 Δ k/k for Xe poisoning (operation) -0.039
 Δ k/k for Sm poisoning (operation) -0.010
 Δ k/k Xe (maximum @ 11 hours after shut-down) -0.435
 Δ k/k Sm (@ 11 hours after shut-down) -0.012 (0.033 Max)
 Δ k/k 25 Depletion (1% of 25 per day) -0.0033/day
 Δ k/k Temperature (maximum) -0.008
 Δ k/k Thorium -0.007
 Δ k/k allowed for experiments -0.050
 Δ k/k allowed in Design (total) 0.190
 k of pile at end of operating cycle (288 hrs) 1.31
 Thermal Utilization 0.77
 Minimum Critical Mass 0.95 kg
 Average energy of Pile Neutron 1 Mev
 Neutrons escaping lattice per fission $0.9(k=1.6)0.67(k=1.37)$
 Neutrons absorbed in aluminum and water
 per fission 0.40

2.2 Neutron Properties

Material	Diffusion Area (L^2) cm ²	Neutron Age (τ) cm ²	Transport Mean Free Path	
			Avg. Slow cm	Avg. Fast cm
In Pile	3.32	60.3	0.8	3.73
In Water	8.3	33	0.5	3.43
In Be(2% H ₂ O by vol)	400	96	1.9	2.01
In H ₂ O - Al	15.6	60.3	0.75	3.73
In Thorium Production Rod	7.3	-	3.9	-

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2.2 Nuclear Data (cont'd)

	<u>Diffusion Area</u>	<u>Neutron Age</u>	<u>Transport Mean Free Path</u>
In Thorium Shim Rod	5.1	59.6	0.83
In Graphite	2500	350	2.71
In Pure Be	625	98	2.3
In Aluminum	296	10,400	12
In Bismuth / 10% H ₂ O	-	224	-
In Bismuth / 30% H ₂ O	-	76	-
			3.80
			3.34
			1.89
			8
			4.23
			4.25

2.3 Gamma Ray Spectrum

Source Material	γ /fission Mev	Energy/ γ Mev	Energy/fission Mev	avg.watts/cc
Fission of U ²³⁵				
Prompt Fission	5	1	5	8.4
Fission Product	2	2.5	5	8.4
Capture in "25"	0.18	6.0	1.1	1.02
Capture in Al	0.11	8.0	0.9	1.48
Capture in H ₂ O	0.24	2.2	0.5	0.88
Capture in Th	0.08	6.0	0.5	0.46
		Total	13.0 Mev	20.6 Total

3.0 Pile Control3.1 Regulating Rod

Composition	Cd-Al.
Total weight of combined rods	55.5 pounds
Buoyant force of regulating rod	8.1 pounds
Total buoyant force	17.9 pounds
Resistance force of rod thru pile section (friction)	7.4 pounds
Acceleration	~1.3 g
Accelerating force required at rack to raise rod (combining only mass of rod & g)	~130. pounds
Total Force required at rack to raise rod	~111 pounds
Accelerating force required to lower rod	
Total force required at rack to lower rod (combining only mass of rod & g)	~16.8 pounds
	~71 pounds
The horsepower required to move the rod at four feet per second within 1/10 second has been estimated to be approx. 20 per cent of the total load and the total horsepower required is estimated at 5 hp.	
Factor of safety of regulating rod	2.8
Factor of safety of drive rod	5.5
$\Delta k/k$ per rod	0.005

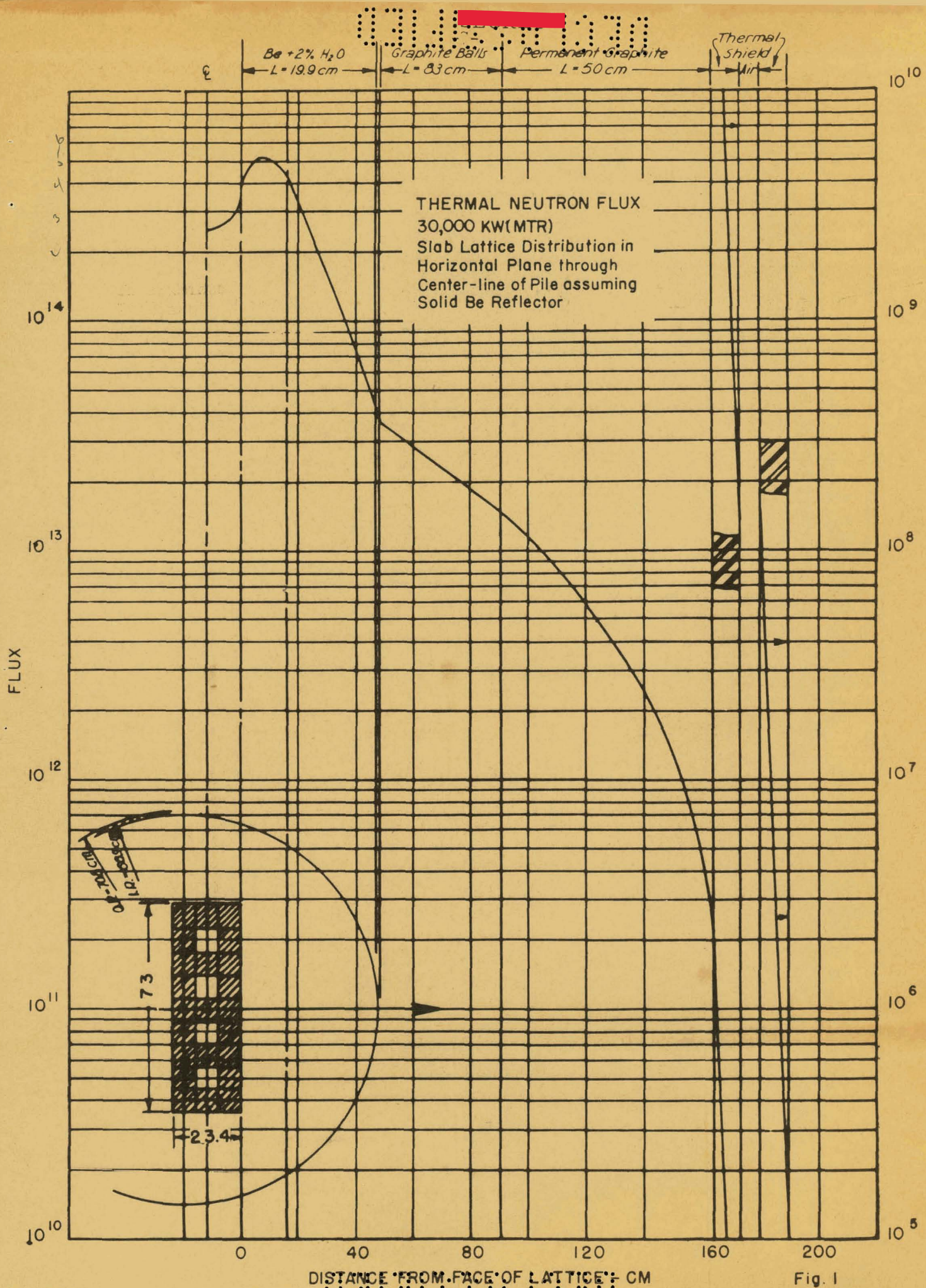


Fig. 1
 Dr. 7948

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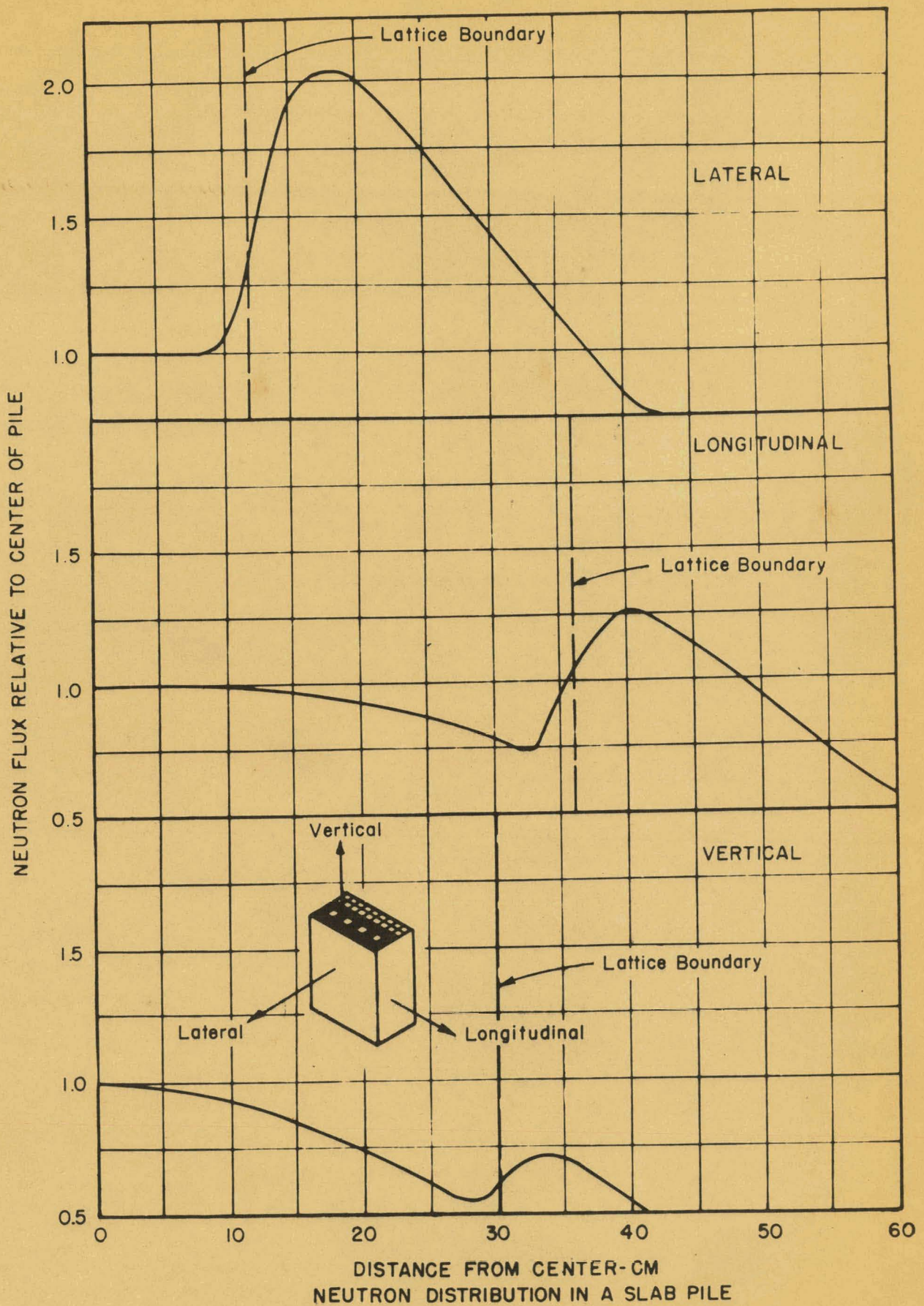


Fig. 2

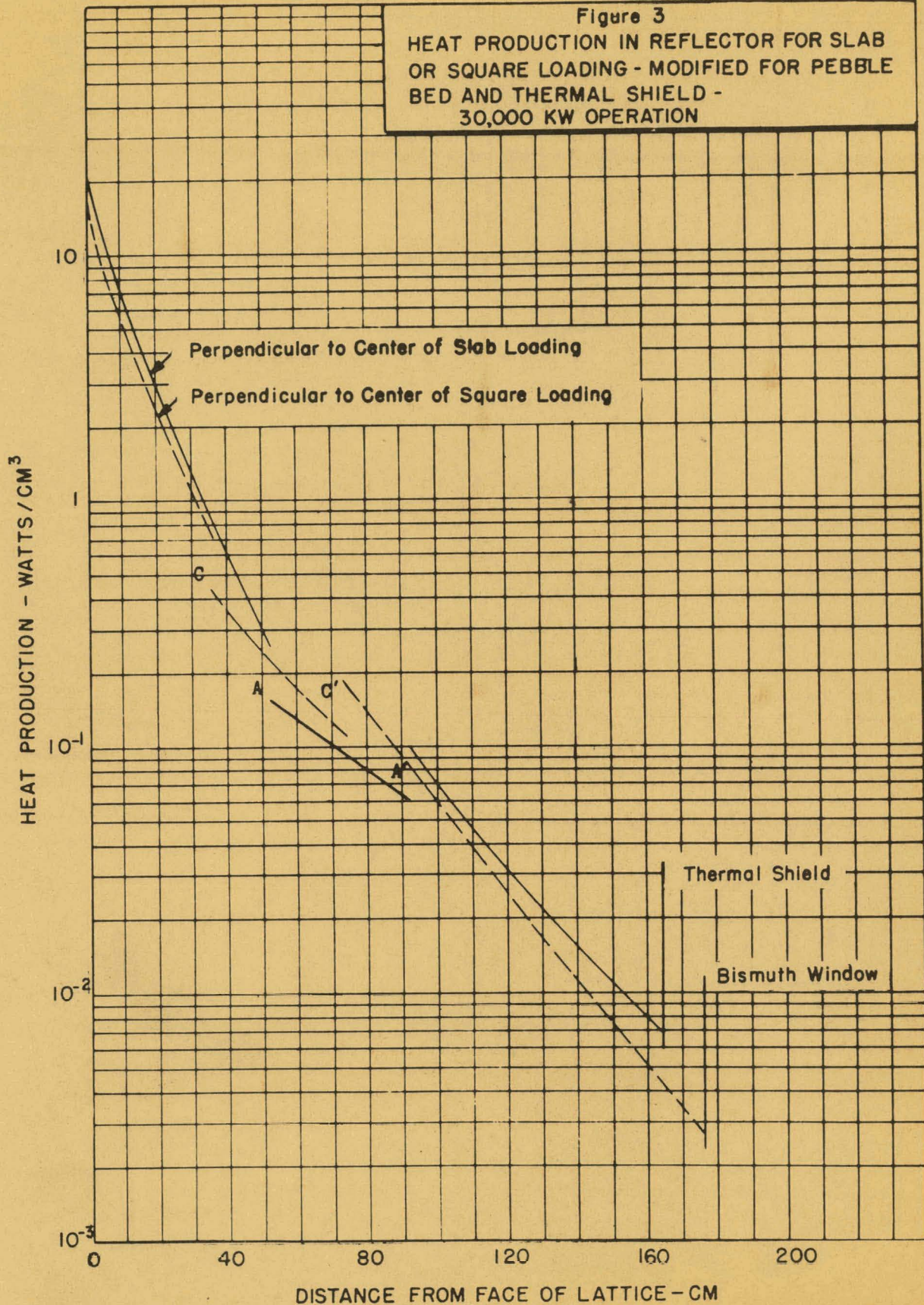
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Figure 3
HEAT PRODUCTION IN REFLECTOR FOR SLAB
OR SQUARE LOADING - MODIFIED FOR PEBBLE
BED AND THERMAL SHIELD -
30,000 KW OPERATION



