Preliminary Report and Hazards Analysis

OHIO STATE UNIVERSITY

NUCLEAR TRAINING REACTOR

LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORPORATION, GEORGIA DIVISION

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PRELIMINARY REPORT & HAZARDS ANALYSIS

OHIO STATE UNIVERSITY

NUCLEAR TRAINING REACTOR

SEPTEMBER 1959

LOCKHEED NUCLEAR PRODUCTS

LOCKHEED AIRCRAFT CORPORATION,

Georgia Division, Marietta, Georgia

ERRATA

Page	
--	For all references to Drawing "PD-10-0013" read "PD-11-0013."
4	In the thermal column shutter description, for "0.5 in. boral plate" read "1/4 in. boral plate."
24	In Paragraph 1.8, for "$CaCo_3$" read "$CaCO_3$"; for "al" read "Al."
26	Pages 26, 27, and 28, which contained the Modified Mercalli Intensity Scale of 1931, have been deleted.
37	For "cyclotrooneering" read "cyclotron engineering."
42	In Paragraph 2.7.8, for "Permi age" read "Fermi age."
43	In Paragraph 2.7.15, for "n, process" read "n, γ process."
54	Transpose the "High" and "Low" column headings in the comparative water-level data tabulation.

ADDENDA

The following information is presented to supplement the material contained in the paragraphs indicated.

1.2

Reactor core loading for operation with beam ports and pneumatic tube evacuated, central glory hole in position, all control rods fully withdrawn, all isotope elements in position containing graphite cores, and a total of 1.5% excess reactivity has been calculated as approximately 2965 grams U 235.

1.3

Fuel requirements are given for fully loaded elements. Actually, in initial loading, partial elements will be used to attain required excess reactivity.

1.7.11

Control rods and control rod elements are constructed to prevent the rods from dropping out the bottom of the core in the event they become detached from extension tubes.

Addenda (cont.)

1.8

Process system drawing PD-10-0013 has been replaced by PD-11-0013, which shows the reactor pool system only. The system for the bulk shield pool is identical, except that the temperature control system is not included.

4.3

Maximum credible accident analysis and the selection of the maximum credible accident has been changed to classify the flooding of the beam ports or a step input of 1% ΔK as the maximum credible accident. Inadvertent or deliberate insertion of a fuel element in the glory hole position with the reactor critical is not considered credible with the glory hole element detent mechanism. This results in a less severe accident. Analysis of this excursion will be supplied in a supplementary report.

FOREWORD

This report is in fulfillment of the Atomic Energy Commission requirements for a preliminary hazards summary report. It has been prepared by Lockheed Nuclear Products for the Ohio State University and incorporates information provided by the University.

CONTENTS

FIGURES

INTRODUCTION

This hazards summary report is presented in four sections. The first section, a complete description of the reactor, is followed by a discussion of reactor operation. Section 3 contains a description of the reactor building and its site. The hazards associated with the proposed reactor are presented in Section 4, and the maximum credible accident is discussed.

The following drawings, referred to throughout the report, have been inserted in the back of this publication:

PD-9-0002
PD-9-0003
PD-9-0004
PD-9-0005
PD-9-0009
PD-9-0010
PD-9-0011
PD-9-0012
PD-10-0007
PD-10-0008
PD-10-0013

1. REACTOR DESCRIPTION

This section, describing the reactor, presents first the general specifications for the reactor and its components. Following this is a description of the core, fuel requirements, shielding, experimental facilities, pool and bulk shield construction, instrumentation and control, and process system.

1.1 GENERAL SPECIFICATIONS

Reactor

Type	Swimming pool (BSR)
Design Power Level	10 kw continuous
Maximum Thermal Flux in Fuel Region	2.23×10^{11} n/cm^2-sec
Average Thermal Flux in Fuel Region	1.0×10^{11} n/cm^2-sec
Fuel Elements	
Type	Flat plate
Number of plates	10 per element
Dimensions	3 x 3 x 35 in. overall
Fuel material	U-A1 alloy, Aluminum clad
Enrichment	93% U-235
Cladding	0.036 in. type 1100 aluminum
Number of standard elements	20
U-235 per element	140 grams
Number of special elements	4
U-235 per element	84 grams

Constant Volume Critical Mass of U-235	2840 grams
Operating Mass of U-235	3136 grams
Moderator	Light water
Reflector	Light water and graphite
Thermal neutron lifetime	7×10^{-5} sec

Excess Δk_{eff}	Equilibrium X_e	0.03%
	Fuel burnup	0.12%
	F. P. Poisons	0.05%
	Experiments	1.30%
	Total Excess	1.50%

Reactivity Effects

Temperature coefficient	-2.1×10^{-4} Δ k/k per ^{o}C
Void coefficient	-2.8×10^{-3} Δ k/k per % void
Regulating rod	0.47% (calculated)
Shim safety rods	-9.04% for 3 (calculated)

Control Rods

Number of regulating rods	1 - 304 stainless steel
Number of shim safety rods	3 - boron carbide
Operating rates	
Regulating rod	14.4 in/min
Shim safety rod	3.6 in/min
Scram	500 milliseconds (from signal to complete insertion)

Nuclear Instrumentation

Startup channel	
Detector	Fission chamber
Range	2.5 to 2.5×10^{5} n/cm^{2}-sec
Indicators	Log CRM
Range	1 to 10,000 counts/sec
	Recording Potentiometer
	Binary scaler

Log-N period channel	
Detector	CIC
Range	10^4 to 10^{10} n/cm^2-sec
Indicators	Log-N meter
Range	1×10^{-1} to 3×10^5 watts
	Period meter
Range	∞ to 3 sec period
Power channel	
Detector	CIC
Range	10^4 to 10^{10} n/cm^2-sec
Indicator	Linear level meter
Range	1×10^{-1} to 10^6 watts
Safety channels	
Detector (Period)	CIC
Range	10^2 to 10^{10} n/cm^2-sec
Detectors (Level)	2 - PCP ion chambers
Range	5×10^5 to 5×10^{10} n/cm^2-sec
Rod position indicators	
Detectors	
Regulating	Slide wire potentiometer
Range	0-100 ohms
Shim safety	Slide wire potentiometer
Range	0-100 ohms
Indicators	Digital registers
Range	0-99.99 inches
Area monitors (4)	
Detectors	Ion chambers
Range	0-125 mr/hr
Indicators	Remote meters
Range	0-125 mr/hr

Shielding	
Type	Water, portland and barytes concrete
Maximum dose rates	
Surface of concrete	0.25 mr/hr
Surface of water	2.5 mr/hr
Experimental facilities	
Bulk shielding facility	
Fission plate assembly	
U-235 content	3850 grams
Thermal power	~10 watts
Heating coil power	100 watts
Thermal column	
Shutter	0.5 in boral plate
Shield	3 in lead
Flux monitor	1/2 in x 6 in BF_3 chamber
Main thermal column	
Removable stringers	13 - 4 x 4 x 57 in bars
Shielding	
Core end	3 in lead
Outer face	2 ft barytes concrete and 1/4 in boral
Thermal flux	8.8×10^{10} n/cm^2-sec at core
Beam ports	2
Size	6 in diameter stepped to 7 in
Water Purification System	
Water capacity	
Reactor pool	11,200 gal.
Bulk shielding pool	5800 gal.
Pumps (2)	Centrifugal
Capacity	
Normal at 100°F and 70 ft head	12 gpm
Vacuum at 100°F and 50 ft head	50 gpm

Motor rating	1 hp
Ion exchangers (2)	Mixed bed (replaceable cartridge type)
Rated capacity	15,000 grains as C_aCO_3
Flow rate (Total)	12 gpm
Design pressure	100 psi
Heat exchanger	Shell and tube (Water chiller)
Heat load	25,000 Btu/hr
Design pressure	
Tube	150 psi
Shell	225 psi
Pool temperature control	65 - 75°F
Filter	Cartridge
Maximum flow	80 gpm
Pressure drop	2 psi

1.2 REACTOR CORE

The arrangement of the core and associated experimental facilities for the proposed reactor was designed to meet the requirements of the Ohio State University for a flexible, safe, research and training reactor. The design configuration for the pool-type core shown on Drawings PD-9-0002 and PD-10-0007 consists of a 5-by-5 array of fuel elements reflected on three sides with graphite and the beam port side, top, and bottom with water. The core contains 20 standard, 10-plate fuel elements, 4 rod-well fuel elements, and a central, canned-graphite, glory-hole element. One row of canned graphite reflector elements designed to hold samples for isotope production is added along the pool face of the core.

With the exception of the fuel loading, the material composition of the core is similar to that of the BSR, and it is identical to that of the Critical Experiment Reactor and the Radiation Effects Reactor at the Georgia Nuclear Laboratories. The average metal-to-water volume ratio in the fuel element region of the core is 0.766, which yields a relatively dry core. This choice results in a critical mass slightly higher than the theoretical minimum, but yields several important advantages for the intended application. Neutron leakage in the fast and epithermal energy groups is greater (per unit power) than would be obtained with a lower metal-to-water ratio, because of the larger neutron age. Also, negative void and temperature coefficients are assured, since the core is slightly undermoderated at room temperature.

After the physical dimensions and the design configuration of the core were established, a range of fuel loadings was investigated to determine the nominal fuel element loading. Total U-235 loadings of 2240 gms, 3136 gms, and 3584 gms were chosen to establish a curve of effective multiplication versus fuel content in the core. These correspond to fuel element loadings of 100, 140, and 160 grams of U-235. The 4 fuel elements containing rod wells had 6 fuel plates rather than 10 as in the 20 standard elements and the plate spacing was slightly less than in the standard elements.

Therefore, for every loading two fuel regions were considered. For each region at each loading the MUFT-IV, IBM-704 computer code was used to obtain the fast group nuclear constants of D_1, Σ_a and Σ_r. This particular code employs 54 neutron energy groups from 10 mev to 0.0625 ev and uses the Watt's fission spectrum as the source of neutrons in the diffusion calculation. The thermal constants, $\nu\Sigma_{f_2}$, D_2, and Σ_{a_2}, were obtained using the SOFOCATE IBM-704 code. This code averages the constants over a Wigner-Wilkens spectrum and accounts for the hardening of the spectrum resulting from neutron absorption. Similar constants were obtained for water and graphite.

The nuclear constants thus obtained were used as input into the IBM-704 PDQ code, which solves the two-dimensional diffusion equations for a multiregion few group problem. The rod wells and glory hole element were put in as distinct regions. The vertical buckling was calculated using an iterative technique, which consisted of taking the flux averaged constants and the horizontal plane buckling from the two-dimensional PDQ problem and using them as input in a one-dimensional calculation in the vertical direction. The code used for the vertical direction was the IBM-704 WANDA code which solves the multiregion few group problem. The WANDA calculation gave a multiplication constant from which the vertical buckling was obtained; this vertical buckling was then used as input for a second PDQ calculation. The process was repeated until a consistent value of the multiplication constant and total buckling was obtained. The resultant curve of k_{eff} versus fuel loading is shown in Figure 1.