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Regulating Rod (cont'd)

Force Acting for Safety

365#

Acceleration required for safety

1 G in first 3" of travel

3.2 Shim-Safety Rod--General Data

Composition	Th-25 and Cd-25
Max. No. of Rods	8
Max. Speed of Travel (up and Down)	0.2 cm/sec (0.0798 in/sec)
Minimum acceleration of fall	1 gravity
Total travel	2 ft. 6-7/8 inches
Permissible overtravel upward	1/2 mm (0.0197) inches
Permissible overtravel downward	2 mm (0.078 inches)
Length of Thorium, Cadmium or U. Sections	60 cm (23-5/8 inches)
External Dimension, Cross Section	7.21 x 7.8 cm
Thickness of Thorium plate	5.1 mm
Number of Thorium Plates	6
Water Gap in Thorium Plates	9.48 mm
Number of Uranium Plates	14
Thickness of Uranium Plates	0.060 inches
Thickness of Uranium End Plate	0.241 inches (1.55 cm)
Space between Plates	0.113 inches
$\Delta k/k$ for Thorium Section	0.04 per rod
$\Delta k/k$ for Cadmium Section	0.07 per rod
Magnetic release time	0.03 second

3.21 Thorium-Uranium Rod

Weight of rod	132 lbs.
Weight of magnetic coupling	40 lbs.
Total weight-rod, shaft, coupling	207 lbs.
Water pressure load on rod	290 lbs.
Weight of rod and water pressure minimum brg. friction	352 lbs.
Load on Magnet (rod being withdrawn)	467 lbs.
23 Production in Thorium (max)	2.85 g/day or 0.08 atoms
	Th per atom 25 destroyed

3.22 Cadmium-Uranium Rod

Weight of rod	135 lbs.
Total weight-rod, drive shaft, coupling	210 lbs.
Water pressure load on rod	290 lbs.
Load on magnet (rod being withdrawn)	470 lbs.

3.3 Shock Absorber

Length of Shock Absorber	8 inches
Maximum Downward Acceleration of Cadmium Rod	2.7 g
Max. velocity of rod in fall at 35 psi press.	
drop	15 ft/sec.
Maximum force to stop rod	1274 lbs.
Pressure developed in dash-pot	175 psi

4.0 EXPERIMENTAL FACILITIES

4.1 Available Flux (nv 10^{14}) (Fast and Thermal Neutron)

- 4.11 Active Lattice Hole - A 3" x 3" vertical hole through a shim rod location in the active lattice.
- 4.12 Horizontal Through Hole - A 5" x 5" horizontal hole extending from side to side of reactor adjacent to active lattice.
- 4.13 Horizontal Beam Holes - Six - 6" dia. (at inner ends) horizontal holes penetrating permanent Be to lattice, two of these to be 8" I.D. outside the reactor tank.
- 4.14 Horizontal Pneumatic Rabbits - Two 1" I.D. rabbits penetrating permanent Be near Active lattice edge.
- 4.15 Vertical Hydraulic Rabbits - Four holes situated in permanent Be near lattice edge and penetrating vertically through bottom shield; two of these are 1" I.D. and two are $1\frac{1}{2}$ " I.D. (The latter are regulating rod spaces not in use).
- 4.16 Replaceable Beryllium Positions - At least forty sections of the Be reflector are removable without appreciably reducing flux values for other experiments.

4.2 Available Flux (nv 10^{13}) (Thermal Neutrons)

- 4.21 Horizontal Neutron Channel - A 10" wide x 16" high approach hole to the tank wall. Actual neutron channel comes from 5" x 12" tunnel through cans in Be to face of active lattice. Vertical access to hole close to the aluminum tank from the top of pile structure.

- 4.22 Angular Beam Holes - Six downward sloping 2" I.D. holes in the permanent Be that do not pierce tank. Access through 6" I.D. holes to outside of reactor tank.
- 4.23 Oriented Sample Holes - Two 4" dia horizontal holes penetrating to tank wall.
- 4.24 Pebble Zone Holes - Six 4" dia. vertical holes penetrating from top of reactor to bottom of pebble zone. Two of these holes actually penetrate through to the canal.
- 4.25 Inclined Pebble Zone Holes - Thirteen 2½" dia. inclined holes, slightly offset from vertical, penetrating from top of reactor to pebble zone near tank wall.
- 4.26 Vertical Instrument Holes - Six 2½" dia (actually inclined) holes penetrating from top of reactor to tank wall. Two of these holes terminate in the oriented sample holes and two terminate in holes described in 3.27.
- 4.27 Horizontal Approach Holes - Two 6" dia. horizontal holes, approaching tank wall at angle of 30° to N-S centerline, and penetrating to tank wall.

4.3 Available Flux (nv 10¹²)

- 4.31 Test Holes - Approx. thirty-seven 2" dia vertical holes penetrating from top of reactor to bottom of permanent graphite.
- 4.32 Horizontal Through Holes - Two 8" dia. and two 4" dia. holes extending from side to side of the reactor and penetrating through the permanent graphite.
- 4.33 Horizontal Instrument Holes - Two 10-3/4" I.D. horiz. holes extending from side to side of the reactor (parallel to N-S centerline) and penetrating through the permanent graphite.

4.4 Available Flux (nv 10¹⁰)

- 4.41 Vertical Thermal Column Holes - Two 1 ft x 1 ft holes penetrating from top of reactor to and including the Horizontal Neutron Channel (see 3.21 above).
- 4.42 Horizontal Thermal Column Holes - Two 4" x 4" horizontal holes extending from side to side of the reactor through the Thermal Column. Six horizontal holes extending to thermal shield.

- 4.5 Available Pure Gamma Flux (10⁶) - Two vertical holes penetrating from top of reactor to a point 10 ft. above pile ϵ and into the pile exit water line cross-overs (one per cross-over.)

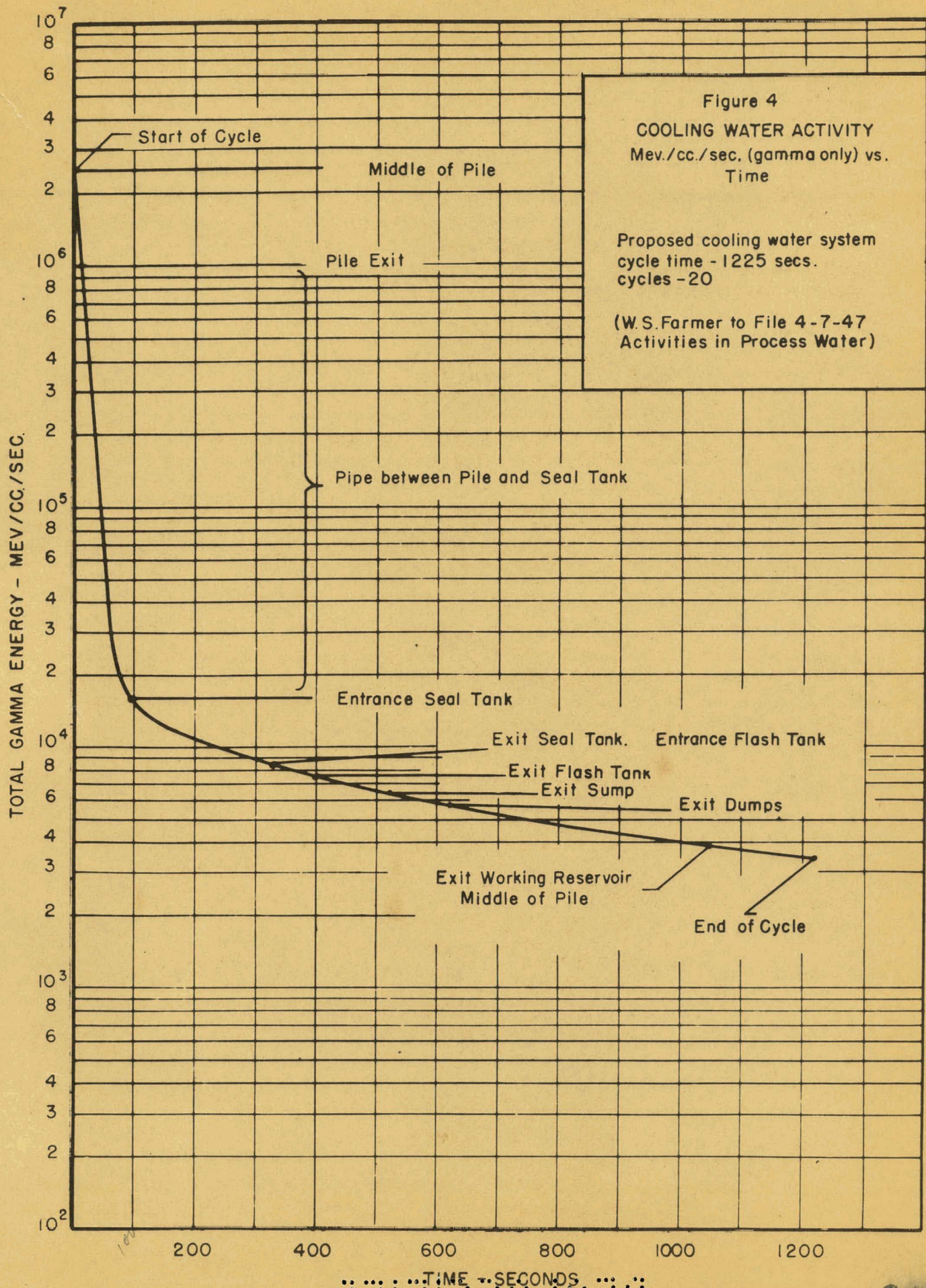
5.0 Miscellaneous Data

5.1 Health Hazards

Tolerance for areas 1 cm²
(total additive)

0.1 r of β , γ /24 hrs.
0.025 roentgen equivalent
physical of f.n., 200 n/cm²
sec. 1,500 n/cm² sec. th.
neuts.

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Tolerance for areas 1 cm²

Gases

River

Normal Background

0.1r/8 hrs for beams
on average

General Tolerance

dusts 3.1×10^{-11} c/cc airNa²⁴ 6.3×10^{-14} c/cc air0.5 μ c/liter water

0.015 mr/hr

5.2 Induced Activities in Pile Structure (3 months operation) (Values to be used for Major Overhaul)

		Doses at top of well <u>No Water</u>
Stainless Steel Bottom Tank	90 millicuries	
Stainless Steel tank section	25 curies	2.5
Upper Support Ring	1.5 curies	9.0
Beryllium Reflector	2600-10,000 curies	250-1000
Aluminum Tank	500 curies	
Replaceable Graphite Ring (no self absorption)	60 curies	
Total Graphite Reflector	500-5,000 curies	8.5
Magnets	3.7 curies total	

6.0 RESUME' OF DATA SHEETS ISSUES

6.1 Issue No. 1 Memo to file by John R. Huffman June 1, 1948
ORNL CF 48-6-68

Issue No. 2 Memo to Steering Comm by J.R.H. April 1, 1949
ANL-ABS-3

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APPENDIX E

"Reduction in Reactivity of Heterogeneous Reactor by
Expulsion of Coolant," by S. Untermyer ANL-WPB-60

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ANL-WPB-60

Reduction in Reactivity of Heterogeneous Reactor

By Expulsion of Coolant

Introduction

Several recent reports (A) (B) (C) have questioned whether plate type heterogeneous water-cooled and moderated reactors fail safe in event of a major reactivity excursion. As it seems difficult to calculate accurately whether the moderator can be expelled or heated quickly enough to stop the pile before the fuel plates melt, we have done experiments which indicate that expulsion of the coolant starts within 30 milliseconds after application of heat. In most thermal reactors, this appears to be fast enough to prevent melting of fuel plates.

Apparatus (Figure 1)

Heat was applied to a thin-walled Al tube by the momentary passage of a large, low voltage alternating current. Current was obtained from the one turn secondary coil of a small transformer. The 220v ac primary input was controlled by an electric timer, so that the duration of heat input could be varied.

The sample was an Al tube, about 15 cm long, 1 cm ID and, originally, about 50 mil wall. The central heated section of the sample was machined to 5 mil wall thickness. The sample was mounted vertically so that the tube could either be empty or filled with stagnant water.

One pen of a Brush oscilloscope was connected to the heating circuit. The other pen was actuated by immersion of two probes in water expelled from the tube. Thus, a permanent record was obtained of heating time and of the time delay in expelling water.

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A low heat capacity thermometer was used to measure the initial and final water temperature.

Data

The minimum time to melt the tube wall was determined by applying successively longer impulses until the tube melted.

<u>Condition</u>	<u>Time to Melt Tube (cycles = $\frac{\text{sec}}{60}$)</u>
1" long tube - no water	$5\frac{1}{2}$
1" long tube - filled with water	11
2" long tube - no water	11
2" long tube - filled with water	> 31

The water temperature rise was measured for a two inch long tube. The average value was:

0.9°C/cycle

or

54°C/sec

The time delay in start of expulsion of water was measured on the oscilloscope tape.

<u>Tube Length</u>	<u>Time Delay (cycles = $\frac{\text{sec}}{60}$)</u>
1"	$1\frac{1}{2}$
2"	7

Typical oscilloscope records are shown on Figure 2.

Calculations

A. Heat Input

From "Metals Handbook" we find that the heat required to heat and fuse 20°C aluminum is 256 cal/gr or 692 cal/cc.

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Oxidation of Uranium and Zirconium - Uranium Alloy at
1000° C. in Oxygen

U. Metal

49% Zr, 51% U Alloy

90% Al, 10% U Alloy

T increased to 1150°C.

mm penetration of oxide coating

Time in Minutes

0.20

0.00

0.0

10

20

30

40

50

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A 5 mil tube wall contains $0.012 \frac{\text{cc}}{\text{cm}^2}$ metal. Thus, to melt such a wall requires $0.012 \times 692 = 8.3 \text{ cal/cm}^2$ of heat.

The burnout time for the 1 inch tube was $5\frac{1}{2}$ cycles or 0.092 sec, corresponding to a heat input of $\frac{8.3}{0.092} = 91 \text{ cal/cm}^2 \text{ sec}$. The 2 inch tube had a burnout time of 11 cycles, corresponding to a heat input of $\frac{8.3}{0.183} = 45 \text{ cal/cm}^2 \text{ sec}$. The over-all tube length was 15 cm and the ID was 1 cm so the water volume was 12 cc. Thus the heat input deduced from the temperature rise of the water in the 2 inch tube was $54 \times 12 = 650 \text{ cal/sec}$. The heated surface area was 16 cm^2 so the heat input was $\frac{650}{16} = 41 \text{ cal/cm}^2 \text{ sec}$. Considering heat losses and other uncertainties, this seems a reasonable check on the value of $45 \text{ cal/cm}^2 \text{ sec}$ calculated from the time required to melt the Al.

B. Heat Transferred to Coolant

At the time the 1 inch water-filled tube melted, 8.3 cal/cm^2 must have been retained in the metal wall. However at this time heat had been generated for 11 cycles (0.183 sec) at a rate of $91 \text{ cal/cm}^2 \text{ sec}$. This corresponds to $16.7 \frac{\text{cal}}{\text{cm}^2}$. Thus, $16.7 - 8.3 = 8.4 \text{ cal/cm}^2$ must have been transferred to the water in 0.183 sec, giving an average heat transfer rate of $46 \text{ cal/cm}^2 \text{ sec}$ to the water.

C. Rate of Steam Formation

It requires about 600 cal of heat to convert one cc of 40°C water to 1700 cc of 1 atmosphere steam. With a heat flux of $40 \text{ cal/cm}^2 \text{ sec}$ steam is generated at the rate of

$$\frac{40}{600} \times 1700 = 110 \text{ cc/cm}^2 \text{ sec}.$$

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Thus, the steam generated in 10 milliseconds would fill about 10 per cent of a 1 cm cooling passage.

Reactivity Decrease in a Heterogeneous Reactor

The diffusion area (L^2) is given by $\frac{\lambda_a \lambda_s}{3}$. λ_a depends principally on fuel concentration, while λ_s depends on moderator concentration. Thus, expulsion of 10 per cent of the moderator has about the same effect on diffusion area as removing 10 per cent of the fuel. In addition, τ depends on λ_s^2 ; thus it is seen that expulsion of (say) 10% of the moderator increases L^2 by 10% and increases τ by 20%. For enriched piles, where $k-1$ is over $1/2$, this will cause reactivity reductions of about 5%; which will promptly stop the chain reaction.

For reactivity excursions beyond prompt criticality, the pile period is approximately $\frac{\text{prompt neutron lifetime}}{\text{excess } k \text{ beyond prompt}}$

A prompt lifetime of 1 millisecond and an excess k of 2% gives a period of 50 milliseconds, which is appreciably longer than the time required to expel water from the reactor.

S. Untermeyer

References

- A. - NAA-SR-31. "A Study of Reactor Hazards" - by M. M. Mills.
- B. - NAA-SR-34. "Design Study of Low Power Research Reactor" by A. S. Thompson and T. Fahrner.
- C. - NAA-SR-47. "Transient Heat Transfer in Heterogeneous Reactor" by W. C. Parrish.

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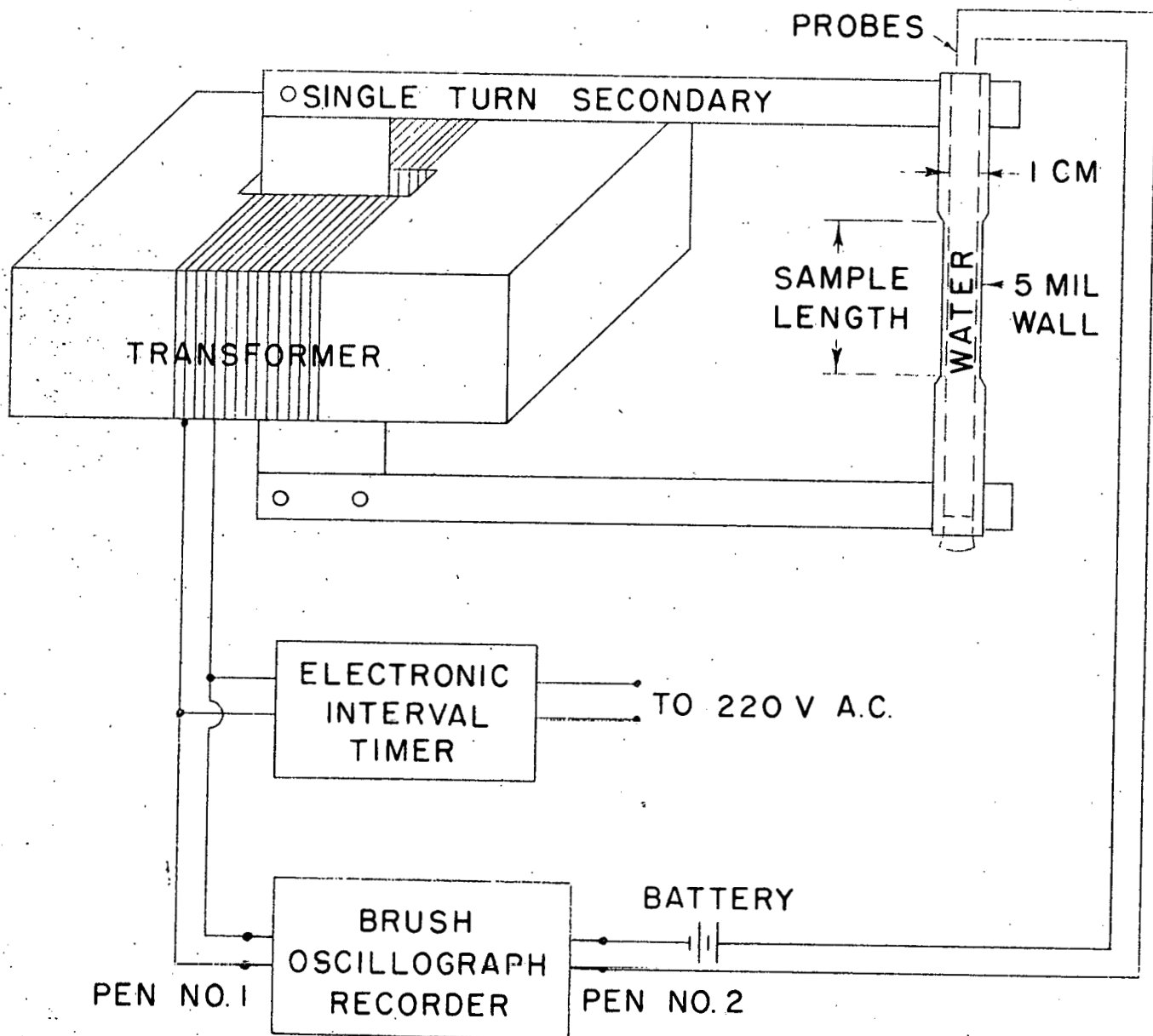
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APPARATUS FOR STEAM FORMATION EXPERIMENT



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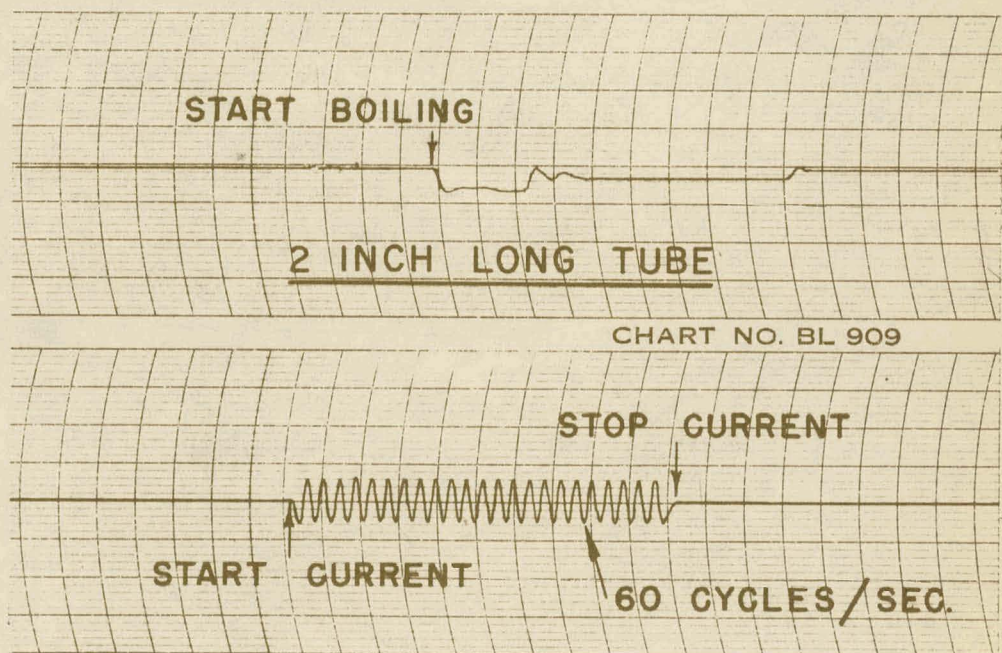
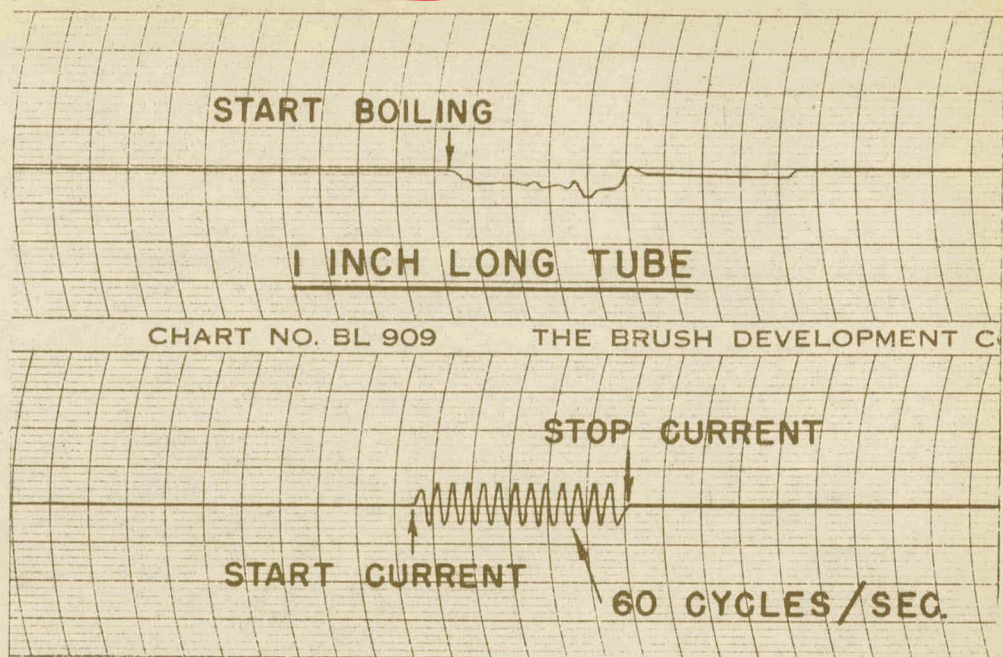
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FIGURE 1

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TYPICAL RECORDS OF EXPULSION OF
WATER FROM HEATED 5 MIL WALL 1 CM
I.D. ALUMINUM TUBES.

No: RDG-36
DATE:
TIME:
DATE:

FIG. 2

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APPENDIX F

"Reactor Hazards," by J. J. Katz and H. R. Hoekstra
dated January 27, 1950 ANL-MM-596

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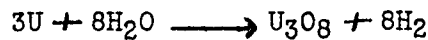
January 27, 1950

W. M. Manning

J. J. Katz and H. R. Hoekstra

Reactor Hazards

A theoretical examination of reactor hazards has recently been made in North American Report No. NAA-SR-31, Dec. 7, 1949. In addition to the nuclear phenomena resulting from a sudden increase in k , the heat evolved as a result of complete chemical reaction of the fuel elements with the coolant has been computed. The reactions which are considered are:



It is assumed that these reactions would occur when the fuel elements reach a sufficiently high temperature. In the case of CP-3, it is estimated that the heat evolved by these chemical reactions is comparable to that evolved by the nuclear reaction.

There is little doubt that the thermochemical calculations presented in NAA-SR-31 are correct or are at least of the right order of magnitude. However, it is implicitly assumed that these are essentially instantaneous self-propagating reactions. This situation would very likely obtain if the fuel elements were vaporized, or if they were in some way rapidly distributed throughout the coolant as very finely divided solids. We have examined the behavior of massive uranium (and some uranium alloys) with steam and

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oxygen at temperatures close to or above the melting point of uranium.

The rates of these reactions appear to be such as to require hours rather than seconds for completion.

Reaction of Uranium Metal and Alloys with Steam:--Samples of metal and alloys were treated with steam at 1000°C, 15 psi, with the following results;

	Rate of formation of oxide in mm/hr at 1000°C*	gm. metal reacted/cm ² /hr at 1000°C*
U	0.28	0.55
U(6%)-Zr	0.20	0.12
U(10%)-Al	not measurable	not measurable

*Results calculated from results obtained on 15 min. exposure.

The samples were fairly small, of the order of 1-3 g with surface areas of 1-3 cm². Uranium metal was examined in the temperature range 500-1000°C; the rate at the higher temperature is four times that at the lower.

This indicates a modest temperature coefficient for the steam reaction.

In the case of the uranium-aluminum alloy, the sample was far above its melting point. Deformation occurred, but the oxide coating which practically prevents reaction was also effective in stopping free flow of the alloy.