The curve shown applies to a configuration in which all of the control rods are fully withdrawn, both beam ports and the pneumatic tube are removed (or filled with water), and the central glory hole element is in place and filled with water. Criticality would be attained in this configuration with a core loading of 2840 gms U-235, or approximately 127 gms per standard element. A design loading of 140 gms per element, or 3136 gms total, is required, however, to provide sufficient reactivity to overcome the effects of the beam

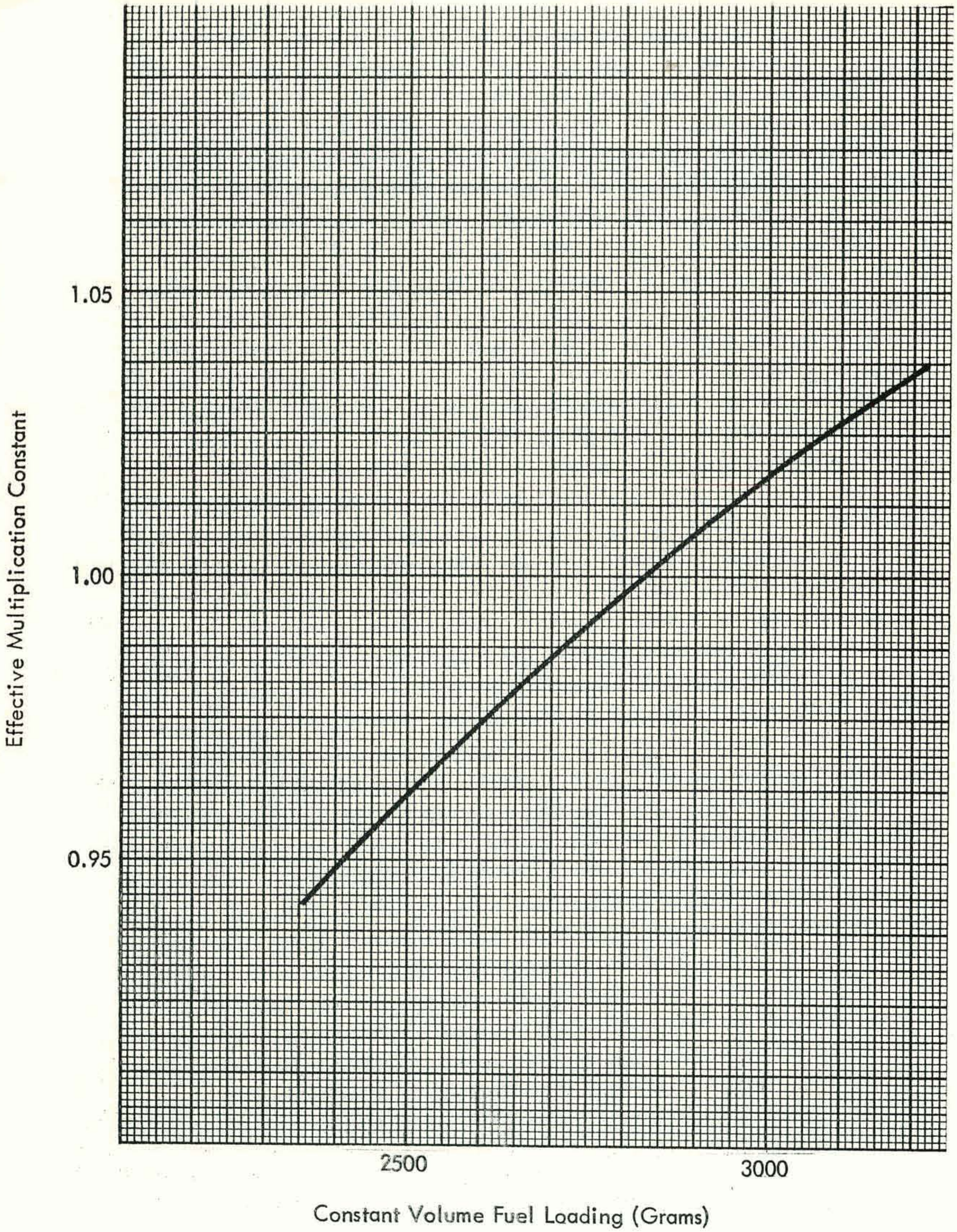


FIGURE 1 EFFECTIVE MULTIPLICATION CONSTANT VS CONSTANT VOLUME FUEL LOADING

ports, pneumatic tube, temperature, poison, and experiments placed in and near the core.

An investigation of the literature available indicated that the best estimate of beam port reactivity coefficients would be obtained from data measured at the BSR at Oak Ridge and reported in ORNL-1871. Some of the results are reported in terms of the simulated beam port area. From these data, the worth of the beam ports and the rabbit tube is estimated to be 1.0% Δk. The clean, cold reactivity is then estimated to be approximately 1.5% Δ K. Equilibrium xenon and a nominal pool water temperature rise will reduce the reactivity available for experiments to about 1.3%.

The control rod worth was obtained by putting a "black" boundary condition on the three shim safety rod areas. This boundary condition applied only to the thermal neutrons. The resulting PDQ calculation gave a total rod worth of 9.04% Δ k, which is less than the actual rod worth since the epithermal absorption in the B_4C, which was not accounted for, would increase the reactivity of the rods to approximately 9.5% Δ k.

The regulating rod is a waterfilled 1/16-inch stainless steel shell. A flux depression factor for the steel wall was used to weight the steel volume fraction in homogenizing the water and steel in the control rod area, and the homogenized constants were placed in the explicit regulating rod area in the core for a PDQ calculation. The change in reactivity due to the regulating rod was 0.47% Δk.

Additional analytical work was performed to calculate the temperature coefficient, the worth of the central element, the most valuable position of a fuel element, and the approximate number of fuel elements necessary to go critical. The temperature intervals were from 60°F to 100°F. The value was negative and equal to $2.1 \times 10^{-4} \Delta k/^{\circ}C$. The reactivity due to replacing the central glory hole element by a fuel element was computed by the PDQ code; it indicated that the worth was 2.1% Δk. This central position is not the most important one, however, because the maximum values of the flux occur away from the core center at a point nearer the graphite reflector than the water reflector. The two-dimensional flux plots indicate that when the control rods are withdrawn, the maximum flux occurs in the fuel elements which are located between the rods and nearest to the graphite, where the flux is about 10% higher than in the central element. The worth of these two fuel element positions was computed to be approximately 2.6% Δk.

Several configurations of fuel elements that could be assembled during the initial startup procedures were analyzed to find the approximate point at which criticality would occur. These configurations all included a fuel element in the central position; that is, no glory hole element was present. The analysis indicated that the core would probably achieve criticality with the 4 rod-well fuel elements and 16 standard fuel elements in place.

1.3 FUEL REQUIREMENTS

The fuel requirements are discussed as to standard fuel elements, control rod fuel elements, and fission plate assembly.

1.3.1 Standard Fuel Elements

Fuel elements to be used in the Ohio State University reactor will consist of 10 fuel plates, and each will contain 140 grams of U-235. Each fuel plate will contain a fuel section 0.036 inch thick, clad on each side with 0.036-inch thick type 1100 aluminum; the active fuel length will be 24 inches. The fuel plates will be fabricated by the picture frame method and will be joined mechanically to the two sideplates. This assembly will be fastened at the bottom to a cylindrical transition end piece that will fit into the grid plate.

Elements mechanically assembled as described above are stronger and more rigid than elements fabricated by brazing the fuel plates to the side plates. Since the mechanical assembly will be performed at room temperature, the stiffness and rigidity of the cold-rolled aluminum plates will be maintained. This method of assembly will eliminate the severe localization of corrosion attack that would otherwise result from brazing.

Burnup of the U-235 at the normal operating power level (10 kw) of this reactor will not determine the life of the fuel elements; corrosion of the cladding will be the determining factor. The fuel plate cladding will have a uniform corrosion rate less than one mil per year in demineralized water at 50°C. However, corrosion and pitting of the cladding, hence the fuel element life, will be extremely sensitive to water purity. With continuous control of the pool temperature, and with demineralized water resistivity greater than 330,000 ohm-cm, a life of 6-8 years may be expected for the elements.

1.3.2 Control Rod Fuel Elements

Control rod fuel elements will be modified standard fuel elements with four central fuel plates eliminated to provide a channel for vertical movement of the rods. Each element will contain 84 grams of U-235 distributed equally among six fuel plates. These elements are readily distinguished from the standard fuel elements by means of a magnet guide tube bolted to each of the assemblies and extending upward about 15 feet, as shown in Drawing PD-9-0002.

1.3.3 Fission Plate Assembly

This item is described under the Bulk Shielding Facility section. The fuel content of the fission plate will be approximately 4000 grams of U-235.

To summarize the fuel requirements for the reactor facility, the following table shows the allotment of U-235 among the various items.

TABLE II

Fuel Requirements for the Reactor Facility

20 fuel elements, 140 grams each		2800 grams
4 control rod fuel elements, 84 grams each		336 grams
Fission plate		4000 grams
4 Partial elements		252 grams
1 @ 42 gms		
1 @ 56 gms	Total	7388 grams U-235
1 @ 70 gms		
1 @ 84 gms		

1.4 SHIELDING

The biological shield shown in Drawing PD-10-0007 will consist of light water, structural concrete, and barytes concrete. Shielding over the core will be provided by 15 feet of light water, which will reduce the radiation level to 2.5 mr/hr when the core is operating at 10 kw. The concrete biological shield has been designed for a maximum radiation level at any point along its outer lateral surface of 0.25 mr/hr at 10 kw.

There will be 10 feet of water above the fission plate assembly and varying amounts of water and concrete in the lateral directions. Radiation levels at the surface of the bulk shielding pool and at the outside surfaces of the walls will not exceed 0.25 mr/hr for 10-kw operation of the reactor and a maximum of 100 watt operation of the fission plate.

Calculations indicate that a thermal shield will not be required for 10-kw operation.

1.5 EXPERIMENTAL FACILITIES

The experimental facilities with this reactor are described in the following paragraphs.

1.5.1 Thermal Column

The thermal column will be constructed of 4-inch by 4-inch machined graphite bars. Thirteen of these bars will be removable for foil or sample insertion. As shown in Drawing PD-10-0008, the column will be stepped twice to prevent radiation streaming. A 1/4-inch thick boral liner will surround the graphite to prevent neutrons from escaping into and activating the surrounding concrete and air. A 1/4-inch thick aluminum plate with boral liner and gasket will provide an airtight seal over the front face of the thermal column.

The supporting liner will be constructed of heavy steel plate and reinforcing members to assure proper fit of graphite as well as to prevent any shifting or cracking. A movable concrete door will provide shielding at the thermal column face to limit the dose rate to a maximum of 0.25 mr/hr at 10-kw operation. Four hours after shutdown from extended operation at 10 kw, the radiation level at the column face with the door open will be approximately 20 mr/hr.

1.5.2 Bulk Shielding Facility

The bulk shielding facility will consist of a short graphite thermal column, a shutter mechanism, a neutron monitor, an enriched uranium fission plate assembly, and a concrete-walled water pool. The pool will be built integrally with the reactor pool and also will be lined with polyester reinforced with fibreglass cloth.

The thermal column proper will consist of a 36-inch square boral-lined steel box between the pyramidal stub and the inner surface of the bulk shielding pool. Removable blocks of machined graphite will fill the box to provide replacement with D_2O or additional shielding if required. A gasketed aluminum plate will ensure watertight closure of the assembly.

The shutter mechanism shown in Drawing PD-10-0008 will provide for hand motor driven interposition of a boral plate between the fission plate and the core. An aluminum plate having a thickness approximately equal to that of the aluminum in the boral plate will follow the shutter.

Drawing PD-9-0009 shows the detail of the fission plate assembly. The uranium plate will be composed of 93% enriched uranium in a circular plate 28 inches in diameter and approximately 0.075 inch thick, including a clad of 0.025 inch of aluminum. An electrical heater plate capable of supplying 100 watts of power will be used for calibrating the fission plate. Resistance thermometers will be attached to the fission plate to provide for a comprehensive indication of the temperature. The fission plate and the heater plate will be in intimate contact, but they will be insulated both thermally and electrically from the front and rear cover plates. With 100 watts of power emitted from either plate, the separation distances are such that the difference between the ambient temperature of the pool water and the temperature of the fission plate will not exceed 75°F.

1.5.3 Beam Ports

Two beam ports will abut upon one face of the core as shown in Drawing PD-10-0007. Details of the ports with the removable gamma shutter drawer are shown in Drawing PD-9-0010. Neutron fluxes quoted in the specifications are with no shield plugs in the ports; however, each port will be provided with shield plugs shown in Drawing PD-9-0011 to reduce the radiation level to 0.25 mr/hr at the exterior face. With no shield plugs present, the lead gamma shutter will reduce the radiation level at the exterior face to 25 mr/hr two hours after shutdown from extended operation at 10 kw.