Reaction of Uranium Metal and Alloys with Oxygen:--Samples were treated with oxygen at temperatures up to 1150°C, which is above the melting point of the metal and the aluminum alloy. The results are given graphically in Figure 1. It can be seen that although considerable reaction occurs, the rates are slow, even above the melting point. This must certainly be the result of the oxide films which form very rapidly and which are quite adherent. In the case of the aluminum alloy, after a rapid initial uptake of oxygen, little reaction occurs even after long continued heating.

Reactions of Aluminum:--The usual method of preparing aluminum shot is to pour the molten metal into water. The literature shows quite clearly

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that the reactions of molten aluminum with water and air are very slow.

Conclusions:

1. The reaction of uranium metal, U-Zr, and U-Al alloys with steam at 1000°C and with oxygen up to 1150°C are relatively slow. We have seen no evidence for self-propagation, but this may be due to the small samples used here.

2. The aluminum and zirconium alloys are less reactive than the pure metal. This is true for the aluminum alloy even at temperatures far above its melting point.

3. The contribution of the chemical reaction to the heat evolution as a result of a run-away will depend on the degree of subdivision of the fuel elements. If they are vaporized or dispersed in very finely divided form, then the contribution will approach that resulting from the thermochemical considerations. If, however, the fuel rods melt and persist in massive form, then the heat evolution from chemical reaction may be negligible compared to the nuclear energy.

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APPENDIX G

"Meteorology of the Reactor Test Site, Arco, Idaho,"
prepared by U.S. Weather Bureau, September 16, 1949,
less the detailed tabular data.

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Handle this document asUnclassifiedUNITED STATES DEPARTMENT OF COMMERCE
WEATHER BUREAU
September 16, 1949.

Meteorology

of the

REACTOR TEST SITE, ARCO, IDAHO

Introduction

The weather of the Snake River Plains in which the reactor test site is located is principally determined by the combined effects of altitude, topography and the moderating Pacific Ocean operating in a belt of prevailing westerlies. On occasion cold waves and blizzards of Canadian origin cross the Continental ~~Divide~~ ^{Divide} to the north and east, but on the whole the climate is milder than the latitude and altitude would lead one to expect.

The egg-shaped area, 160 miles long in a northeast-southwest direction and 120 miles wide, with which this report is concerned, is shown in Fig. 1. This area is bounded by the Snake-Salmon watershed on the northwest, the Big Wood River and Gooding on the west, the Snake River and Camas Creek on the south and east, and the Continental Divide on the north. Within this area lies the reactor test site, defined for purposes of this report (Fig. 2) as a rectangle without a northwest corner, 25 miles in the east-west direction and 32 miles in the north-south, enclosing the Arco Navy Proving Ground. In the southeast corner are Middle and Little Buttes, reaching heights of 6394 and 6605 ft. above sea-level in contrast to the average of about 5000 ft. over the gently rolling remainder, which is the focus of several rivers that disappear in or near the site. The largest of these are Big Lost River, Little Lost River, Birch Creek and Camas Creek.

Three separate meteorological reports dealing with this area have been prepared in recent months. The U. S. Weather Bureau submitted two reports to the Washington AEC Division of Engineering, namely, "Preliminary Climatological Report of the Pocatello Area," dated December 21, 1948, and "Cold Weather Climatological Study of Arco, Idaho," dated February 12, 1949. Additional information on the weather of the site neighborhood has also appeared in the "Survey Report," by Smith, Hinchman and Grylls under the general title of "Climate." This report draws freely from these various publications and in some instances abstracts or includes complete sections. All data and estimates contained herein have been examined critically and in some cases are revisions or extensions of earlier material based upon a rather complete examination of all available data, field trips in the area, and conversations with local residents. The needs of the engineers, designers, contractors, and the problem of atmospheric pollution and its behavior in the atmosphere have been kept in mind as much as possible during the writing of this report. It is hoped that the present report will provide a useful estimate of the climatology of the site until actual "on the spot" measurements are available.

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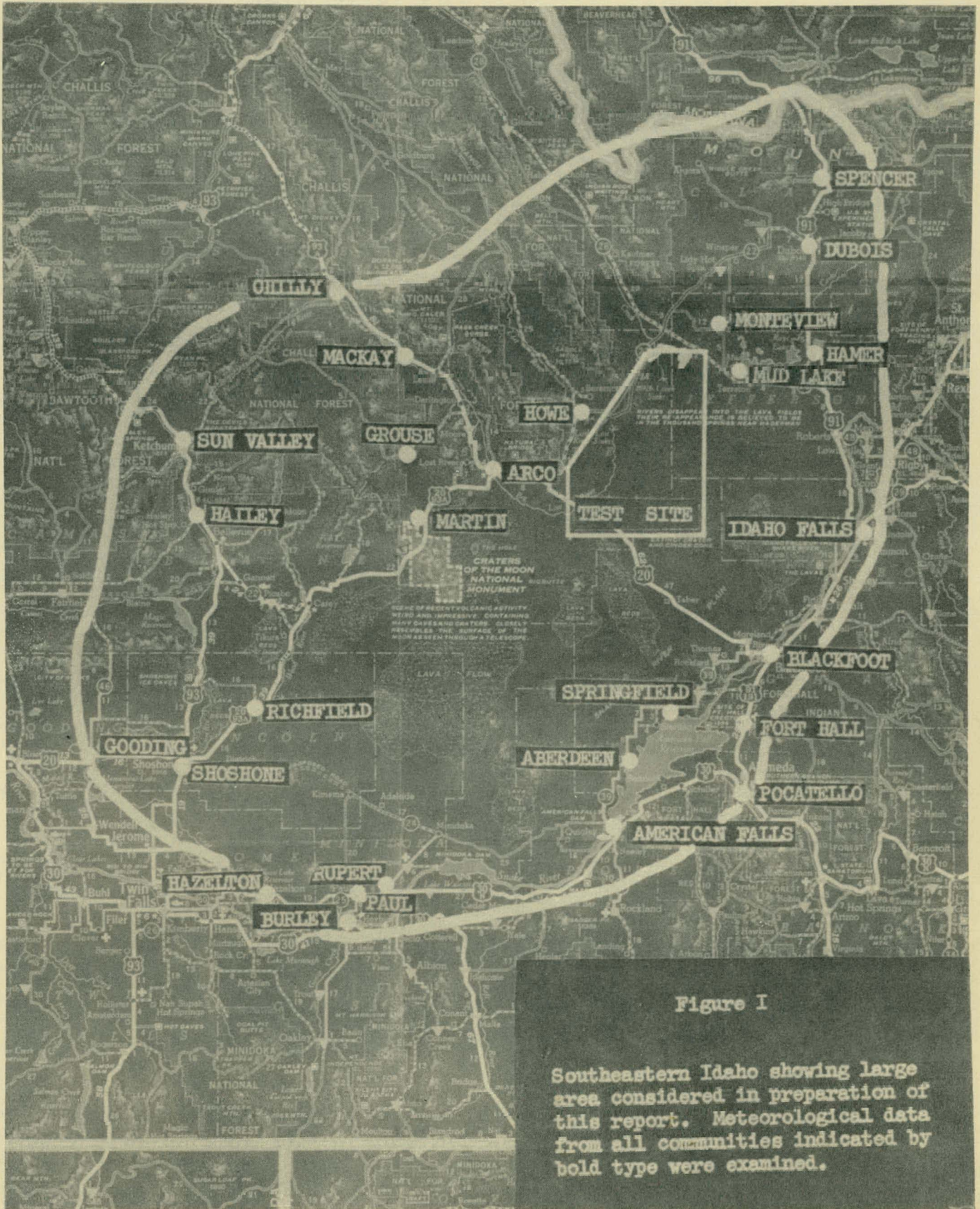


Figure I

Southeastern Idaho showing large area considered in preparation of this report. Meteorological data from all communities indicated by bold type were examined.

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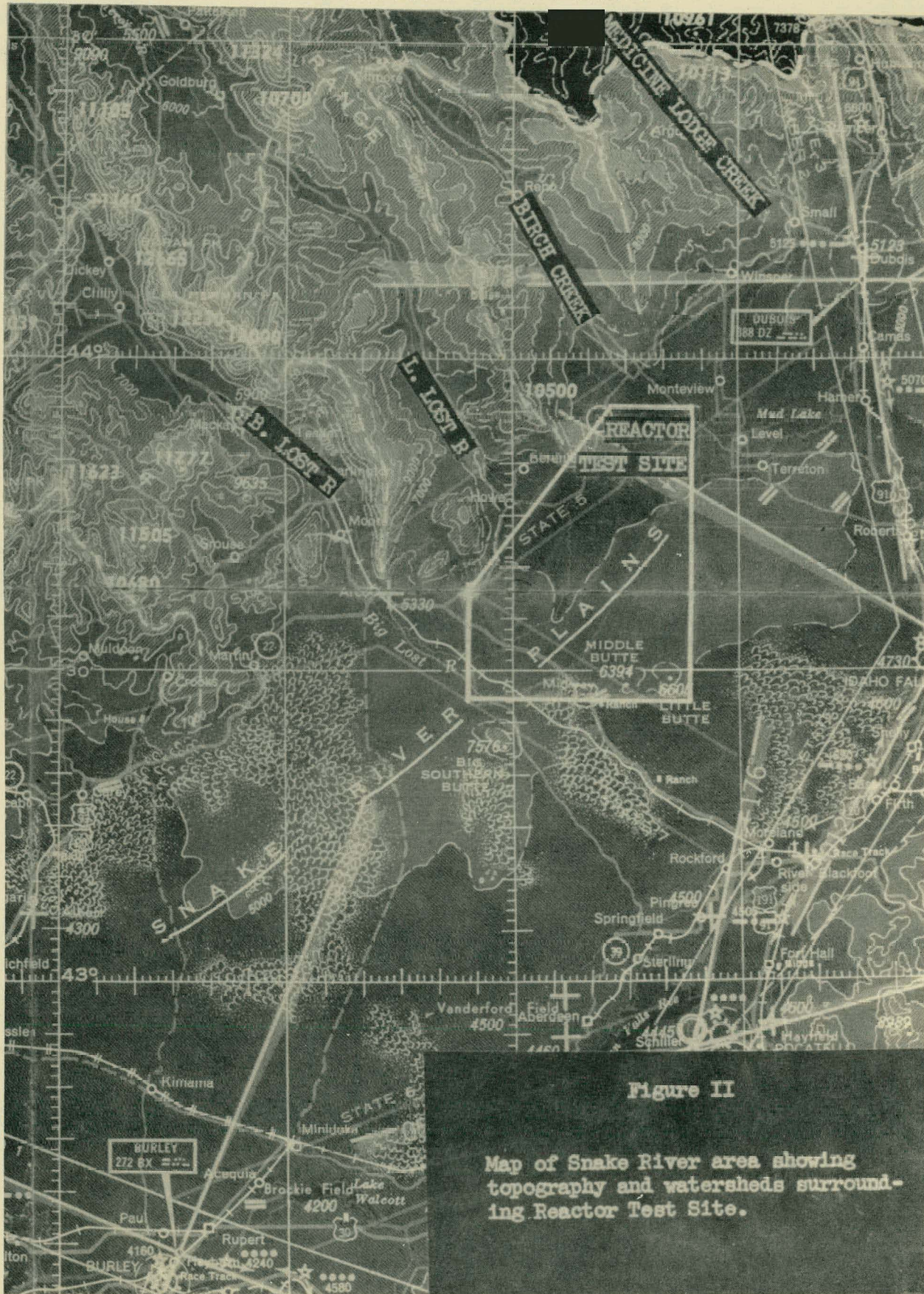


Figure II

Map of Snake River area showing topography and watersheds surrounding Reactor Test Site.

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USWB "Meteorology of the Site "

-2-

September 16, 1949

The first of the three principal sections comprising this report discusses the meteorological expectations of the site, weather data outside of the site being mentioned when considered helpful or documentary. The second summarizes the principal weather estimates for the area. Tables and sources of climatological data for the area are given in the final section.

I. METEOROLOGY OF THE AREA

The site embraces perhaps the coldest and driest portion of the Snake River Plains, and one of the coldest and driest regions of the State. The southwest and northeast quadrants describe at least two thirds of all surface winds, and these occur in about the ratio two to one. Calms represent a large part of the remaining third.

Details of the wind, temperature, precipitation, humidity, frost, fogs, clouds, and miscellaneous weather features are discussed below in order.

1. WINDS

Because of its location in a valley oriented southwest-northeast in the zone of the westerlies, the test site has a prevailing southwest wind. Winds from the opposite direction, northeast, may be expected often also, largely due to gravity effects when radiation outward cools and hence makes the air at higher levels of the large valley more dense. Variations in this general wind picture may be expected due to the local circulations in the mountains to the west and north.

Ordinarily, the direction and speed of air flow is governed pretty much by widespread pressure systems such as are observed on weather maps. However, in hilly country this is not always true. Because of the lowered density, air that is being strongly heated during the day tends to flow up the valleys and mountain sides. At night, the air cools off rapidly at the high elevations of the area and then flows, much like water, downhill out onto the plains. This cycle of "drainage wind" movement is evident on a large scale in the Snake River Valley, but also appears in the small valleys like these of Little Lost River or Birch Creek Valleys to the northwest of the site. The up slope and drainage effect should not be considered merely as super-positions on the free atmospheric wind, since they may at times be so powerful as to reverse direction of the surface wind from that expected from a study of weather maps.

During the cold half of the year, frontal wind shifts at the ground often do not occur or else last only a few hours as the fronts ride over the surface pool of cool air.

A. Winds in the General Area.--Wind records were not available for the vicinity of the Naval Proving Grounds so that a detailed summary of the frequency of the wind directions and speeds for the site itself would be guesswork. However, such summaries are available for a few meteorological stations along the Snake River such as Idaho Falls, Pocatello, Burley, and Boise. The wind at the site had to be evaluated from these data along with a study of the daily prevailing wind as recorded by cooperative observers who take one meteorological reading each day at most of the other stations shown in Fig. 1.