The design of the beam port and the gamma shutter mechanism will facilitate shield manipulation during experimentation at the ports. Shutter drive consists of a continuous loop drive cable, lead screw, hand drive motor, and position indicator. The use of relatively small, easily handled, plug-transfer casks will be possible with the sectional plugs.

1.5.4 Isotope Irradiation Facility

The facility will consist of canned graphite elements with a central hole designed to receive sample capsules as shown in Drawing PD-9-0002. Insertion of these elements is restricted to the five positions on the pool face of the core. A sleeve-capsule handling tool will permit withdrawal of the central sleeve containing up to six capsules, or single capsules may be withdrawn. When no samples are being irradiated, a graphite plug is placed in the central hole.

1.5.5 Glory Hole

An element of canned graphite with a 1-1/2-inch diameter hole for vertical access will occupy the central grid position. This element will permit small sample irradiations in the central flux of the core or the insertion of an aluminum tube with a 1-1/2-inch outside diameter extending to the surface of the pool. A detent mechanism is provided to prevent withdrawal of this element without unloading of the core. Also, no fuel element can be inserted in the central grid position.

1.5.6 Rabbit Tube System

The system will be a vacuum-operated, single-tube type, pneumatic rabbit device as shown in Drawing PD-9-0012. A vacuum system will be used in preference to a pressure system so that evacuated air will be carried away rather than expelled at the position of operation. The solenoids to return the rabbit to the receiving terminal will be operated either by manual push button or by the automatic trigger actuated by the timer mechanism. In the event of an error in timer setting or unintentional release of operation of the system, a manual control can be used to override the automatic timer control.

1.6 POOL AND BULK SHIELD CONSTRUCTION

The shield construction of the reactor pool, the fuel element storage pit, and the bulk shielding facility are discussed here.

1.6.1 Reactor Pool

Drawing PD-10-0007 shows the reactor pool configuration along with that of the bulk shielding pool. The experimental facilities and the fuel element storage pit are shown also in this drawing. These items will be described in subsequent paragraphs.

The pool will be lined completely with polyester reinforced with fiberglass to prevent leakage, leaching of the materials in the concrete by the pool water, and facilitate decontamination and repair of the walls. Provisions for recirculating and control of the pool water will be provided. Draining of the pools can be accomplished only by a separate system under control of the reactor supervisor.

1.6.2 Fuel Element Storage Pit

A fuel element storage pit will be located below the pool floor and at the end opposite from that of the core as shown in Drawing PD-10-0007. The pit will be large enough to accommodate two complete fuel charges and will contain neutron absorbing material to prevent inadvertent criticality. Shielding for the radioactive fuel will be provided by an 8½-inch thick lead cover over the pit, thus permitting personnel to work in the reactor pool when it is drained. The radiation level directly over the cover, with a full core of elements in the pit, is expected to be 15 mr/hr 24 hours following shutdown after prolonged operation at 10 kw. Locating the pit in this manner will preclude the use of fuel transfer casks or of a fuel transfer to the bulk shielding pool when work in the reactor pool becomes necessary.

1.6.3 Bulk Shielding Facility

The bulk shielding facility, which is described in the Experimental Facilities section, has been designed to provide a fission plate power level of approximately 10 watts and a fast neutron dose background of less than 10%. The design analyses and procedures used to optimize the facility consisted of a series of four-group diffusion calculations.

The question of reactivity coupling between the core and the fission plate was investigated by performing a few-group diffusion analysis in slab geometry using the WANDA code with both the core and the fission plate included as explicit source regions. Thermal neutron constants averaged over a Maxwellian distribution were used for both the uranium plate and the boral plate which is attached to the fission plate assembly on the shield pool side. Although the diffusion theory approximation is not valid in thin regions with high absorption, it can be shown that the major effect is to underestimate the flux depression in the fission plate and overestimate the flux depression in the medium. Correcting for both these effects, the change in the effective multiplication of the coupled system due to opening and closing the shutter is estimated to be 10^{-4} per cent Δk. It is thus concluded that the fission plate is effectively uncoupled from the reactor and will not affect the reactivity by any measurable amount.

1.7 INSTRUMENTATION AND CONTROLS

The design of the instrumentation and control system proposed for the Ohio State University reactor is based on systems in use at such reactors as the Bulk Shielding Reactor, the Tower Shielding Reactor, and the Pennsylvania State University Reactor. All of these systems have operated satisfactorily and safely for many years.

Drawing PD-9-0003 shows a block diagram of the instrumentation system, which will consist of three operational channels and two safety channels. The operational channels will include a startup channel, a log-N period channel, and a linear level channel. The log-N period channel will actually serve as a third safety channel, since it will be connected to a third composite safety amplifier by the sigma bus. Scram systems will be interconnected into these channels to effect reactor shutdown in the event of an emergency or abnormal conditions. An annunciator and alarm system will be included to indicate specific trouble.

1.7.1 Startup Channel

The startup channel will be used to monitor the neutron flux from the source range to an overlap of the linear level and log-N ranges. This channel will share a period recorder with the log-N channel, and a remote log count-rate indicator will be mounted on the control console. An interlock switch in the log count-rate recorder will prevent startup when the neutron source strength is below a preset count. The startup channel range can be extended to include full-power operation by raising the fission chamber a maximum of 24 inches into a cadmium shield by means of a drive mechanism similar to the control rod drive units.

1.7.2 Log-N Period Channel

The log-N channel will indicate the reactor power level on a six-cycle log scale from 0.1 watt to 300 kw. The log-N amplifier will also initiate the fast scram signal to a composite safety amplifier in the event of a short period. Should there be a loss of high voltage to the compensated ion chamber, a slow scram signal will be initiated. The compensated ion chamber and the log-N amplifier will be the type developed at Oak Ridge.

1.7.3 Linear Level Channel

The linear level channel will be capable of measuring neutron flux in a reactor operating range of 0.1 watt to 1 megawatt. Loss of high voltage to the ion chamber will cause a slow scram.

The micro-microammeter will be the vacuum tube electrometer type covering a range of 10^{-3} to 10^{-11} amperes. A remote switch at the control console will be used to select the range.

1.7.4 Safety Channels

Three composite safety amplifiers of the type developed at Oak Ridge will be incorporated into the safety channels. They will provide, in addition to safety protection, a source of controlled magnet power. Each safety amplifier will be connected to a common sigma bus to provide a scram signal from either amplifier to all magnet amplifiers simultaneously. A period scram signal from the log count-rate meter of the startup channel will feed into the slow scram circuit. High-power-level, fast scram protection will be provided by two parallel-circular-plate ion chambers in connection with high-level trips in the composite safety amplifiers. The PCP ion chambers will be the type developed at Oak Ridge.

1.7.5 Automatic Shutdown Systems

Two types of scrams will be included in the control system to effect shutdown of the reactor in the event of emergency conditions. A fast scram will be accomplished when a short period signal or a high power-level signal serves through the sigma amplifiers to remove magnet current.

A slow scram will remove power from the magnet amplifier by means of a relay. This relay will be actuated by the following controls:

Bulk-shielding area manual switch

Rod-drive area manual switch

Thermal column area manual switch

Beam-hole area manual switch

- Reactor console manual switch
- Compensated chamber power supply #1 high-voltage relay
- Compensated chamber power supply #2 high-voltage relay
- Minimum source strength switch (permissive circuit)
- Rod-drive area monitron relay
- Thermal column area monitron relay
- Servo error relay
- Low-level period relay
- Two scram connections for incorporation in experiments that may be desired at a later date

In addition to the slow scram provisions, a key switch will be included in the magnet power supply circuit to prevent unauthorized withdrawal of the shim-safety rods.

1.7.6 Annunciator and Alarm Systems

When a scram condition occurs, or when other trouble arises, an alarm (buzzer) will sound, and a lamp or lamps will be lighted on the control console indicating the source of the trouble. An annunciator acknowledge button will be used to turn off the buzzer. If the trouble results in a scram, a scram reset switch must be actuated before magnet power can be reapplied. In the event of trouble other than a scram, the corresponding lamp-switch will be pushed to extinguish the lamp and reset the annunciator system after the trouble has been corrected. If the trouble is not corrected, the indicator light will remain lighted.

An annunciator test switch will be provided for checking the lamps and the buzzer. The annunciator will provide the following indications:

Scram-console

Scram-bulk shielding area

Scram-rod drive area

Scram-thermal column area

Scram-beam port area

Scram-experiment #1

Scram-experiment #2

Scram-servo trouble

Scram-low level period

High radiation level -- rod drive area

High radiation level -- thermal column area

High radiation level -- beam hole area

Fast scram -- high power level

Fast scram -- period

Trouble -- safety amplifier #1

Trouble -- safety amplifier #2

Trouble -- safety amplifier #3

Source missing

An evacuation alarm will consist of a klaxon operated from the control console by a switch. This klaxon will sound automatically if the annunciator acknowledge button is not pushed within a predetermined time after a trouble signal is received.

1.7.7 Shim-safety Rod Controls

Three identical control channels will be used for the shim-safety rod system. A rotary switch will select an individual rod to be controlled. All shim-safety rods will be inserted into the core by a gang lower switch when shutdown is desired or when the rods are recovered after a scram. This switch will not cause the control rods to be gang-raised under any circumstance.

Control console indicators for each rod will include the following: magnet coupled, drive motor on, upper limit, and lower limit. Rod positions will be indicated by a digital readout device having a resolution of 0.01 inch.

1.7.8 Regulating Rod Control

Manual control of the regulating rod will be possible; on the other hand, power level control at any power from 0.1 watt to more than 10 kw will be provided by a servo system activated by the linear level recorder. The servo system may be actuated only when the reactor power level is within plus or minus 5% of the set point. A meter relay included in the servo error meter will generate a scram signal at plus or minus 8% servo error.

1.7.9 Area Monitors

Drawing PD-9-0003 shows a block diagram of the monitoring system, manufactured by the Victoreen Instrument Company. These monitors will measure the radiation level in the area and will be set to scram the reactor if the dose rates in the vicinity of the chambers exceed predetermined levels. Areas monitored are the following:

Thermal column face

Beam port openings

Pool surface above core

Ion exchange beds and process system

1.7.10 Control Console

The reactor console shown in Drawing PD-9-0004 has been designed to provide maximum visibility of the reactor and maximum accessibility of the controls and indicators. All indicators and controls necessary for startup and shutdown operations will be located in one group in front of the operator. An enlarged view of the control panel is shown in Drawing PD-9-0005.

Colors for the indicator lights on the console will show the operator the status of the reactor at a glance. All scram and warning lights will be red. Operating procedures, as well as interlocks, will keep the operator from withdrawing the control rods when a red light is showing.

1.7.11 Control Rod Systems

Oval aluminum cans 7/8 x 2-1/4 x 28-5/8 inches containing boron carbide power will be used for the shim-safety control rods. Each rod will contain two aluminum tubes filled with lead, which will add ballast to facilitate the rapid fall of the rod. A shaft attached to each rod will support an iron armature and a pneumatic shock absorber assembly which will snub the rod after a scram.

The regulating rod will be a hollow oval of type-304 stainless steel having approximate dimensions of 7/8 x 2-1/4 x 28-5/8 inches overall. Penetrations in the top and bottom end plugs will allow water to enter the rod.

Control rod drive mechanisms will be supported above the core as shown in Drawing PD-9-0002. The maximum stroke of these units will be 24.0 inches.

Waterproof electromagnets will suspend the shim-safety rods when coupled to the armatures at the top of the rod extensions. The armatures and shock absorbers are always above water level. The electromagnets will be raised and lowered by the control rod drive mechanisms. The regulating rod will be mechanically connected to the rod drive and will not be released in the event of a scram. Overall full travel scram time will be less than 500 milliseconds.

1.8 PROCESS SYSTEM

The water purification system will limit system corrosion rate by pH control and corrosion product buildup by ion exchange. A flow diagram of the system is shown in Drawing PD-10-0013.

A conservative estimate of the corrosion rate to be expected is 4.5 milligrams per square decimeter per day for the first ten days of operation; after that the formation of a protective oxide coating will reduce the rate to 2.5 milligrams per square decimeter per month. The major corrosion product to be removed will be the al^{+++} ion. Water purity is intended to be maintained at 2 ppm as $CaCo_3$ or 0.36 ppm as al^{+++}. The pH of the water can be maintained at 6.5 by the weakly basic anion resin in the mixed-bed ion exchanger when the measured water conductivity is 2.2 micromhos per ppm electrolyte as $CaCo_3$.

After the initial filling has been completed, 16 hours per day for 10 to 14 intermittent days of recirculation through the ion exchangers will be needed to maintain a water purity of 2 ppm. Normal operation of the water purification system will consist of recirculating the reactor and bulk shielding pool water through the ion exchanger for about 4 hours and 2 hours respectively each day at a rate of 12 gpm, and will be started (after the completion of the time period as stated above).

The heat exchanger system has been designed to maintain the reactor pool temperature from 65 - 75°F during continuous operation at 10 kw. City water will be used as the secondary cooling medium of water chiller and will be dumped into the OSU sewer system.

MODIFIED MERCALLI INTENSITY SCALE OF 1931

All intensities used by the Coast and Geodetic Survey refer to the Modified Mercalli Intensity Scale of 1931.[1] The abridged version of this scale is given here with equivalent intensities according to the Rossi-Forel scale.

I. Not felt except by a very few under especially favorable circumstances. (I Rossi-Forel scale.)

II. Felt only by a few persons at rest, especially on upper floors of buildings. Delicately suspended objects may swing. (I to II Rossi-Forel scale.)

III. Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motorcars may rock slightly. Vibration like passing of truck. Duration estimated. (III Rossi-Forel scale.)

IV. During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls make creaking sound. Sensation like heavy truck striking building. Standing motorcars rocked noticeably. (IV to V Rossi-Forel scale.)

V. Felt by nearly everyone, many awakened. Some dishes, windows, etc., broken; a few instances of cracked plaster; unstable objects overturned. Disturbances of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop. (V to VI Rossi-Forel scale).

VI. Felt by all, many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight. (VI to VII Rossi-Forel scale.)

VII. Everybody runs outdoors. Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving motorcars. (VIII Rossi-Forel scale.)

VIII. Damage slight in specially designed structures; considerable in ordinary substantial buildings with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Persons driving motorcars disturbed. (VIII+ to IX - Rossi-Forel scale.)

IX. Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb; great in substantial buildings, with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken. (IX+ Rossi-Forel scale.)

X. Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations; ground badly cracked. Rails bent. Landslides considerable from river banks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks. (X Rossi-Forel scale.)

XI. Few, if any, (masonry) structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipelines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.

XII. Damage total. Waves seen on ground surfaces. Lines of sight and level distorted. Objects thrown upward into air.

Epicenter maps. - Figure 12 is designed to show the existence of destructive and near destructive earthquakes in the United States through 1954. The smallest dot indicates the shock was strong enough to overthrow chimneys or affect an area of more than 25,000 square miles (intensity VII to VIII); the largest solid dot may be associated with damage ranging from several thousand dollars to one hundred thousand dollars, or to shocks usually perceptible over more than 150,000 square miles (intensity VIII to IX); the smaller encircled dots represent damage ranging from approximately one hundred thousand to one million dollars, or an affected area greater than 500,000 square miles (intensity IX to X); the larger encircled dots represent damage of a million dollars or more; or an affected area usually greater than 1,000,000 square miles (intensity X to XII).

[1]Modified Mercalli Intensity Scale of 1931. Harry O. Wood and Frank Neumann, Bulletin of the Seismological Society of American, vol. 21, No. 4, December 1931.

2. REACTOR OPERATION

The administrative organization and procedures that have been established for reactor operation are discussed first in this section, along with a summary of the professional backgrounds of the staff to be associated with the reactor. Following this are descriptions of the pre-operational tests, startup procedures, normal operations, health physics requirements, emergency procedures, and experimental program.

2.1 ADMINISTRATIVE ORGANIZATION AND PROCEDURES

The safe and orderly operation of a reactor, with proper coordination of all activities centered around it, requires a unified organization headed by a competent supervisor. He must direct technical, administrative, and scientific activity involving the reactor; establish operating rules and controls; guide the reactor operations; schedule operations to ensure that maintenance, instruction, research, and services are in proper balance; and ensure that all proposed operations are reviewed for safety. His authority is from the Office of the Vice President, Instruction and Research. The supervisor must understand fully the technical operations to be made with the reactor. It is especially important that he have sufficient breadth to consider impartially, as well as understand, a wide range of experiments or applications in determining programs and schedules. The need for frequent, on-the-spot decisions demands a single individual exercising full authority, rather than a committee. While a university reactor operations committee will set broad policy, the conduct of actual operations should be the responsibility of a single individual with adequate authority to assign personnel and with recognized competence to make decisions.

2.1.1 Reactor Supervisor

The Ohio State reactor is to be under the immediate operational control of a Reactor Supervisor whose duties are as follows:

> Supervise the technical, administrative, and scientific activities at the reactor
>
> Review all proposed operations in order to ensure that all reasonable requirements of safety to students, personnel, public, and equipment are met and that the proposed operation is consistent with the objectives of the OSU reactor program
>
> Schedule times at the reactor for maintenance, instruction, research and development, or service, to achieve a proper balance between the requirements of each in the long-range interest of all phases of the OSU reactor program
>
> Enforce operating rules and controls governing activities at the reactor, including the safe and orderly handling of students and visitors to the reactor
>
> Submit periodic reports on the operation of the reactor to the Office of the Vice President, Instruction and Research, and special reports on any unusual operations or circumstances during operation of the reactor which may result in a major interruption of the program or which will have special significance in relations with other University organizations, the AEC, or the public.

2.1.2 Scientific Staff

The scientific staff will normally be made up of OSU employees of faculty rank. Under special circumstances, OSU research associates and research assistants may be assigned temporarily to the staff. Visiting faculty from other institutions may be temporarily assigned to the scientific staff. Under the supervision of the Reactor Supervisor, this staff will plan and perform experiments and supervise students performing experiments.

2.1.3 Reactor Operators

The reactor will be operated only by persons duly licensed pursuant to the regulations of Part 55, Title 10, of the Atomic Energy Act of 1954. In general, operators will be of an educational and experience level approaching that required for faculty status in nuclear physics or nuclear engineering.

2.1.4 Students

A project or individual experiment proposed by a student will be supervised by a member of the scientific staff, and the request-for-operation form submitted in order to perform such an experiment or project must be countersigned by a member of the scientific staff.

2.1.5 Scheduling of Reactor Experiments

In accordance with the reactor program originally proposed for the Ohio State University Reactor, in the University's request to the AEC for a grant, first priority will be given to class instruction. Specific times will be scheduled for routine preventive maintenance.

Initially, reactor operation will take place only from 8:00 a.m. to 5:00 p.m., Monday through Friday. At least one day per week of this time will be devoted to maintenance. In addition, a daily instrument checkout of prescribed thoroughness will take place before the reactor is operated.

After the reactor has been turned over to the Ohio State University, a series of standard experiments will be conducted to determine the characteristics of this particular reactor facility. These experiments will comprise flux maps of the core and experimental facilities, power calibrations, rod worth determinations, fuel worth determinations, etc.

2.1.6 Proposals for Reactor Operations

Before any reactor operation is performed, the experimenter will submit a "Request for Reactor Operation" form to the reactor supervisor. This form, properly filled in, will state the purpose, procedure, apparatus, intended power-level history, reactor conditions, and expected results of the experiment, with supporting reasons. The supervisor will review the request and consult members of the scientific staff if needed to establish the type of experiment. The experiments will be categorized as either developed or new experiments.

A developed experiment is any experiment which has been performed before, and for which a written description of apparatus, procedures, pertinent data, and results are on file with the reactor supervisor. Whenever a developed experiment is repeated, it will be necessary to ensure that it will be done in the standard manner or that any changes involve no issue of safety. If a question of safety exists because of changes in method of performance, the experiment must be treated as a new experiment.

A new experiment is any experiment not meeting the tests of a developed experiment; it must then be reviewed for safety. The reactor supervisor will decide upon the safety of a proposed experiment unless review by the reactor operations committee is requested by either the reactor supervisor or a staff member.

Upon satisfactory completion of review of a proposed experiment, the reactor supervisor will schedule a tentative time for its performance. Any dissatisfaction with the judgment or action of the reactor supervisor may be referred to the committee on reactor operation.

Each request-for-operation form will be signed by the experimenter. If the experimenter is a student, the request form will be countersigned by the staff member acting as advisor to the student.

REQUEST FOR REACTOR OPERATION

Date ____________________

TO: Supervisor, OSU Educational Reactor

Request is hereby made for operation of the OSU Educational Reactor on ____________________, as described below.

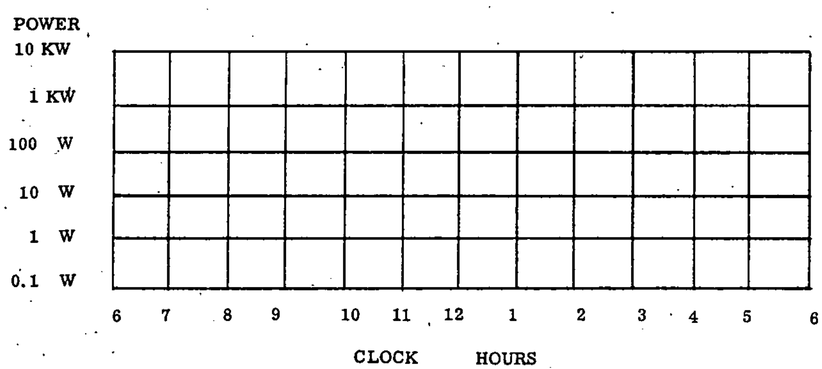
TYPE OF OPERATION: Reactor Experiment ☐ Performance Test ☐

DESCRIPTION OF EXPERIMENT OR OPERATION (Give title and brief description; include description of any auxiliary apparatus required; attach additional sheets if necessary)

REACTOR CONDITIONS:

ESTIMATED REACTIVITY EFFECT OF EXPERIMENT: ______

PROPOSED RUN HISTORY (Include setup and startup time, indicate periods, etc.)



COMMENTS OR REFERENCES

____
signed (experimentor or experimentors)

Isotope Production Involved in Experiment - ☐ Yes ☐ No

Material is to be irradiated:
Estimated amount of activity to be produced:
Amount of material to be inserted:
Proposed location of the material:
Expected Flux:

ACTION TAKEN: Date ____________________

In the judgement of the reviewer(s) this is a

☐ Routine operation involving no significant modification of the reactor.

☐ Developed experiment, as described in:

☐ New Experiment.

____
Signature(s) and Date

Comments or Suggestions

Disposition of Experiments

☐ Schedule for performance on ____________________

☐ Referred for further review to:

☐ Not allowed for following reasons:

____
Reactor Supervisor and Date

DATE ____________ page no. ____________

					ROD POSITIONS				INSTALLMENT READINGS									
Time	Run no.	Operator	Supervisor	Loading (Kg of U 235)	Shim 1	Shim 2	Shim 3	Regulating Rod	Power	Log CRM	Log -n Meter	Linear Level	Scaler	Period Meter	Fission Chamber Position	Temperature	Automatic Control On	Operation

Operating Log (Sample Form)

2.1.7 Professional Backgrounds of Staff

FRANCIS J. BRADLEY

Manhattan College, B.S. in E.E., 1949; University of Pittsburgh, M.S., 1949; National Research Fellow Health Physics, Oak Ridge National Laboratory, 1949-1950; Industrial Hygiene Department, Westinghouse Electric Co., 1950-1953; Radiation Superintendent, Ohio State University, 1953 - present. Experience in all aspects of radiation health physics; in charge Radiation Health Physics Conference, co-sponsored by OSU and AEC, June 1955.