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For Idaho Falls, six years of hourly observations have been summarized in Table I. Tabulations both for a year and by months are given of the frequency of occurrence of wind directions and speeds. Winds of 40 mph or more have occurred in all months of the year, the highest frequency of wind speeds over 31 mph occurring in February, March and April, the lowest in July and August. Calms (0-3 mph) occur each month with a frequency between 8 and 17%, are least frequent in March through June, and most frequent in October through January. High winds (over 31 mph) occur almost exclusively from the south through west-southwest, the ratio of annual frequency in this sector to the balance of the wind rose being approximately 20 to 1. High winds outside this sector occurred almost exclusively in April through July, and arrived from all directions except east. With regard to direction frequencies, on an annual basis southwest and south occur 43% of the time and are to the secondary direction, north and northeast (23%), in almost the ratio 2 to 1. However, in January and February these direction groups occur with practically equal frequency, while in April and May the ratio is practically 3 to 1. Furthermore, if attention is confined to the light wind class (4-15 mph) it is seen that about three fourths of these winds annually are southwest to south or northeast to north, and these are in the ratio of only 13 to 11 (cf. 2 to 1 if all speed classes are considered). In the months November through February, northeast to north in the light wind class even exceeds southwest to south, in January and February, the ratio being approximately 2 to 1. The directions east and west occur with minimum frequency.

Six years of hourly observations at Pocatello Airport ending in 1938 are summarized in Table II which gives the annual frequency of occurrence of wind directions and wind speeds. Calms (0-3 mph) occurred more frequently (17% compared with 13%) than at Idaho Falls. The frequency of southwest and south winds is to the frequency of northeast and north winds almost as the ratio 5 to 1. At Pocatello Airport winds from the southeast are just as frequent as those from the southwest. Table III presents average wind speeds at Pocatello and Pocatello Airport by months for two different periods of record. There is a suggestion that in spring and summer the city is shielded somewhat by the Bannock Range from the prevailing southwesterly winds.

In Table IV, the percentage frequency of occurrence of various wind speeds by seasons is given for Idaho Falls, Pocatello, Burley, and Boise, based upon 5 to 8 years of record for each station. Calm and light winds (0-3 mph) occur most frequently in fall and winter at Idaho Falls and generally at the other stations, and least frequently in spring. Moderate (16 - 31 mph) and heavy (over 31 mph) winds are more frequent at Idaho Falls than at the other stations downstream on the Snake, occur in all seasons, and are most frequent in spring.

Boise frequencies of occurrence of wind speeds in the classes 0-3, 16-31, and 32 mph or over for a period of approximately 8 years were studied because they exhibit features of the diurnal variation of wind speeds in the Snake River Valley which apply to the site neighborhood, but which are not available for stations nearer the site. A common feature of the data for all four seasonal mid-months (January, April, etc.) is a high frequency of calm and light winds (0-3 mph) between 2130 and 0730 hours, with the maximum frequency at 0730.* Minima occur at 1330 and 1630 hours, the frequencies being $1/3$ to $1/10$ of the maxima. Calm and light winds occur most often in October, and least often in April. Winds of 16-31 mph occur most often at 1330 in all months except July when they occur most often at 1630. This class of wind speed is most frequent in April and least frequent in October and July.

*Times in this paragraph represent the 3-hour period centered on the given time. All are local times.

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The prevailing winds by months for each of 16 stations in the area under consideration, and including Boise are given in Table V. It should be remembered in the interpretation of this table that no information whatsoever is given about the second most frequent direction. Furthermore, at cooperative stations (those not marked with an asterisk,) the prevailing direction tends to be based by the daytime wind occurring at the most regular time of observation. The prevailing wind directions given for Pocatello, Pocatello Airport, Idaho Falls Airport, and Boise on the other hand are determined from hourly observations. The more detailed Idaho Falls and Pocatello Airport records were mentioned above. Most striking is the influence of the Portneuf Valley on the Pocatello City prevailing direction. In the five miles from the city to airport, going away from the Portneuf Valley, the prevailing direction changes from southeast to southwest in every month of the year. This is mentioned since it shows the effect of valleys on prevailing winds.

For stations on the Snake River Plains, the downstream direction prevails in January and February at Idaho Falls, and similar shifts from southwest and west to north, northeast or east may be noted for Richfield, Shoshone, Gooding, Aberdeen, and Hailey. Mackay, up in the Big Lost River Valley, has prevailing northwest winds in all months, due to the combined effect of valley air drainage downstream and the reaction of westerly flow aloft, upon the valley topography.

Tornadoes are of rare occurrence in southern Idaho.

B. Maximum Winds in the General Area.--Maximum reported winds over the Snake River Plains area are comparable at Idaho Falls, Pocatello, Burley, and Boise (see Table IV). Maximum winds for each month for Idaho Falls are given in Table I and for Pocatello in Table VI. The later table contains records for both city (40 years record ending in 1938) and the airport (10 years record ending in 1948). Maximum winds of 38-47 mph have occurred in every month of the year in the city and of 39-56 mph in every month of the year at the airport. The average number of days in each month that the wind exceeded 31 mph during the 8 year period ending in 1946 is also given in Table VI. General winds of 20-30 mph often continue for two days at a time in the spring.

C. Winds Aloft.--Upper air movements to 3,000 feet follow closely the surface winds, except that the frequency of southwest and west winds will tend to increase largely at the expense of north and northeast winds. From 3,000 to 25,000 feet, the winds are predominately from the west, favoring southwest in summer and northwest in winter. Average wind speed at 10,000 feet mean sea level is approximately 15-25 mph.

D. Average Wind Conditions over the Test Site.--From detailed study of the wind information discussed above, it is believed that the wind over the site follows the general wind picture as depicted by the Idaho Falls wind summary in Table I, modified to some extent because of the neighboring mountains to the west and northwest of the site, so that a greater frequency of west and northwest winds (15-20% instead of 10%) is expected at the expense of the prevailing southwest and northeast wind directions. The most frequent direction of strong winds will be southwest*, of light winds, northeast. The season of lightest winds will be late winter and early spring. Winters may be expected when stagnant near calm conditions will prevail in the site neighborhood and over most of the Snake River Plains generally for several weeks without interruption. The prevailing wind should still be from the southwest quadrant, and there should be a definite secondary maximum from the northeast quadrant. Wind

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flow from the northwest quadrant should follow in percentage frequency. The least frequent winds should arrive from the east and southeast. One or two chinooks may be expected annually, occurring in January or February.

The percentage frequency of the wind speeds will probably be similar to those reported for Idaho Falls, although it is expected that a greater percentage (18-23% instead of 13%) of calm to light winds shall be encountered. Maximum winds should average 55-65 mph although gusts may reach 20 or more mph higher.

Winds aloft should be approximately the same as those discussed in section 1-C. A good example of the change in the wind direction with height can be noted by examining pictures of the "Big Blast" at the Naval Proving Grounds in August 1945 which are shown in Fig. 3. From special meteorological observations which were taken during that experiment, it has been noted that winds from the surface to approximately 900 feet were northeast 4 mph shifting gradually to southwest 4 mph at 1600 feet above the ground and increasing to southwest 18 mph at 4900 feet above the ground. The temperature was nearly constant at 16°C to 130 feet above the surface, then rose to 18°C at 1500 feet, remained at 18°C to 2220 feet, and was adiabatic above. Note how the smoke in the lower portion of the smoke cloud drifts to the left (southerly) and aloft it drifts to the right (northerly).

2. TEMPERATURES

A. Variation of Surface Temperature.--More reliable temperature data are available over the Snake River Plains than wind data described earlier, and it is therefore easier to state with some fair degree of accuracy the average surface temperature conditions which may be expected over the site.

Table VII shows that the highest temperatures recorded in this area range from 99°F at Idaho Falls to 109° at Hailey, while the lowest temperature ranges from -23°F at Pocatello Airport to -46° at Arco. The winter climate is tempered somewhat by warm air currents moving in from the Pacific Ocean, but this effect is noticed less at the site than southeast of the Midway rise. The yearly average minimum temperature ranges from 26.0°F at Arco to 36.8° at Pocatello, while the annual average maximum temperature ranges from 56.4°F at Arco to 62.0° at Burley. The yearly average temperature ranges from 41.8°F at Arco* to 49.4° at Hazelton.

From a detailed study of these temperature data, there appears to be a definite "cold tongue" protruding over the area from the northeast and north roughly centered on the site. During the winter, this tongue is much more marked.

A brief cooperative weather record available from Midway for the period October 1941 through September 1943 was examined. The mean monthly temperatures were compared with those of Pocatello and Pocatello Airport for the same period. In every case the Midway temperature was lower than the corresponding Pocatello temperatures, and most of the time exceeded the 2 to 3 degrees to be expected due to altitude differences, reaching 8 to 9 degrees lower during this short period of record.

To get some idea of the duration of cold weather in the neighborhood of the site, a detailed tabulation of approximately 16 years of maximum and minimum temperatures records for Arco was undertaken. The minimum temperatures observed for each winter month during the period studied is tabulated in Table VIII. The sub-zero minimum

* 38.9° at Chilly, 38.7° at Spencer, and 37.5° at Sun Valley, all at considerably higher elevation than Arco. Average maximum and minimum temperatures for these Stations do not appear in this report.



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September 16, 1949

temperatures observed during all of the above months are tabulated by ten degree intervals in Table IX.

To obtain some idea of the duration of this cold weather, the maximum temperature for each day of sub-zero weather is also tabulated in the same table. It is of interest to note in Table IX that during the entire period studied, there were 510 days with temperature below 0°F, or approximately 33 days per year. This may be expanded in the following distribution:

Temperature range	Average number of days per year with sub-zero temperature.
0° to -10°F.	21
-11 to -20	8
-21 to -30	3
-31 to -40	1
-41 to -50	-

In Table IX it is also of interest to note that out of the 510 days with sub-zero temperatures, 117 days (23%) had a maximum temperature greater than 32°F. Also there are only 2 days when the maximum temperature remained below 0°F. It appears that whenever low minimum temperatures occur, the temperature rises most frequently during the day to approximately 21-32°F. January is by far the coldest month on the average, as shown below:

Month	Average number of days per year with sub-zero temperature.
November	1
December	7
January	14
February	9
March	1

In order to provide additional information on the duration of the cold weather, all data which shows sub-zero minimum temperature on consecutive days (2 or more), during this period of record are tabulated in Table X. The following distribution of this cold weather may be noted:

Minimum temperature	Average number of consecutive cases
0°F or below	6
-11° or below	2
-20° or below	1
-30° or below	less than 1

As might be expected, the most frequent number of consecutive cases calls for 2 cold days in a row, although there were many cases when the minimum temperature remained below zero for 10 or more consecutive days.

To give further information about the duration of cold weather, Table XI shows the number of consecutive days (2 or more) when the maximum temperatures never exceeded 32°. Over this long period of record the following distribution was found:

Maximum temperature	Average number of consecutive cases per year.
32°F or below	8
20° or below	1
10° or below	less than 1

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It is believed that actual measurements at the site of vertical temperature distributions (and wind) are most desirable for purposes of stack design and analysis of safety problems.

B. PRECIPITATION

Precipitation in the area is low mainly due to the intervention of the Olympic and Cascade mountain ranges in the westerly flow from the Pacific. Average yearly precipitation in the area ranges from 7.38 inches at Mud Lake, 7.45 at Chilly, and 7.80 at Montevieu to 17.93 inches at Spencer, and 18.88 inches at Sun Valley (see Table XIII). The greatest precipitation is in the late spring and mid-winter with nearly twice as many hours of precipitation expected in winter as in spring. Precipitation has varied in wet to dry years from 18-22 inches to 6-8 inches in Pocatello. The area average is about 11 inches annually, and the most frequent values are 9-11 inches. The maximum precipitation in one month was 3.2 inches at Arco and 4.34 inches at Pocatello. In 24 hours it was 2.6 inches at Pocatello (see Table XIV). The maximum precipitation in one hour may be expected to be approximately 1-2 inches. This indicates that it is unlikely that floods would occur in the area because of heavy rains alone. (however, see section 2-B).

While snow has fallen in all months except August in this general area, practically none has occurred from June through September. The average annual fall ranges from 23.7 inches at Aberdeen to 81.0 at Hailey (see Table XV). The maximum snowfall in one month was 27.4 inches at Pocatello (see Table XVI), while the greatest for Pocatello was 14.6 inches. The greatest number of consecutive days that 1 inch or more of snow cover remained on the ground was approximately 68.

For the site itself, precipitation will average about 9 inches annually, falling in amounts of .01 inch or more on about 50 days of the year (see Table XVII). Most precipitation falls in late spring and mid-winter, whereas, least may be expected in late summer and fall.

The average winter precipitation will be 3-4 inches (melted equivalent). The average precipitation from a wet to a dry year may vary from 13-16 inches to 4-6 inches. As much as 4 inches may fall in a month, and most of this in one storm. Sprinkles resulting from showers which largely evaporate before reaching the ground will occur often in the warm months. Snow may fall in all months but August (small amounts in June through September). The average annual snowfall will approximate 35-40 inches, falling on perhaps 60 days of the year. It is ventured that snow (one inch or more) may be expected to remain on the ground annually for 80-100 days. Maximum snowfall in one month is not expected to reach the Pocatello figure of 27.4 inches nor is the maximum depth on the ground likely to exceed 20 inches except in drafts.

The thunderstorm season extends from April to October with a maximum in the summer months. The average total number per year is perhaps 30. More than this will be observed over neighboring mountain ranges. Hail accompanies these thunderstorms on an average of 4 days per year.

4. HUMIDITY

Relative humidity records have been collected by the Pocatello and Idaho Falls Weather Bureau Offices. However, only the former records were found to be

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tabulated and are presented in Table XVIII. At Pocatello, average relative humidity ranges from 53 to 79% in the morning (8:00 a.m.), 26 to 71% at noon, and 24 to 72% in the late afternoon. At Pocatello Airport, average relative humidity ranges from 54 to 89% at 5:30 a.m., 29 to 80% at 11:30 a.m., 22 to 77% at 5:30 pm, and 37 to 87% at 11:30 p.m. The higher average relative humidities occur in winter and in the early morning.

Because of the slightly higher elevation of the site area as compared with Pocatello, (refer to Figure 2), its situation in the lee of prevailing westerlies, and its distance from the moisture supply represented by the Snake River and its irrigational system, the average wet-bulb temperature at the site will be somewhat lower than that of Idaho Falls, which has in July a wet-bulb temperature of approximately 55°F.