WALTER E. CAREY

Kenyon College, A.B. in Physics, 1954; Ohio State University, M.S., 1957. Research Assistant, Ohio State University, 1954-1957. Resident Research Associate, Argonne National Laboratory, 1958 (duties were those of a reactor supervisor at the Argonaut Reactor). Research Associate, Ohio State University, 1959.

CHARLES E. DRYDEN

Drexel Institute of Technology, B.S. in Chemical Engineering, 1939; Princeton University, M.S., 1942; The Ohio State University, Ph.D., 1951. Research Chemist, Silmo Chemical Corporation, 1935-1941; Development Engineer, M. W. Kellogg Co., 1942-1943; Head, Process Development, Nopco Chemical Co., 1943-1948; Battelle Memorial Institute, Research Engineer, 1948-1951; Assistant Division Chief, 1951-1954; The Ohio State University, Assistant Professor, 1954-1955, Associate Professor, 1955 - present. On loan to General Electric Co. Atomic Power Study Group, 1955-1956; teaching nuclear engineering since 1955; Research and industrial experience includes vitamins, fats, oils, diffusional operations, absorption, extractive metallurgy, chemical pilot plant design, construction and operation, quarrying, heat transfer, chemical kinetics, rocket propellant, nuclear chemical engineering.

ALFRED B. GARRETT

B.S., Muskingum College, 1928; M.S., Ohio State University, 1931; Ph.D., 1932. Teacher, high school, Pennsylvania, 1928-1929; Chemistry, Ohio State, 1929-1932; Assistant Professor, Kent State College, 1932-1935; Instructor, Ohio State, 1935-1957, Assistant Professor, 1937-1940, Associate Professor, 1940-1944; Professor 1944 - present. Chemical Society Fellow, Institute of Chemistry; Ohio Academy, Chemistry; photovoltaic cells; ionic equilibria in solution; physical and organic chemistry; low-temperature studies of electrolytes; alkyl derivatives of boron hydrides; radio chemistry.

CHARLES D. JONES

Lehigh University, B.S. in M.E., 1947; University of Kentucky, M.S. in M.E., 1948; the Ohio State University, Ph.D., 1952; Instructor, University of Kentucky, 1948-1949; Instructor (part-time), Ohio State University, 1953-1955, Assistant Professor, 1955 - present; teaching nuclear reactor power plants since 1957. Research and teaching interests include heat transfer, thermodynamics, fluid dynamics, nuclear power plants.

KARL E. KRILL

B.S. in Ceramic Engineering, Missouri School of Mines and Metallurgy, 1941; M.S. (Geology), University of Colorado, 1948; Ph.D., Ohio State University, 1951. Chief Chemical Engineering and Materials Branch, Office of Ordnance Research, U.S. Army, 1952-1953; Assistant to the Director, Ohio State University Research Foundation, 1954-1957; Assistant to the Vice President, Ohio State University, 1957 - present, inorganic thermochemistry, research administration.

M. L. POOL

B.S., University of Chicago, 1924; Ph.D. (physics), 1927, Assistant Physics, Chicago 1925-1928; Instructor, Ohio State University, 1928-1932; Assistant Professor, 1932-1936; Associate Professor, 1936-1941; Professor, 1941 - present. Howald scholar, Michigan 1936-1937; member radiological safety section, Bikini 1946. Civilian with U. S. N.; Office Scientific Research and Development, U. S. A. F., 1944. Fellow Physical Society; Physics Teachers; Fellow Ohio Academy (Vice President, 1942-1957). Artificial radioactivity, cyclotrooneering, nuclear reactions cross-sections, nuclear spectroscopy.

2.2 PRE-OPERATIONAL TESTS

Each operating day, before the reactor is started up, the following procedures will be carried out by the reactor operator for that day.

a. A visual inspection will be made to ensure that all beam ports, shields, etc., are in place, and that the water in both pools is at a safe level.

b. The logbook, fuel inventory board, and experimental facilities board will be checked to acquaint the operator with the fuel loading and whatever experiments may be loaded into the reactor.

c. An instrument checkout will be conducted and the instrument checkout list will be filled out, dated, and signed by the operator.

d. The appropriate "Request for Reactor Operation" form(s) will be checked for necessary signatures and posted in a conspicuous spot in the control room.

2.3 STARTUP PROCEDURES

Information to be supplied at a later date.

2.4 NORMAL OPERATION

Information to be supplied at a later date.

2.5 HEALTH PHYSICS REQUIREMENTS

The health physics aspects of waste disposal, access to the reactor area, and fuel inventory are discussed in the following paragraphs.

2.5.1 Waste Disposal

There will be very little radioactive waste from the OSU educational reactor. Since no fuel processing is proposed, no fission product wastes are anticipated. Radioactive wastes such as are encountered in nonreactor nuclear laboratories will be low-level waste and will be handled by the campus Radiation Safety Officer in the same manner as that presently used. Some examples of such waste are contaminated hardware, filters from the pool water process system, ion exchange resins, and residues or spillage from isotopes produced in the reactor.

No water will be drained from the reactor pools until the Radiation Safety Officer has checked samples of the water for possible contamination. Upon receipt of information regarding the activity of the pool water, the water will be siphoned or pumped by means of an auxiliary system into the drain lines at a rate so as not to exceed the maximum permissible limit (0.1 mC per day).

2.5.2 Access Rules

The Reactor Building will be equipped with a conventional lock for which keys will be issued only to authorized staff members. In addition, the control room and supervisor's office will be equipped with separate locks. Keys to the control room will be issued only to licensed reactor operators or -- on a temporary basis -- to staff members who have a specific reason for access to the control room. The key to the supervisor's office will be issued only to the supervisor. The control console will also be equipped with a key lock. The key must be inserted and the lock disengaged before power is supplied to the magnets. The reactor supervisor will retain control of this key.

It will be the policy of the reactor installation to welcome all visitors. However, in the case of scheduling difficulties, instructional activities will take precedence over tours of the facility by visitors. A record of all visitors to the reactor will be kept. Care will be taken that visitors under the age of 18 will not be allowed in areas where the radiation field is greater than one-tenth (0.1) the permissible level for adults (2.5 mr/hr).

2.5.3 Fuel Inventory

Each fuel element of the OSU Education Reactor fuel inventory will be represented by a small card bearing the U-235 content (in grams) of that particular element. A fuel inventory board will be placed in the Reactor Supervisor's Office, in view of the control console, representing the possible fuel locations in the reactor core. Whenever a fuel element is placed in position in the reactor core, that fuel element's inventory card will be placed in a corresponding position on the inventory board. For each loading, the inventory board will be photographed and the photograph will be placed in the log book.

A similar board will be used to show fuel elements stored in the storage pits; and a third board will be used to show experiments or experimental apparatus loaded into the experimental facilities of the reactor.

Changes in the location of fuel elements will be made only after such changes are authorized by the reactor supervisor.

2.6 EMERGENCY PLAN

In case of emergency, an evacuation alarm (klaxon) will notify personnel to leave the building. Two doors are available for evacuation, both on the ground floor. One is at the north end and the other is at the south end of the building. Personnel leaving the building under emergency conditions will assemble at the rear (north) entrance to the Van de Graaff Building.

Staff members conducting experiments with students present will be responsible for these students. The reactor supervisor will be responsible for seeing that the building is cleared.

Names and telephone numbers of staff members to be called in case of emergency will be posted in various places in the reactor building as well as outside both doors to the building.

Procedures in case of emergency at a laboratory housing radioactive materials have been agreed upon by the OSU Radiation Safety Office, the OSU police, and the Columbus Fire Department. These procedures will apply to the reactor facility.

2.7 EXPERIMENTAL PROGRAM

The experimental program of the Ohio State University Educational Reactor will be conducted at essentially two levels, undergraduate and graduate. The undergraduate instructional program, consisting of regularly scheduled laboratory periods, will be conducted by faculty members from the various cooperating departments in the College of Engineering and the College of Arts and Sciences.

The graduate program will be carried out at a higher level of instruction and more freedom of reactor time will be given to the individual students. It is anticipated that much of the actual work in the original checkout of the characteristics of this reactor, after the reactor has been turned over to the Ohio State University by Lockheed Nuclear Products, will be done by graduate students from various departments.

Following is a list of proposed experiments to be performed with the Ohio State University Educational Reactor.

2.7.1 Reactor Safety Procedures

This experiment is to familiarize the students with reactor radiation safety instruments and procedures. The students will locate and diagram the location of the reactor experimental facilities, scram buttons, survey instrument storage area, fire fighting equipment and alarm, telephones, and area monitoring stations. In addition, they will use a beta-gamma survey meter, a neutron survey meter, and a Cutie-Pie survey meter to conduct a radiation survey of the reactor area. From data collected, they will plot isodose curves and calculate maximum permissible working times at various positions in the reactor area for various power levels of the reactor.

2.7.2 Neutron Detectors and Reactor Instruments

This experiment is to familiarize the student with neutron detecting devices used in reactor control and evaluation and with the electronic circuits necessary to apply them to these ends. The experiment is primarily directed toward the operation and characteristics of the BF_3 counter, the compensated ionization chamber, and associated electronic circuits with which the student has probably had no previous experience, namely the logarithmic amplifier and the period meter. These instruments will be set up independent from the actual control of the reactor.

2.7.3 Reactor Operation and Control

The student will study the control and safety mechanisms of the OSU Educational Reactor and become familiar with its startup and operating procedure. The student will study the operation of a control rod set up apart from the reactor. The student will learn the location and function of each instrument on the control console. The student will learn the actual startup and shutdown procedures used with the reactor.

2.7.4 Control Rod Calibration

The student will calibrate one or more control rods by period measurements and rod drop measurements.

2.7.5 Foil Activation and Flux Mapping

In this experiment, the student will be introduced to the technique of flux measurement by foil activation and will use this technique in mapping the flux distribution in various parts of the reactor.

2.7.6 Reactor Power Calibration

The student will measure the thermal flux distribution in the reactor core and from this will determine the reactor power.

2.7.7 Diffusion Length of Neutrons in Water

In this experiment, the student will measure the diffusion length of thermal neutrons in water, using the Bulk Shielding Facility.

2.7.8 Determination of Permi Age

In this experiment, the Permi age for thermal neutrons in graphite will be determined.

2.7.9 Albedo Determination

The albedo of graphite will be determined in this experiment.

2.7.10 Absorption Cross Section Measurements

In this experiment, an absorption cross section will be measured both by the method of activation analysis and the danger coefficient method; the results will be compared.

2.7.11 Absorption Cross Section Measurements by the Pile Oscillator

The student will use a pile oscillator to determine the neutron absorption cross section for certain materials.

2.7.12 Delayed Neutron Periods

In this experiment the student will measure the periods and relative percentages of the delayed neutrons from V 235 fission.

2.7.13 Reactor Shielding -- Reduction of Neutron Dose Rate

In this experiment, an investigation will be conducted into the relative stopping power, for neutrons, of various materials that might be used for reactor shielding. In the course of this investigation, certain physical characteristics, such as relaxation length, buildup factor, and effective removal cross section for the proposed shielding materials will be determined.

2.7.14 Reactor Shielding -- Reduction of Gamma Dose Rate

This experiment is very similar to the experiment described in 2.7.13, except that gamma radiation will be the type of radiation investigated. The student will determine the replacement length for several proposed shielding materials.

2.7.15 Reactor Shielding -- High Energy Capture Gamma-Rays

This experiment concerns itself with the measurement of the intensity and energy of those gamma-rays emitted when a neutron is captured by a nucleus composing the shielding material. A spectrum of the gamma radiation in the energy range from 2.5 to 10 Mev ensuing from the n, process is to be obtained.

2.7.16 Reactor Shielding -- Cloud Chamber Observations

This experiment is to provide a visual observation of the very complex problems arising in the shielding studies carried out quantitatively in experiments 2.7.13, 2.7.14, and 2.7.15.

3. REACTOR BUILDING AND SITE DESCRIPTION

This section contains a description of the location and construction of the reactor building and a discussion of the site, including utilities available, population distribution, climatology, and geological characteristics.

3.1 LOCATION

The OSU training reactor will be located on property owned by the Ohio State University west of the main campus. Figure 2 shows a scale plot plan of the site and some of the surrounding buildings. This site is to be used principally for research and development programs conducted by the University. Aerial photographs of the location are shown as Figures 3, 4, and 5. A map of the surrounding territory with 1/2, 1, 2, and 5-mile radius circles sketched in is presented as Figure 6; property owned by the University is indicated by the cross-hatched areas.

3.2 CONSTRUCTION OF BUILDING

The Reactor Building is a 48-foot by 62-foot steel framed structure with insulated metal wall panels and built-up roof. The ground floor is a concrete slab on grade. The second floor slabs are concrete supported on steel beams. Elevated platforms are checkered plate supported on steel beams. Interior partitions are plasterboard. Sanitary facilities and water service are provided. Floor drains provide for drainage. A forced warm air system heats the building. The electric service will be 120/240 volts, three-wire, single-phase; 240 volts, three-phase; and 120/208 volts, four-wire, solid neutral for lighting and power.

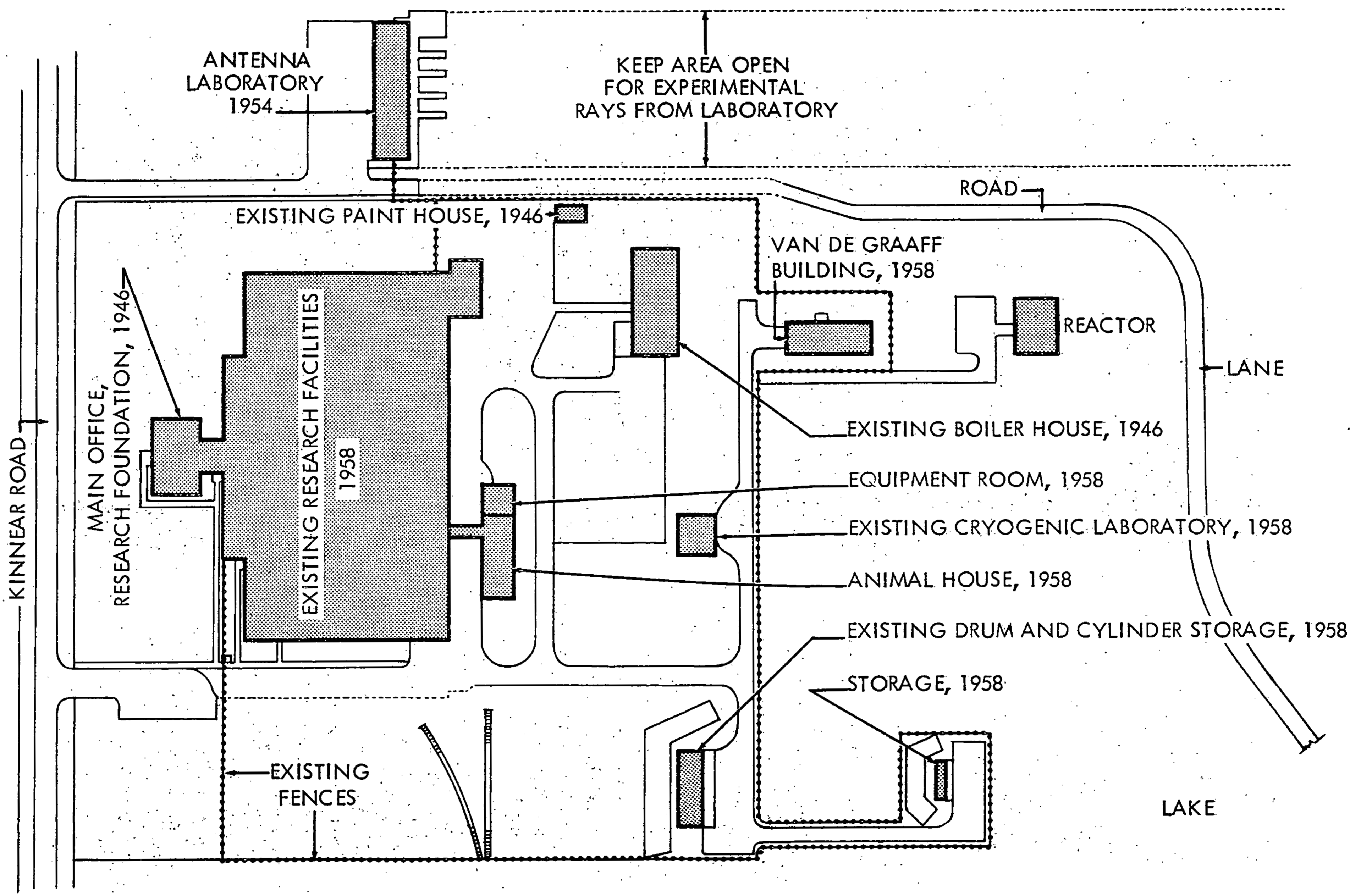


FIGURE 2 SCALE PLOT PLAN OF SITE

459 024

FIGURE 3

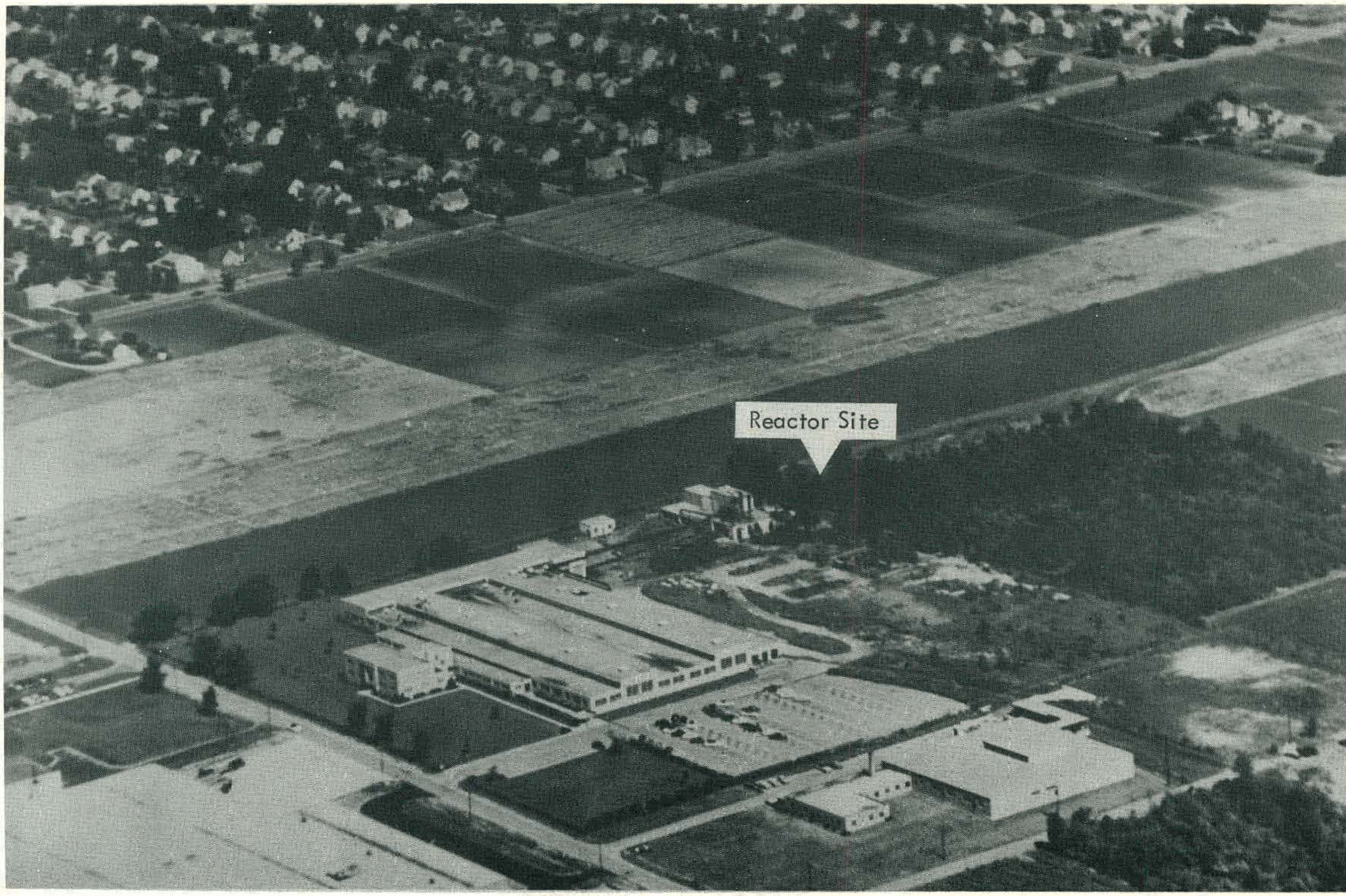


FIGURE 4

FIGURE 5

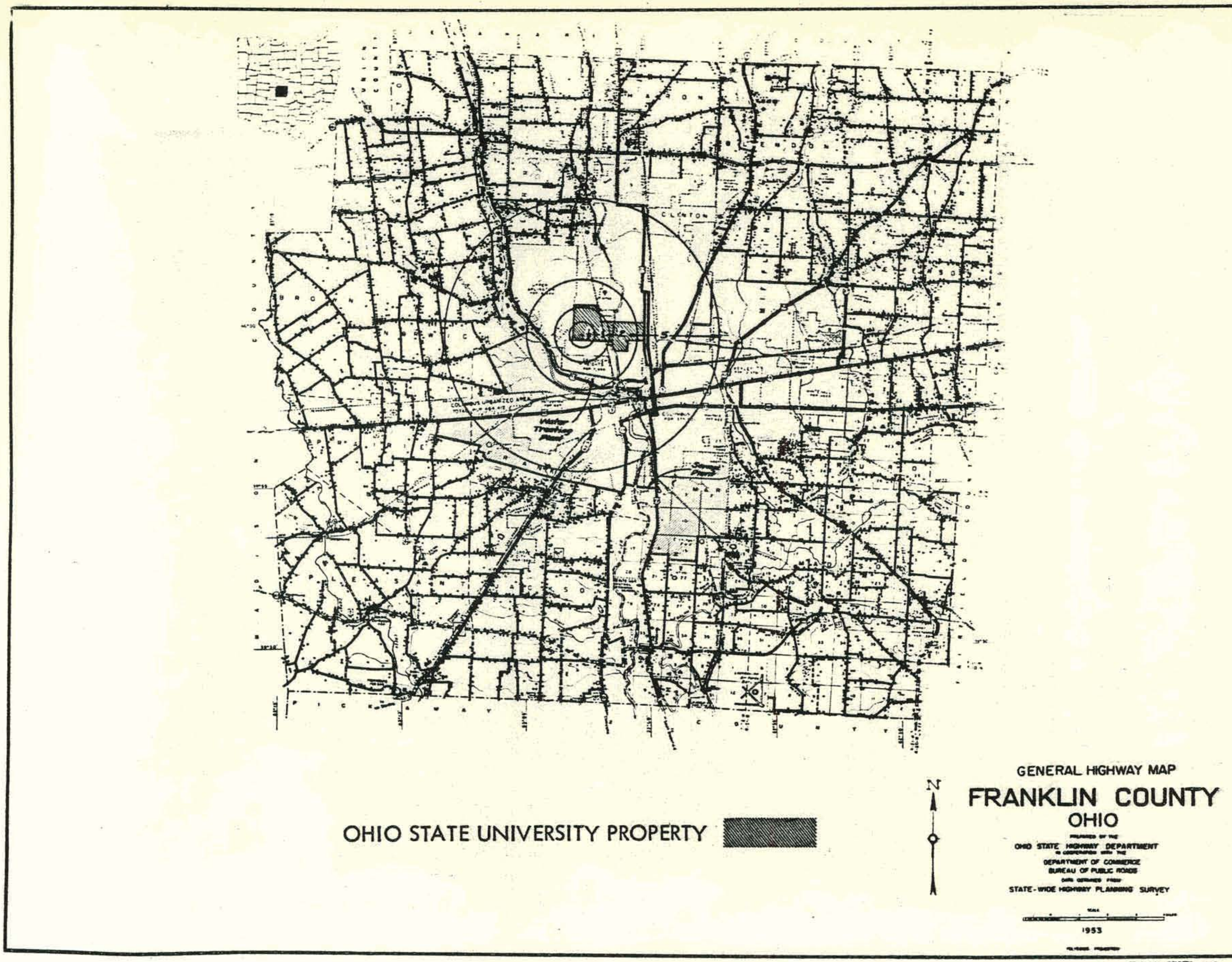


FIGURE 6 MAP OF SURROUNDING TERRITORY

3.3 UTILITIES

The reactor building area is serviced electrically by the Columbus and Southern Ohio Electric Company via 13,800 volt-ampere lines. The area has an electric failure probability of 0.2 failure per year. Water is supplied by the City of Columbus by piping distribution from its treating and pumping plant located 2.2 miles south by southeast of the reactor site. Fire protection for the area is maintained jointly by the Clinton Township Fire Department, located 1.2 miles away, and the City of Columbus Fire Department. Police protection is handled internally by the OSU Campus Police and externally by two township constables.