5. FOG AND CLOUD COVER

Dense fogs are not frequent throughout the Snake River Valley. Idaho Falls experiences light to moderate fog approximately 15% of the time during the winter, although relatively little dense fog (see Table XIX). The effect of this fog on visibility can be noted in Table XX which shows that visibility on the average may be reduced to 6 miles or less approximately 20% of the time during the winter months. Cloud cover also has a winter maximum. A study made of the instrument, closed or contact flying conditions experienced at various airfields along the Snake River indicates that the maximum frequency of instrument or closed conditions occurs at all fields except Pocatello from December through February. The highest frequency occurs at Idaho Falls, which may experience approximately 22% instrument or closed conditions on the average in January. Pocatello shows spring maximum in April through June and an August through October minimum. Similar conditions are also noticeable from Table XXI which shows the percentage frequency of occurrence of various ceiling heights at Idaho Falls, Pocatello, Burley and Boise. From this table, ceilings less than 5000 feet above the ground can be expected nearly 40% of the time during the winter months in the immediate vicinity of the Snake River around Idaho Falls. Ceilings less than 1000 feet may be expected approximately 15% of the time during winter months. However, much sunshine exists in late spring, summer, and early fall as can be seen in Table XXII.

For the site itself, it may be expected that long and almost cloudless days in summer with invariably cool nights due to rapid radiation. Because the location of the site is on the lee side of the mountains, both fog and low cloudiness should be less frequent and the air drier than experienced along the Snake River. Clouds may form and hang on the Buttes on many days during the winter. On the average, when clouds do exist over the site area they should be much higher above the ground than the ceilings measured along the Snake River, and rarely should get below 1000 feet above the ground except during storms. Little fog is experienced over the site area according to local observers, and during those occasions, it will more than likely be ice fog.

6. DUST

There are no data regarding the dust content of the atmosphere in the Snake River Plains area. Local observers report occasional "dust devils" or small twisters which wend a zig-zag course across the plains. These appear to have a strong twisting force but not enough to damage buildings and other structures.

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However, these storms have been reported to carry dust many hundred feet into the air. Pilots have reported these twisters to be windy and turbulent and have stated that some of them are large enough to carry "mud balls." Local weather personnel estimate that the average diameter of a large dust devil may approximate 5-10 miles and that there may be at least six of these occurring during each of the months of May, June and July.

In dry years there is some "soil-blowing" resulting in dust storms over considerable areas, especially where the soil cover has been removed. This type of storm is reportedly worse close to the American Falls Reservoir where local observers estimate that 6-12 light to moderate dust storms may be expected annually. The best months for these dust storms appear to be June and July, and in rare occasions visibility may be reduced to zero. Pilots have reported that on the average, the dust in these storms may be carried to 2-3,000 feet above the surface. A competent observer who has lived for at least 10 years in Eastern Washington and for a similar period in the Snake River Plains, states that the dust storms experienced in the latter locality are definitely not as severe, numerous or as lengthy as those experienced in the Yakima, Washington area. Continuous dust storms of several days duration have not been observed.

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The following summary is presented as a brief resume of the average meteorological conditions which may be experienced in the vicinity of the test site over a period of years based upon study of such neighboring records as are available.

A. Surface Winds

Prevailing direction	SW
Next most frequent direction	NE
Least frequent direction	E
Frequency of wind speeds	
0-3 mph	15-20%
4-15	60-70
16-31	10-20
32 and over	1-2
Most frequent direction of strong winds	SW
Most frequent direction of light winds	NE
Season of highest winds	late winter and early spring
Season of lightest winds	fall and early winter
Annual number of chinooks	1-2 per year
Maximum winds	55-65 mph with gusts increasing wind by 20 or more mph.

B. Upper Winds

Most frequent direction below 3000 ft. above surface	S thru W
Most frequent direction 3000-25,000	W
Average speed 10,000 ft	15-25 mph

C. Temperature

January	
Average daily maximum	high 20's°F
Average daily minimum	4-6 °F
July	
Average daily maximum	mid 80's°F
Average daily minimum	high 40's°F
Extremes	
Maximum	105°F
Minimum	-45°F

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Average no. days annually with temperature 90°F or over	20 days
32°F or below	150 days
0°F or below (see pp. 6 to 7 Part I for details.).	30 days
Average no. Degree Days annually (base 65°F).	8300

D. Inversion

Season of maximum number of inversions	fall and winter
Ratio of inversions to lapse rate conditions.	2 to 1

E. Precipitation

Average annual precipitation	9 inches
Season of most precipitation	late spring and mid winter
Season of least precipitation	late summer and fall
Average total precipitation October to March	4.5 inches
Average total precipitation April to September	4.5 inches
No. days of .01 in. or more annually	50 days
Precipitation in wet/dry year	13-16 in./4-6in.
Max. precipitation in 1 month	4 inches
Maximum precipitation in 1 day	2 to 3 inches
Maximum precipitation in 1 hour	1 to 2 inches
Average annual snowfall	35 to 40 inches
Month of greatest snowfall	January
Month of no snowfall	August (June through Sept.)
Average number days annually with 1 inch or more snow on ground	80 to 100 days
Maximum snowfall in 24 hours	25 inches
Average no. thunderstorms per years.	30 (more will be visible over distant mountains.)
Months of maximum frequency of thunderstorms(no.)	June-August (4 to 5 per month).
Average number days annually with hail	4

F. Frost

Average date first killing frost	2nd week September
Average date last killing frost	1st week June
Length of growing season	95 to 100 days
Range in length of growing season	50 to 130 days
Average depth of frost penetration	3 feet

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G. Humidity

Average annual relative humidity	40% early afternoon
	75% near sunrise
Average summer relative humidity	20% early afternoon
	55% near sunrise
Average winter relative humidity	70% early afternoon
	85% near sunrise
Average July wet bulb temperature	50 degrees

H. Fog and Clouds

Season of Maximum occurrence of fog	Winter
Percent of occurrence of fog in winter	5 to 10%
Season of maximum cloudiness	winter and spring
Percent of occurrence of ceilings less than 5000 feet in winter	25 to 35%
Percent of occurrence of ceilings less than 1000 feet in winter	5 to 10%.

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APPENDIX H

"Argon Activity in Reactor Cooling Air," by J. A. Lane

dated November 17, 1949, ORNL CF# 49-11-180

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Oak Ridge National Laboratory
Central Files Number
49-11-180

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Date: November 17, 1949
Subject: Argon Activity in
Reactor Cooling Air
To: MTR Steering Committee
From: J. A. Lane

Before reading this document, sign and date below:

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November 17, 1949

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To: MTR Steering Committee
From: J. A. Lane
Subject: Argon Activity in Reactor Cooling Air

The following calculations indicate the activity of the pile cooling air due to neutron capture in argon and resulting A^{41} decay for the existing design of the graphite reflector. Contributions from the pebble zone, plenum chambers and permanent graphite are considered separately for operation of the MTR at 30 M.W.

Pebble Zone

maximum thermal neutron flux = 5×10^{13}
flux at top of pebble zone = 5×10^{11}
flux at bottom of pebble zone = 1.5×10^{12}
average flux at tank wall = 1.2×10^{13}
average flux at permanent
graphite = 3.6×10^{12}
average flux in pebble zone = 6×10^{12} n/sec(cm)²

The absorption cross section of A = 1.24 barns

N = .00002 atoms/cc

at 60°C, 720 mm

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The concentration of argon in air is 0.94%.

$$N \sigma_a = 2.5 \times 10^{-5}$$

The number of atoms of A^{41} formed per second is,

$$N_A = (N \sigma_a) n v (.0094) V$$

where, V = volume of air in pebble zone

$$= 0.40 \times 9.5 \times 10^6 \text{ cc (not including plenum)}$$

$$= 3.8 \times 10^6 \text{ cc air}$$

$$N_A = (2.5 \times 10^{-5}) (6 \times 10^{12}) (9.4 \times 10^{-3}) 3.8 \times 10^6$$

$$= 5.37 \times 10^{12} \text{ atoms per second}$$

$$= 4.64 \times 10^{17} \text{ atoms per day}$$

$$\lambda = \frac{0.693}{t_{1/2}} = \frac{0.693}{110 \times 60} = 1.05 \times 10^{-4} \text{ secs}$$

$$N_A \lambda = \frac{4.64 \times 10^{17} \times 1.05 \times 10^{-4}}{3.7 \times 10^{10}}$$

$$= 1.3 \times 10^3 \text{ curies} = 1300 \text{ curies per day}$$

In Plenum Chambers

Top plenum

$$n v (\text{avg}) = 2 \times 10^{11}$$

$$V = 150 \text{ cu ft}$$

$$= 4.25 \times 10^6 \text{ cc}$$

$$N_A = 2.5 \times 10^{-5} (2 \times 10^{11}) (.0094) 4.25 \times 10^6$$

$$= 2 \times 10^{11} \text{ atoms per second}$$

$$N_A \lambda = 50 \text{ curies per day}$$

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Bottom plenum chamber

$$(nv)_{avg} = 6 \times 10^{11} \text{ neutrons/cm}^2(\text{sec})$$

$$N_A \lambda = 145 \text{ curies per day}$$

Permanent Graphite

700 cooling holes, 1/2" diameter 9 1/4" long

$$= 2.6 \times 10^5 \text{ cc}$$

Assume half of holes are in an average flux of 2×10^{12} and half of holes in flux of 1×10^{12}

$$N_A = (2.5 \times 10^{-5})(1.5 \times 10^{12})(9.4 \times 10^{-3}) 2.6 \times 10^5$$

$$= 9 \times 10^{10} \text{ atoms A}^{41}/\text{sec}$$

$$N_A \lambda = 23 \text{ curies per day}$$

Total activity in air is equal to 1500 curies/day.

$$50000 \text{ cfm up the stack} = 7.2 \times 10^7 \text{ cf/day}$$

$$= 2 \times 10^{12} \text{ cc/day}$$

$$\text{Activity at top of stack} = 7.5 \times 10^{-10} \text{ curies/cc}$$

The dilution due to wind velocity over the stack is given as follows:

The width of the cloud is roughly equal to one seventh of the distance from the stack. The height of the cloud depends on inversion factors, etc. Conservatively it may be taken as twice the stack height. For a 2 mph wind, the volume of air past the stack per day, 700 feet from the stack is,

$$2(5280)24 = 2.54 \times 10^5 \text{ ft/day}$$

For a 250' stack the width and height of the cloud are 100 ft. and 500 ft. respectively.

$$\text{The air flow is } 1.27 \times 10^{10} \text{ cu ft/day}$$

$$= 3.6 \times 10^{14} \text{ cc/day}$$

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The activity in this cloud due to A^{41} is

$$4 \times 10^{-12} \text{ curies/cc}$$

The maximum β energy is 1.18 mev/disintegration and γ energy 1.37 mev/photon. Assume that range of gammas is 1000 ft and betas 5 ft. The radiation intensity for the β rays is equal to,

$$(3.7 \times 10^{10})(4 \times 10^{-12}) \frac{1.18}{3} \times 10^6 = 5.8 \times 10^4 \text{ ev/sec (cc)}$$

$$\frac{(5.8 \times 10^4)(4.8 \times 10^{-10})}{32} 3600 \times 10^3 = 3.1 \text{ mr/hr from } \beta \text{ rays}$$

The radiation intensity due to γ rays is approximately

$$I = \frac{I_0}{2} (1 - F_1(\mu h)) \quad \text{where, } h = 50 \text{ ft.}$$

$$\mu = 1/1000 \text{ ft.}$$

$$= (0.34) \frac{I_0}{2}$$

$$I = (0.17) 1.37 = 0.234 \text{ mev/dis.}$$

The radiations from the γ rays will therefore be equal to 1.9 mr/hr.

The total intensity is 5 mr/hr for the above conditions.

Experience at ORNL indicated that in all probability, the dilution due to wind is considerably greater than estimated above. At Oak Ridge the maximum argon activity ever observed on the ground in the vicinity of the 105 stack amounts to $\sim 10^{-13}$ curies/cc, while the concentration at the top of the stack was 10^{-10} curies/cc. This means a minimum dilution factor of 1000 compared to the calculated value of 200. Even under unfavorable wind conditions at Arco, therefore, the activity at the ground level due to argon will probably not exceed 1 mr/hr.

signed J. A. Lane
J. A. Lane

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APPENDIX I

"Stack Height Calculations"

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APPENDIX I

Stack Height Calculations

The stack height of 250 feet was determined by the following calculations made in the main by Mr. Carl Leyse⁽¹⁾ of Argonne National Laboratory.

a. Horizontal Cone Analysis.

This approach has been used previously in CH-504, Report for Period Ending February 27, 1943, Metallurgical Project, Health Radiation and Protection. It is assumed that the vertical cross sectional area of the contaminated cloud at the point where it reaches the ground is h^2 , h being the height of the stack.

The total volume of the cloud past this point per day is
then

$$\pi h^2 v t \quad \text{where} \quad \begin{array}{l} h = \text{stack height, cm} \\ v = \text{wind velocity, cm/sec} \\ t = \text{time for days activity to} \\ \quad \text{pass observer, 86,400 sec.} \end{array}$$

Then, if C is the total curies produced per day and c , in curies/cc, is the tolerance at ground level, we have

$$C = \pi h^2 v t c$$

The tolerance for an observer immersed in a semi-infinite medium of A^{41} is 10^{-6} micro-curies/cc for plant areas.

The total activity for one day is 1500 curies.

The above then reduces to

$$v = \frac{0.55 \times 10^{10}}{h^2}$$

which enables us to calculate the minimum velocities necessary to provide sufficient dilution for different stack heights.

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<u>Stack Height (ft)</u>	<u>Minimum Wind Velocity (mph)</u>
200	3.4
250	2.15
300	1.5

In a report by the Weather Bureau on the Arco, Idaho Site, the wind is less than 4 mph only 15-20% of the time. Thus, a 250-foot high stack appears adequate.

b. Oak Ridge Experience.

In CF-49-11-134, Dr. K. Z. Morgan of Oak Ridge National Laboratory gave the following information. The argon intensity at the exit of the 200-foot high stack at ORNL is 1.2×10^{-4} micro-curies/cc. Instruments on and off the plant at ORNL show that activity has not exceeded 2×10^{-7} micro-curies/cc at any time in the past five years and has reached this on only a few rare occasions which are believed largely due to dissolver gases and not argon. This represents a dilution factor of 600 which has always been exceeded or a factor of about 10^3 which probably has always been exceeded for argon.