The reactor site is located about 780 feet above sea level. The site is drained by seepage to an underground water table which directs a flow 1.1 miles in an easterly direction to the Olentangy River, shown in Figure 6. Both rivers join at a point about 2.5 miles from the site and flow past the City of Columbus in a direction due south. The City of Columbus water-treating plant is located on the Scioto River 2.2 miles south by southeast of the reactor site and uses water collected along the Scioto basin which flows in an easterly direction towards the treating plant. These data show that contamination of the City of Columbus water supply by water drainage would be impossible. The next major town using water is Circleville, located 30 miles south of Columbus, as shown in Figure 7.,

The ground water level is approximately 45-50 feet below the surface of the site. Detailed information is contained in Table I.

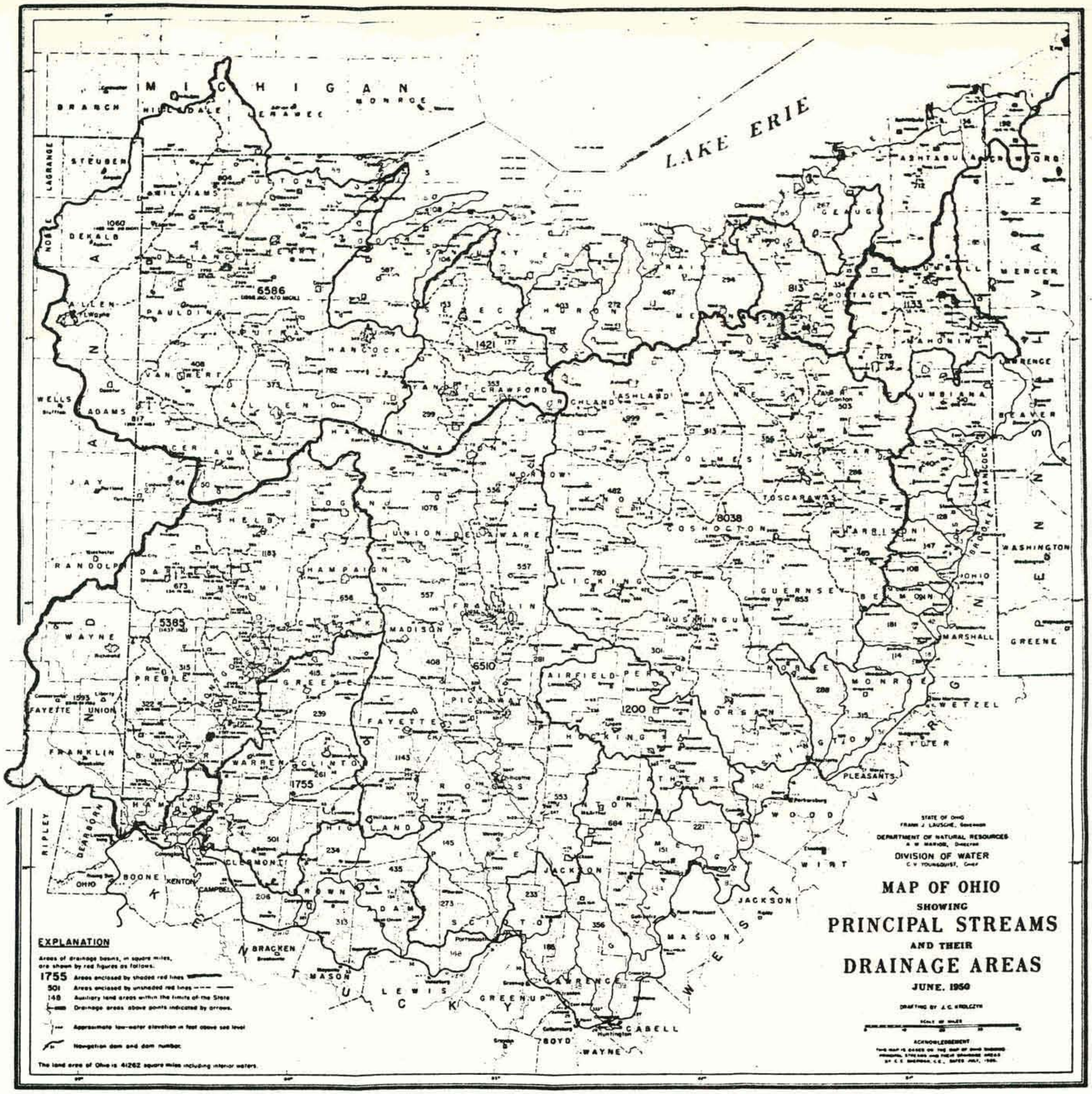

LAKE ERIE
MICHIGAN
STATE OF OHIO
FRANK J. LAUSCHE, Governor
DEPARTMENT OF NATURAL RESOURCES
DIVISION OF WATER
MAP OF OHIO
SHOWING
PRINCIPAL STREAMS
AND THEIR
DRAINAGE AREAS
JUNE, 1950
EXPLANATION
Areas of drainage basins, in square miles, are shown by red figures as follows.
1755 Areas enclosed by shaded red lines
501 Areas enclosed by unshaded red lines
148 Auxiliary land areas within the limits of the State
Drainage areas above points indicated by arrows.
Approximate low-water elevation in feet above sea level
Navigation dam and dam number
The land area of Ohio is 41262 square miles including interior waters.

FIGURE 7

TABLE I

GROUND WATER LEVEL DATA *
OBSERVATION WELL Fr-10

LOCATION

Ohio State University farm, 200 ft. west of Kenny Rd., 600 ft. south of Ackerman Rd., 6,000 ft. west of the Olentangy River, Clinton Tp., Dublin quad., T. 1 N., R. 18 W., lat. 40° 01' long. 83° 02', elev. 775 ft. above m.s.l.

DESCRIPTION

Unused drilled well, property of Ohio State University, dia. 4 in., depth 75 ft. Gravel aquefier. Measuring point at floor of instrument shelter, 4.0 ft. above land-surface datum. Observation by period measurement from March 1944 to December 1948 and by automatic recorder from December 1948 to the present.

REMARKS

Daily fluctuations of water level in this well show the influence of atmospheric pressure. Annual fluctuations are in response to recharge apparently at some distance from the well. The lack of recharge is clearly shown in the hydrograph for the years 1953 and 1954, in fact, the trend was almost continuously down during the period. Both the highest and lowest levels were below those of previous years of record.

Comparative water-level data for well Fr-10
(in feet below land surface)

Year	High	Date	Low	Date
1944	43.2	May 24	46.4	Nov. 23, 29 Dec. 6, 13
1945	42.5	July 11	46.5	Jan. 31
1946	42.0	Apr. 10	44.6	Jan. 3
1947	39.2	May 21, 28	44.0	Jan. 2
1948	38.6	Apr. 14, 21	44.0	Dec. 1
1949	38.7	May 1	44.2	Dec. 24, 25
1950	39.5	Apr. 24	44.1	Nov. 12
1951	37.8	Apr. 13	44.3	Nov. 5, 6
1952	38.4	Apr. 28	44.4	Dec. 19
1953	43.4	May 31	46.6	Nov. 28
1954	45.9	May 7, 16, 20	48.2	Oct. 7

* State of Ohio, Department of Natural Resources, Division of Water. Bulletin 29 (July, 1956).

The reactor site will be connected to the City of Columbus closed sewer system through the Ohio State University sewerage network, with proper monitoring devices installed at the site to prevent contamination of the sewer system with radioactivity. The sewage from the reactor site flows in a closed system to the City of Columbus Sewage Plant, located 5.9 miles south by southeast from the reactor site shown in Figure 9.

3.4 POPULATION DISTRIBUTION

A study of these maps and photographs shows that the territory within a 1/2-mile radius of the reactor site consists largely of University-owned farm land and some industrial buildings. There is no land available for new enterprises. Only a small portion of the circle on the west side is taken up by one or two-story residential dwellings. The residential and industrial population statistics are given in Tables II and III.

One point of interest is the concentration of about 80,000 people on five or six Saturday afternoons during the autumn quarter in the OSU Stadium located 1.3 miles from the reactor site.

TABLE II

RESIDENTIAL POPULATION DISTRIBUTION
(Reference Figure 9)

Region	Est. Residential Population 1950 Census	Area Sq. Miles	Density, Av. Population Sq. Mile
1/2-mile radius	200	0.785	260
1-mile radius	2600	3.14	820
2-mile radius	16000	12.53	1280

TABLE III

INDUSTRIAL POPULATION DISTRIBUTION

1/2-Mile Radius

Company	No. of Employees
OSU Building	250
Alexander Smith Rug	27
Simmons Mattress	360
Buckeye Telephone and Supply	36
Scott Viner	120
Fishel Construction	150
Allis-Chalmers	60
Total	1003

1/2-Mile - 1-Mile Radius

Company	No. of Employees
Suburban Motor Freight	150
National Heat Treating Service	25
Buel Gatterdam Co.	8
Kight Construction	7
Hutting Sash and Door Co.	40
Fifth Avenue Florist	8
Greybar Electric	30
Allis-Chalmers	60
McCune and Co.	8
Ohio Furnace Co.	16
Bituminous Coal Research	25
Ohio Plate Glass Co.	24
Ohio Machinery Co.	53
Livingston Seed	35
Lambert Jones Lumber	8
Lennox Furnace	460
J. I. Case	34
John Deere Plow	74
National Electric Coil	500
Total	1565
Av./Sq. Mile	620

TABLE IV

U. S. DEPARTMENT OF COMMERCE
WEATHER BUREAU

LOCAL CLIMATOLOGICAL DATA

WITH COMPARATIVE DATA

1957

COLUMBUS, OHIO

NARRATIVE CLIMATOLOGICAL SUMMARY

Columbus is located in the center of the state and in the drainage area of the Ohio River. The airport is located at the eastern boundary of the city approximately 7 miles from the center of the business district. The ground elevation of the airport is 815 feet above mean sea level.