If, on the basis of this information, a dilution factor of 10^3 is applied to the MTR, the activity at the ground would be 1.25×10^{-6} micro-curies/cc, which is 1.25 times the tolerance 10^{-6} micro-curies/cc. This figure is for maximum concentration likely to occur, while the tolerance is for continuous radiation. Thus the integrated dose over a period of a week is not likely to exceed tolerance. Since the ORNL stack is 200 feet high, the same height may suffice for the MTR stack. It should be remembered that the terrain at ORNL is hilly, while that at Arco is flat. The minimum dilutions at both places will not necessarily be the same.

c. Use of Sutton's Equations.

The equations derived by Sutton⁽²⁾, see references on page 11, have had some experimental verification as reported by various authors. While the equations cannot be relied upon to give precise data, they are believed to be the best available. The Sutton Equation is

$$C = \frac{M}{\pi C_y C_z u x^{(2-n)}} \times e^{\left[\frac{-y^2}{C_y^2 x^{(2-n)}} \right]}$$

equation continued on page 3.

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$$x \left[e^{\left(\frac{-(z-h)^2}{C_z^2 x^{(2-n)}} \right)} + e^{\left(\frac{-(z+h)^2}{C_z^2 x^{(2-n)}} \right)} \right]$$

where

- C = concentration of gas at any point (x, y, z) down wind.
M = strength of source, curies/sec.
h = stack height, cm.
x = distance down wind, cm.
y = distance cross wind from center line of smoke stream, cm.
z = vertical distance above ground, cm.
u = mean wind velocity, cm/sec.
C_y = diffusion coefficient in horizontal
C_z = diffusion coefficient in vertical
n = 0 - extreme lapse rate
= 1/4 - neutral
= 1/3 - mild inversion
= 1/2 - strong inversion

If we are interested only in the concentration at the ground (z = 0), on a line down wind from the stack (y = 0), the above equation reduces to

$$C = \frac{2M}{\pi C_y C_z u x^{(2-n)}} e^{\left(-\frac{h^2}{C_z^2 x^{(2-n)}} \right)} \quad \text{curies/cc.}$$

The above equation can be differentiated to give the maximum concentration at ground level down wind from the stack. This will give a maximum concentration of

$$C_m = \frac{2M}{\pi u e h^2} \frac{C_z}{C_y} \quad \text{curies/cc}$$

at a distance from the stack of

$$x_m = \left(\frac{h^2}{C_z^2} \right)^{\frac{1}{2-n}} \quad \text{cm}$$

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The diffusion coefficients C_y and C_z are assumed to be equal and are taken as equal to 0.07. The value of (n) is taken as zero; this corresponds to an extreme lapse rate. Both of the above values are reasonable and conservative based on information given by Sutton and on Thomas⁽³⁾ correlation of data from several smoke stacks.

The above equations then reduce to

$$C_m = \frac{2M}{\pi u e h^2} \quad x_m = \frac{h}{C_z} = 14.3 \text{ stack heights}$$

The distance of maximum concentration at ground level for stack heights of 200, 250 and 300 feet would be 2,860, 3,575, and 4,290 feet, respectively. Since the MTR site is within a radius of 2,000 feet from the stack, the maximum concentration will not be within the site area for any of the above stack heights.

The equation for maximum concentration is solved for three stack heights, where the maximum concentration is given as a function of u' , the mean wind velocity in mph. The strength of the source has been taken as 1,500 curies/day. These equations are also solved for several wind velocities and the results are tabulated below.

Maximum Concentration at Ground Level for
MTR with Stack Height of

	<u>200 feet</u>	<u>250 feet</u>	<u>300 feet</u>
Wind Velocity(u')	$C_m = \frac{2.44 \times 10^{-12}}{u'}$	$C_m = \frac{1.56 \times 10^{-12}}{u'}$	$C_m = \frac{1.1 \times 10^{-12}}{u'}$
<u>mph</u>	<u>curies/cc</u>	<u>curies/cc</u>	<u>curies/cc</u>
1	2.44×10^{-12}	1.56×10^{-12}	1.1×10^{-12}
5	0.49	0.31	0.22
10	0.24	0.156	0.11

If we consider tolerance as 10^{-12} curies/cc, the wind velocities necessary to reduce the maximum concentration at ground level to tolerance are 2.44, 1.56, and 1.1 mph for the 200-, 250-, and 300-foot stacks, respectively.

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The above calculations are for maximum concentrations at ground level. These do not occur on the area of the site. Therefore we will next calculate the maximum concentration that will occur on the site. The values $C_y = C_z = 0.07$ and $n = 0$ will again be used. The maximum concentration on the site when calculated by Sutton's equation will occur at the parts of the site farthest from the stack. This distance is taken as 2,000 feet.

The equation used is

$$C = \frac{2M}{\pi (0.07)^2 u x^2} e^{-\frac{h^2}{0.07x}}$$

The expressions for the maximum concentration at ground level on the MTR site are given below for the various stack heights being considered.

<u>Height of Stack, ft.</u>	<u>Max. Concentration on Site, curies/cc.</u>
200	$\frac{1.75 \times 10^{-12}}{u'}$
250	$\frac{0.55 \times 10^{-12}}{u'}$
300	$\frac{0.135 \times 10^{-12}}{u'}$

The wind velocities necessary to reduce the maximum concentration at ground level on the site to tolerance (10^{-12} curies/cc) are 1.75, 0.55, and 0.135 mph for the 200-, 250-, and 300-foot stacks, respectively.

The meteorology report on the Arco site states that winds of 0-3 mph will occur 15 to 20% of the time. This 15-20% will not be distributed equally among all velocities. That is, the probability of the wind being 0-.5 mph is much less than the probability of it being 2.5-3 mph. Then, winds of less than 0.55 mph may occur only about 1/10 as often as winds of less than 1.75 mph. Assume that the winds of less than 1.75 mph occur $1.75/3 \times 20 = 12\%$ of the time. Generally, when the wind blows from the directions between ENE clockwise to SSW the distance from the stack to the edge of the site is 1,500-2,000 feet. We will

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assume that for a 200-foot stack and winds from the above directions the intensity may exceed tolerance on the site if the velocity is less than 1.75 mph. The meteorology report gives the most frequent direction of the light winds as NE. Then it is reasonable to assume that the light winds will be from the directions ENE $\rightarrow$ SSW about 75% even though these directions constitute only 37.5% of the total number of possible directions. Thus, with a 200-foot high stack the concentration would not exceed tolerance on the site more than 9% of the time. However, if we consider the 250-foot stack on the basis that the wind will be less than 0.55 mph only about 1/10 as often as it will be less than 1.75 mph, the calculations show that tolerance will be exceeded less than 1% of the time. Therefore, a stack height of 250 feet is desirable.

d. Gamma Radiation from Elevated Cloud.

The gamma radiation from the argon activity is calculated as follows to determine the minimum wind velocities necessary to prevent excessive radiation at ground level. The calculation was based on a horizontal cloud at stack height.

It is assumed there is little diffusion and that the radioactive cloud is essentially an infinite line source at stack height. This would be analogous to a condition which may arise with low exit velocity and temperature, stable air, and very light winds.

$$E_a = \frac{\mu S_L}{4\pi a} \left[\mathcal{J}(\mathcal{F}_2, B) - \mathcal{J}(\mathcal{F}_1, B) \right]$$

$$\text{For no shielding } \left[\mathcal{J}(\mathcal{F}_2, B) - (\mathcal{F}_1, B) \right] = \mathcal{F}_2 - \mathcal{F}_1, \text{ and for}$$

an infinite line source $\mathcal{F}_2 - \mathcal{F}_1 = \pi$.

$$\text{The above reduces to } E_a = \mu \frac{S_L}{4a}$$

where

S_L = source strength per unit length in ev/sec cm.

a = distance from observer to source, cm.

E_a = energy in ev/sec cc absorbed in air at the observer.

μ = recoil absorption coefficient of air for gamma radiation

$\mu = 3.5 \times 10^{-5}$ per cm for photon energies
between 0.05 and 4 mev.

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E_a is converted to I_o in roentgens/24 hr by the product of the following factors

E_a (air)	ev/sec cc
$1/32$	ion pairs formed per ev
4.8×10^{-10}	esu associated with each ion pair
8.64×10^4	sec/24 hr.

$$\text{Then } I_o = 1.3 \times 10^{-6} E_a = 3.25 \times 10^{-7} \frac{S_L}{a}$$

The total daily activity of 1500 curies is converted to source strength along the line as a dependent of velocity v in cm/sec.

$$S_L = \frac{0.875 \times 10^{15}}{v} \quad \text{ev/sec cm}$$

The above then becomes

$$I_o = \frac{10^4}{a v} \quad \text{r/24 hr.}$$

Tolerance for gammas is taken as 0.1 r/24 hr., which can be set equal to I_o , giving

$$v = \frac{10^5}{a}$$

This can be solved for wind velocity as a function of stack heights.

<u>Stack Height (ft)</u>	<u>Minimum Wind Velocity (mph)</u>
200	0.375
250	0.285
300	0.25

These calculations show that only a very light wind is necessary to prevent excessive radiation at the ground level if the radioactive cloud is at stack height. They also show that it is important to have ample exit velocity and temperature to assure that the radioactive cloud rises to sufficient height to keep radiation at ground level below tolerance during calm periods.

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Likewise calculations of vertical cone above the stack can be made to determine the radiation intensity at the ground in case the exit air raises directly from the stack.

In CE-1398, ⁽⁴⁾ the behavior of stack gases discharging into still air was studied. Theory and experiments were adduced which show that a stack discharging air vertically upward into still air at the same temperature and density as the stack air, produces a cone-shaped body of moving air above the stack. It was assumed in this report that the volume of air induced into the cone at any distance above stack was proportional to perimeter of cone and velocity of gas within cone at that distance. Also it was assumed no friction between gas cone and surrounding air existed.

The radius of this cone is shown to be

$$r = r_s + h \tan \frac{\alpha}{2} = r_s + 0.15h \quad \alpha = \frac{1}{2} \text{ included angle of cone.}$$

$$\tan \frac{\alpha}{2} = \text{approximately } 0.15$$

r_s = stack radius

h = distance above top of stack.

with the use of the principle of the conservation of momentum, the vertical velocity of the gas was determined to be

$$v = v_s \left(\frac{r_s}{r_s + 0.15h} \right) \quad v_s = \text{stack velocity}$$

Likewise, the concentration of a contaminant was found to be:

$$c = c_s \left(\frac{r_s}{r_s + 0.15h} \right) \quad c_s = \text{stack concentration.}$$

These rules seem to break down for stacks less than two inches diameter and were tested for stacks up to eight inches diameter. Several months of qualitative observation of full size stacks found no general contradiction to them. Therefore, the given relations are used in the present calculations.

The exit radius of the stack is taken as four feet. The height of the stack is taken as 200 feet. The total concentration (C) in the cone from the top of the stack to an elevation of 1000 feet (range of

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gammas in air) is found by the following integral.

$$C = \int c \, d(\text{Vol}) = C_s \int_0^{800} \frac{r_s}{r_s + 0.15h} \times \pi r^2 \, dh$$

$$C = \pi r_s C_s \int_0^{800} (r_s + 0.15h) \, dh$$

$$C = \pi r_s C_s \left[r_s h + 0.15 \frac{h^2}{2} \right]_0^{800}$$

$$C = \pi r_s C_s \left[4 \times 800 + 48 \times 10^3 \right] = 6.45 \times 10^5 \times C_s$$

where $r_s = 4 \text{ ft}$

$c_s = \text{ev/sec ft}^3$

The concentration at the stack outlet for operation at 30,000 kw is

$$c_s = 1.75 \times 10^{12} \text{ ev/sec ft}^3$$

The total activity is then $11.29 \times 10^{17} \text{ ev/sec}$. This is assumed to be distributed uniformly along a line source 800 ft long vertically above the stack. The source strength is then

$$S_L = 4.62 \times 10^{13} \text{ ev/sec cm}$$

In TNX-7, (5) the radiation from a line source viewed endwise without self-absorption, receptor at origin, source on x-axis between

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$(x_1 - x_2)$ is given as

$$I = \int_{x_1}^{x_2} \frac{I'_0 dx}{x^2} = I'_0 \left[\frac{1}{x_1} - \frac{1}{x_2} \right]$$

where I'_0 = intensity at unit distance from the activity
in unit length of a line source.

Making the substitution for I'_0 and converting to r/24 hr,
the above becomes

$$I = 1.3 \times 10^{-6} \frac{\mu_{SL}}{4\pi} \left[\frac{1}{x_1} - \frac{1}{x_2} \right] \quad r/24 \text{ hr}$$

$$x_1 = 200 \text{ ft} = 6 \times 10^3 \text{ cm}$$

$$x_2 = 1000 \text{ ft} = 3 \times 10^4 \text{ cm}$$

then

$$I = 22.3 \times 10^{-3} \quad r/24 \text{ hr.}$$

The radiation at the ground is thus well below tolerance when
the air rises in a cone-shaped body above the stack.

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APPENDIX J

Excerpt from "Pacific Coast Uniform Building
Code," 1949.

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Excerpt from "Pacific Coast Uniform Building Code" 1949.

LATERAL BRACING

Lateral
Bracing
(Earthquake
Regulations)

Refer to Sec. 2312. The following provisions are suggested for inclusion in the Code by cities located within an area subject to earthquake shocks:

Sec. 2312. (a) General: Every building or structure and every portion thereof, except Type V buildings of Group I occupancy which are less than twenty-five feet (25') in height, and minor accessory buildings shall be designed and constructed to resist stresses produced by lateral forces as provided in this Section. Stresses shall be calculated as the effect of a force applied horizontally at each floor or roof level above the foundation. The force shall be assumed to come from any horizontal direction.

All bracing systems both horizontal and vertical shall transmit all forces to the resisting members and shall be of sufficient extent and detail to resist the horizontal forces provided for in this Section and shall be located symmetrically about the center of mass of the building or the building shall be designed for the resulting rotational forces about the vertical axis.

Junctures between distinct parts of buildings, such as wings which extend more than twenty feet (20') from the main portion of the building, shall be designed at the juncture with other parts of the building for rotational forces, or the juncture may be made by means of sliding fragile joints having a minimum width of not less than eight inches (8"). The details of such joints shall be made satisfactory to the Building Official.

(b) Horizontal Force Formula. In determining the horizontal force to be resisted, the following formula shall be used:

$$F = CW$$

WHERE

"F" equals the horizontal force in pounds,
"W" equals the total dead load, tributary to the point under consideration, except for warehouses and tanks, in which case "W" shall equal the total dead load plus the total vertical designed live load tributary to the point under consideration. Machinery or other fixed concentrated loads shall be considered as part of the dead load.

"C" equals a numerical constant as shown in Table No. 23-C.

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TABLE NO. 23-C--HORIZONTAL FORCE FACTORSLateral
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<u>Part or Portion</u>	<u>Value of "C"*</u>	<u>Direction of Force</u>
Floors, roofs, columns and bracing in any story of a building or the structure as a whole**	$\frac{.15}{N + \frac{1}{4}}$	Any direction horizontally
Bearing walls, non-bearing walls, partitions free standing masonry walls over 6' in height	.05 With a minimum of five pounds per square foot	Normal to surface of wall
Cantilever parapet and other cantilever walls, except retaining walls	.25	Normal to surface of wall
Exterior and interior ornamentations and appendages	.25	Any direction horizontally
When connected to or a part of a building: towers, tanks, towers and tanks plus contents, chimneys, smokestacks and pent-houses	.05	Any direction horizontally
Elevated water tanks and other tower supported structures not supported by a building	.03	Any direction horizontally

* See Map on inside back cover for zones. The values given "C" are minimum and should be adopted in locations not subject to frequent seismic disturbances as shown in Zone 1. For locations in Zone 2, "C" shall be doubled. For locations in Zone 3, "C" should be multiplied by 4.

** Where wind load as set forth in Section 2307 would produce higher stresses, this load shall be used in lieu of the factor shown.

†N is number of stories above the story under consideration, provided that for floors or horizontal bracing, N shall be only the number of stories contributing loads.

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(c) Foundation Ties. In the design of buildings of Types I, II and III, where the foundations rest on piles or on soil having a safe bearing value of less than 2000 pounds per square foot, the foundations shall be completely inter-connected in two directions approximately at right angles to each other. Each such inter-connecting member shall be capable of transmitting by both tension and compression at least 10 per cent of the total vertical load carried by the heavier only of the footings or foundations connected. The minimum gross size of each such member if of reinforced concrete shall be twelve inches by twelve inches (12" x 12") and shall be reinforced with not less than the minimum reinforcement specified in Section 2620. If the inter-connecting members are of structural steel, they shall be designed as provided in Section 2702, and encased in concrete. A reinforced concrete slab may be used in lieu of inter-connecting tie members, providing the slab thickness is not less than one forty-eighth of the clear distance between the connected foundations; also providing the thickness is not less than six inches (6").

Inter-connecting slabs shall be reinforced with not less than eleven-hundredths square inch (.11 sq. in.) of steel per foot of slab in a longitudinal direction and the same amount of steel in a transverse direction. The bottom of such slab shall not be more than twelve inches (12") above the tops of at least 80 per cent of the piers or foundations. The footings and foundations shall be tied to the slab in such a manner as to be restrained in all horizontal directions.

(d) Plans and Design Data. With each set of plans filed, a brief statement of the following items shall be included:

1. A summation of the dead and live load of the building floor by floor, which was used in figuring the shears for which the building is designed.
2. A brief description of the bracing system used, the manner in which the designer expects such system to act, and a clear statement of any assumptions used. Assumption as to location of all points of counter-flexure in members must be stated.
3. Sample calculation of a typical bent or equivalent.

(e) Stresses. For combined stresses due to lateral forces and other loads the allowable unit stresses and the allowable load on connections may be increase $33 \frac{1}{3}$ per cent. In no case shall the section be less than required if the lateral force stress be neglected.

(f) Detailed Requirements.

1. Bonding and Tying. Cornices and ornamental details shall be bonded in the structure so as to form an integral part of it. This applies to the interior as well as to the exterior of the building.

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2. Overturning Moment. In no case shall the calculated overturning moment of any building or structure due to the forces provided for in this Section exceed two-thirds of the moment of stability of such building or structure. Moment of stability shall be calculated using the same loads as used in calculating the overturning moment.

3. Additions. Every addition to an existing building or structure shall be designed and constructed to resist and withstand the forces provided for in this Section, and in any case where an existing building or structure is increased in height all portions thereof affected by such increased height shall be reconstructed to resist and withstand the forces provided for in this Section.

4. Alterations. No existing building or structure shall be altered or reconstructed in such a manner that the resistance to the forces provided for in this Section will be less than that before such alteration or reconstruction was made; provided, however, that this provision shall not apply to non-bearing partitions, and shall not apply to other minor alterations which are made in a manner satisfactory to the Building Department.

(g) Lime Mortars. Lime mortars shall not be used in any unit masonry construction forming a part of a building.

(h) Veneer Ties. Veneer ties provided in Section 2902 (c) shall be of sufficient strength to support four times the weight of the attached veneer.

(i) Intention or Interpretation of Lateral Force Provisions. These lateral force requirements are intended to make buildings earthquake-resistive. The provisions of this section apply to the buildings as a unit and also to all parts thereof, including the structural frame or walls, floor and roof systems, and other structural features.

The provisions incorporated in this Section are general and, in specific cases, may be interpreted or added to as to detail by rulings of the Building Official in order that the intent shall be fulfilled.

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APPENDIX K

"Earthquake Information of Possible Application at
Arco, Idaho," by J. Stewart Williams.

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EARTHQUAKE INFORMATION OF POSSIBLE APPLICATION

AT ARCO, IDAHO

By

J. Stewart Williams

Introduction

This is a brief compilation of information about the movement of the ground and buildings in an earthquake that may be of possible use at Arco, Idaho. It is compiled at the request of Mr. Frank Smith of the Atomic Energy Commission. It has been compiled in the course of a few days, from sources available at the libraries of Utah State Agricultural College, Logan, and University of Utah, Salt Lake City. It is necessarily brief.

Ground motion in earthquakes

Olympia: The latest information available on ground motion in an earthquake is that obtained from the Olympia accelerograph for the earthquake of 13 April 1949 that was widely felt over Washington and Oregon and did damage estimated at \$20,000,000 (1). Following is a summary of the accelerograph record:

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Earthquake of 13 April 1949 at 115641 P. S. T.

Puget Sound Basin

	Earth Wave Period sec.	Maximum Accelera- tion cm/sec. ²	Maximum Displace- ment cm.	Remarks
Olympia accelerograph				
Vertical-up	.10	107	.027	Very strong sinusoidal wave.
	.15	72	.041	Short-period waves .01 sec. superposed.
	.19	55	.050	
	.11	94	.028	
	.28	43	.086	
SE: 170°	.13	119	.051	Sinusoidal waves.
	.18	131	.107	
	.41	171	.720	Maximum trace amplitude on this component
	.30	145	.332	
	.14	103	.051	
SW: 260°	.14	98	.049	
	.24	98	.145	Short-period waves .01 sec superposed.
	.17	201	.147	Second largest trace amplitude on this component.
	.13	145	.063	
	.28	120	.240	Irregular waves.
	.34	321	.941	Maximum trace amplitude on record.
	.27	42	.078	

It will be noted that the maximum acceleration is 0.33g but that the maximum displacement is less than 1 centimeter. The maximum acceleration and displacement are on a period of 0.3 second.

The intensity (modified Mercalli scale) at Olympia was VIII.

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Imperial Valley: The Imperial Valley earthquake of 18 May 1940 did damage estimated at 5-6 million dollars and killed nine persons (2). The maximum intensity (IX) was at Imperial, 17 miles from the maximum slipping. An intensity of VIII- was recorded for El Centro, where the accelerograph was located, 7 miles from the epicenter and 4 miles from the nearest point on the fault.

Following is a summary of the most important elements of the ground motion as determined from the accelerograph record:

Maximum resultant horizontal acceleration 350 cm/sec.^2 on a period 0.4 second.
Maximum resultant horizontal velocity 59 cm/sec.
Maximum horizontal displacement 21 cm. on period of 3.2 second.
Maximum range of horizontal displacement 39 cm.
Maximum vertical acceleration 220 cm/sec.^2 on a period of 0.1 second.
Maximum vertical velocity 11 cm/sec.
Maximum vertical displacement 5.5 cm.
Maximum range vertical displacement 10.9 cm.

At the time these data were computed a comparison was made with the results obtained previously at Long Beach (1933) and Helena (1935). It is contained in the following table:

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Comparison of strong-motion data on important earthquakes

Earthquake Stations. Epicentral distance.	M-M Intensities			Mean Accel- eration* cm/sec ²	Dura- tion Sec.	"Destruc- tivity" Factor*	Max. Veloc- ity cm/sec.	Lithologic Foundation	
	Epi- center	Town	Station					Town	Station
<u>Long Beach 1933</u>	10								
Long Beach 17 miles		7-9	9	140?	8	1120?	35/	Recent alluvium, also, well consolidated soil	Alluvium
Vernon 33 miles		7	7	110 70 30	1 3 9	110 210 270	24	Broad alluvial plain	Alluvium
L. A. Subway Terminal 37 miles		6/	6	30	10	300	15/	Thin alluvium on tertiary marine sedi- ment.	Tertiary sediments, well consoli- dated.
<u>Imperial Valley 1940</u>	10								
El Centro, 7 miles		8-	8-	170 110 60	5 5 15	850 550 900	42	Surface silt on alluvium.	Surface silt on alluvium.
Brawley, 17 miles		9						do	do
Imperial, 10 miles		9						do	
Holtville, 7 miles		8-						do	
<u>El Centro, 1934</u>	9								
40 miles		6	6	50	15	750		do.	
<u>Helena (Oct. 31)</u>	8-								
3 miles		8-	6?	110	1.8	200	16	Alluvium and limestone	Limestone

* Considered a measure of destructive energy released. Multiplied by the duration it yields a so-called "destructivity" factor.

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Summary: Strong motion seismograms have now been obtained for three destructive earthquakes (Long Beach, Imperial Valley and Olympia) in each of which the intensity was about VIII. In every case the maximum horizontal acceleration has been around 0.3g and has accompanied periods of 0.3 to 0.4 seconds. The maximum displacement varies from 1 cm at Olympia through 6 cm at Long Beach to 21 cm at Imperial Valley.

Natural periods of the ground

Alluvial fills in valleys appear to have definite periods of vibration. Work in the Leine River Valley at Gottingen, Germany (3, p. 181) showed two periods, one 0.3 to 0.4 seconds, the other 1.2 seconds. This was interpreted to mean that vibration of the upper part of the fill only produced the shorter period, while the longer period was produced by vibration of the upper part with the lower part.

Suyehiro (4), used a seismic vibration analyser consisting of a number of compound pendulums of different natural periods. He found the natural period of the ground at Hongo, in the higher part of Tokyo, to be 0.3 second. At Marunouchi, in the lower part of Tokyo, the prevailing periods were 0.7 to 0.9 seconds.

More recent work in California by Gutenberg (3) indicates that ground periods of 0.2 and 0.3 seconds are outstanding. Periods of 0.5, 0.6 and 1.0 are common.

Macelwane (5) summarizes what was known of ground periods in 1940 as follows:

"The following conclusions may be drawn from the five years of experimentation, 1935-1940. First, the ground generally has many natural periods of vibration which vary from place to place and are usually different in different directions at any given place. Second, most ground resonances cover a fairly wide band of frequencies, thus differing from building resonances which are usually sharp and well defined. Third, resonances are more pronounced on soft, marshy, or made ground and less pronounced on solid rock. Fourth, a structure, such as a building, dam, or tower, forms, together with the ground on which it is built, a coupled elastic system in which vibratory energy is readily transferred to resonating parts. Vibrations induced by a shaking machine in the Hollywood Storage Building were picked up in the ground outside as far away as 1.2 miles. However, the natural periods of the ground seem too numerous to make it practicable so to design buildings that they will avoid all resonance with the ground."

Vibration of a rigid building in an earthquake

Suyehiro and Ishimoto (4) have made the outstanding contribution to our knowledge of how buildings vibrate in earthquakes. Studying the building of the Earthquake Research Institute, Tokyo Imperial University, with vibration meters, they found that the building acted as a rigid body vibrating on the ground bed with the period of the ground bed, 0.3 second.

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This was confirmed by seismograms obtained on the penthouse of the building and on the ground adjacent the building. These showed that the building moved exactly as did the ground, except that the building was insensible to vibrations of about 0.1 second period, which showed on the ground seismogram but not on the building seismogram.

This was a strongly constructed building with 53 pounds of steel framing and reinforcing per square foot of floor space.

Suyehiro concludes that:

"From these facts, it can be inferred that the dynamic stress induced in a strongly constructed rigid building by an earthquake is likely to be equal to the static stress which would be induced, had the building been subjected to the static load of the intensity given by the mass of the building multiplied by the horizontal acceleration of the seismic vibration; and, also, that the component of seismic vibrations having a very short period, say, less than 0.1 sec., is not of much consequence for the building under consideration, although there is the possibility in some cases that such a quick vibration may cause a fairly intense acceleration."

A building of medium rigidity was found to move with the ground as regards period, phase and form, but the amplitudes at the top of the building were 20 per cent to 70 per cent higher than those of the ground, and there was imposed upon them secondary vibrations that probably represent the free period of the building.

A non-rigid building was found to oscillate with its own natural period, the amplitude at the top of the building being several times the amplitude of the ground motion.

In summary, Suyehiro says:

"To recapitulate, in an earthquake a rigid building moves nearly the same as the surrounding ground; a non-rigid building moves with its own period; and the motion of a building of medium rigidity is intermediate between these two."

Possible application at Arco

The writer has not found any information that would indicate what might be expected for the ground period at Arco where bedrock of basalt flows is covered by about 50 feet of unconsolidated dry gravel. This could be learned from shaking tests conducted at the site. In the absence of specific information, the best approximation possible would seem to be to expect maximum acceleration on periods of the order of 0.3 to 0.4 seconds.

If the building is constructed as a strong, rigid unit, it and its contents can be expected to be subjected to the same motion as the ground expect that the inertia of the building may prevent it from responding to all of the short period motions.

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