Four nearly parallel streams run through or adjacent to the city. The Scioto River is the principal stream and flows from the northwest into the center of the city and then flows straight south toward the Ohio River. The Olentangy River runs almost due south and empties into the Scioto just west of the business district. Two minor streams run through portions of Columbus or skirt the eastern and southern fringes of the area. They are Alum Creek and Big Walnut Creek. Alum Creek empties into the Big Walnut southeast of the city and the Big Walnut empties into the Scioto a few miles downstream. The Scioto and Olentangy are gorge-like in character with very little flood plain and the two creeks have only a little more flood plain or bottom land.

The narrow valleys associated with the streams flowing through the city supply the only variation in the micro-climate of the area. The city proper shows the typical metropolitan effect with shrubs and flowers blossoming earlier than in the immediate surroundings and in retarding light frost on clear quiet nights. Many small areas to the southeast and to the north and northeast show marked effects of air drainage as evidenced by the frequent formation of shallow ground fog at daybreak during the summer and fall months and the higher frequency of frost in the spring and fall.

The average date of the last freezing temperatures in the spring, within the city proper is April 16th and the average date of the first freeze in the fall is October 31st, but in the immediate surroundings there is much variation; for example, at Valley Crossing located at the southeastern outskirts of the city, the average date of the last 32 degree temperature in the spring is May 2nd, while the average date of the first 32 degree temperature in the fall is October 12.

The records show a high frequency of calm or very low wind speeds during the late evening and early morning hours, from June through September. The rolling landscape is conducive to air drainage and from the Weather Bureau location at the airport the air drainage is toward the northwest with the wind direction indicated as southeast. Air drainage takes place at speeds generally 4 mph or less and frequently provides the only perceptible breeze during the night.

Columbus is located in the area of changeable weather. Air masses from central and northwest Canada frequently invade this region. The tropical Gulf masses often reach central Ohio during the summer and to a much lesser extent in the fall and winter. There are also occasional weather changes brought about by cool outbreaks from the Hudson Bay region of Canada, especially during the spring months. At infrequent intervals the general circulation will bring showers or snow to Columbus from the Atlantic. Although Columbus does not have a "wet" or "dry" season as such, the month of October has a higher frequency of light rainfall than any other month and comes closest to providing a normal dry period.

LATITUDE 40° 00' N
LONGITUDE 82° 53' W
ELEVATION (ground) 815 Feet

METEOROLOGICAL DATA FOR THE CURRENT YEAR

COLUMBUS, OHIO
PORT COLUMBUS AIRPORT
1957

Month	Temperature: Averages: Daily maximum	Daily minimum	Monthly	Extremes: Highest	Date	Lowest	Date	Degree days	Precipitation: Total	Greatest in 24 hrs.	Date	Snow, Sleet: Total	Greatest in 24 hrs.	Date	Relative humidity: 1:00 a.m., EST	7:00 a.m., EST	1:00 p.m., EST	7:00 p.m., EST	Wind: Average hourly speed	Prevailing direction	Fastest mile: Speed	Direction	Date	Percent of possible sunshine	Average sky cover sunrise to sunset	Number of days: Sunrise to sunset: Clear	Partly cloudy	Cloudy	Precipitation .01 inch or more	Snow, Sleet 1.0 inch or more	Thunderstorms	Heavy fog	Temperatures: Maximum: 90° and above	32° and below	Minimum: 32° and below	0° and below
Jan.	33.3	18.2	25.8	57	21	- 7	17	1211	1.94	0.56	9-10	10.2	4.8	6- 7	80	80	70	78	9.0	SSE	35	NW	23	28	8.4	3	4	24	18	2	0	3	0	14	29	2
Feb.	44.7	28.2	36.5	73	25	13	20	792	1.87	0.69	9	3.0	1.9	5	81	84	70	76	8.9	NNW	34	SW	14	38	6.1	2	7	19	13	1	0	3	0	0	22	0
Mar.	50.7	30.7	40.7	74	13	14	3	745	1.61	0.46	25	3.4	3.3	8	73	76	52	63	10.1	WNW	34	W	15	50	6.5	8	7	16	14	1	0	1	0	0	18	0
Apr.	64.4	43.7	54.1	87	27+	23	13+	379	5.70	2.37	3- 4	0.8	0.8	8	78	80	59	69	9.5	SSE	45	SW	5	50	7.3	4	8	18	14	0	7	0	0	0	9	0
May	74.6	50.8	62.7	86	13	32	6	138	5.41	1.46	21-22	T	T	4	80	77	48	60	9.2	NNW	48	W	18	70	6.3	9	10	12	12	0	11	1	0	0	1	0
June	83.6	61.0	72.3	97	17	49	3	9	5.20	1.72	28	0.0	0.0		86	83	55	62	7.7	SSW	34	SW	28	70	7.2	3	10	17	15	0	11	1	5	0	0	0
July	87.5	62.7	75.1	98	20	52	16	0	2.98	1.25	29	0.0	0.0		82	84	48	53	6.1	NNE	35	W	29	77	5.4	6	18	7	7	0	7	1	10	0	0	0
Aug.	86.3	60.2	73.3	95	31+	49	6	0	0.89	0.51	24-25	0.0	0.0		79	86	44	51	5.6	NNW	30	SW	3	77	5.5	5	18	8	6	0	4	0	10	0	0	0
Sept.	77.6	54.9	66.3	95	1	37	27	73	2.99	1.06	12-13	0.0	0.0		83	89	54	63	6.9	SSW	29	SW	2	62	6.4	7	9	14	10	0	3	1	2	0	0	0
Oct.	61.8	41.1	51.5	77	1	27	28	413	1.27	0.91	23-24	T	T	26-27	79	84	53	63	8.1	NNW	26	NW	24	50	6.5	9	4	18	9	0	0	2	0	0	4	0
Nov.	52.9	33.8	43.4	69	16	19	10	640	3.23	1.10	18-19	T	T	30+	78	80	61	68	9.4	SSW	47	SE	14	52	7.0	6	6	18	8	0	0	3	0	0	12	0
Dec.	45.4	28.1	36.8	61	23	8	11	367	4.68	1.09	17-18	8.5	8.1	3- 4	83	82	69	75	11.5	SSW	43	SE	20	44	7.0	6	8	17	14	1	0	0	0	2	22	0
Year	63.6	42.8	53.2	98	July 20	- 7	Jan. 17	5267	37.77	2.37	April 3- 4	25.9	8.1	Dec. 3- 4	80	82	57	65	8.5	NNW	48	W	May 18	56	6.8	68	109	188	140	5	43	16	27	16	117	2

NORMALS, MEANS, AND EXTREMES

Month	Temperature: Normal: Daily maximum	Daily minimum	Monthly	Extremes: Record highest	Year	Record lowest	Year	Normal degree days	Precipitation: Normal total	Maximum monthly	Year	Minimum monthly	Year	Maximum in 24 hrs.	Year	Snow, Sleet: Mean total	Maximum monthly	Year	Maximum in 24 hrs.	Year	Relative humidity: 1:00 a.m., EST	7:00 a.m., EST	1:00 p.m., EST	7:00 p.m., EST	Wind: Mean hourly speed	Prevailing direction	Fastest mile: Speed	Direction	Year	Pct. of possible sunshine	Mean sky cover sunrise to sunset	Mean number of days: Sunrise to sunset: Clear	Partly cloudy	Cloudy	Precipitation .01 inch or more	Snow, Sleet 1.0 inch or more	Thunderstorms	Heavy fog	Temperatures: Max.: 90° and above	32° and below	Min.: 32° and below	0° and below
(a)	(b)	(b)	(b)	79		79		(b)	(b)	79		79		79		73	73		73		18	18	18	18	52	8	55	55		63	68	79	79	79	79	73	79	72	79	79	79	79
J	37.8	21.6	29.7	74	1950	-20	1884	1094	2.94	10.71	1937	0.50	1944	2.92	1952	6.7	25.4	1918	11.9	1910	81	82	71	76	9.8	NW	63	NW	1938	36	7.1	5	8	18	14	2	*	1	0	10	25	1
F	39.6	22.8	31.2	73	1957	-20	1899	946	2.27	7.65	1893	0.43	1907	2.54	1891	5.2	29.2	1910	9.0	1914	78	81	66	73	9.9	NW	58	SW	1946	44	6.8	6	8	14	12	1	1	1	0	8	22	1
M	49.5	30.0	39.8	85	1945	- 1	1943	781	3.43	8.09	1913	0.28	1910	3.26	1913	3.1	25.2	1906	9.6	1906	77	77	58	66	10.3	WNW	68	SW	1942	49	6.6	6	9	16	14	1	2	1	0	3	17	*
A	61.4	38.9	50.2	90	1915	15	1881	444	3.44	7.08	1893	0.83	1889	3.33	1895	0.8	16.9	1886	6.1	1886	77	79	55	63	9.9	SSW	76	W	1920	54	6.2	8	9	13	13	*	4	*	*	*	5	0
M	72.5	49.1	60.8	96	1895	28	1947	180	3.97	9.59	1882	0.33	1939	2.50	1935	T	0.3	1923	0.3	1923	82	79	55	62	8.3	NW	56	NW	1925	68	5.7	9	12	10	12	0	8	*	1	0	*	0
J	82.2	59.2	70.7	102	1944	39	1913	31	4.33	8.52	1302	0.74	1950	3.54	1902	0.0	0.0		0.0		83	81	55	62	7.3	SSW	62	NW	1920	68	5.3	9	12	9	12	0	8	*	5	0	0	0
J	86.2	62.6	74.4	106	1936	44	1940	0	3.85	9.77	1896	0.49	1940	3.87	1947	0.0	0.0		0.0		84	83	52	60	6.6	S	84	NW	1916	71	4.9	11	13	7	11	0	8	*	8	0	0	0
A	84.0	60.8	72.4	103	1918	42	1887	8	3.21	7.16	1898	0.33	1924	3.71	1915	0.0	0.0		0.0		84	86	52	63	6.4	NNW	78	NW	1918	68	4.9	11	12	8	10	0	6	*	5	0	0	0
S	78.3	54.6	66.5	100	1939	31	1942	69	2.91	7.13	1890	0.42	1908	3.91	1938	0.0	0.0		0.0		83	86	50	65	7.0	SE	51	SW	1939	66	4.7	12	10	8	9	0	4	*	2	0	*	0
O	66.0	43.0	54.5	91	1939	17	1952	337	2.18	8.64	1881	0.10	1924	3.18	1910	0.1	3.0	1925	1.5	1925	81	86	53	67	7.9	S	60	S	1946	60	5.1	12	9	10	9	*	1	1	*	0	2	0
N	50.8	33.0	41.9	80	1950	- 5	1880	693	2.86	7.54	1897	0.18	1917	2.81	1881	1.7	14.3	1950	8.2	1950	78	82	62	71	9.4	SSW	61	N	1952	44	6.5	7	8	15	11	1	1	1	0	2	13	*
D	39.4	23.9	31.7	70	1956	-14	1951	1032	2.49	6.12	1923	0.46	1955	2.05	1921	4.3	14.5	1890	8.1	1957	80	82	69	76	9.3	S	58	W	1920	35	7.2	5	8	18	13	1	*	1	0	8	23	1
Year	62.3	41.6	52.0	106	July 1936	-20	Feb. 1899+	5615	37.88	10.71	Jan. 1937	0.10	Oct. 1924	3.91	Sept. 1938	21.9	29.2	Feb. 1910	11.9	Jan. 1910	81	82	58	67	8.5	S	84	NW	July 1916	55	5.8	101	118	146	140	6	43	6	21	31	107	3

REFERENCE NOTES

(a) Length of record, years.
(b) Normal values are based on the period 1921-1950, and are means adjusted to represent observations taken at the present standard location.
* Less than one half.
\- No record.
† Airport data.
‡ City Office data.
\+ Also on earlier dates, months, or years.
T Trace, an amount too small to measure.

Mean values at the end of the Average Temperature and Total Precipitation tables are long-term means based on the period of record beginning in 1879. Values have not been corrected for changes in instrument location listed in the Station Location Table.

Unless otherwise indicated, dimensional units used in this bulletin are: temperature in degrees F.; precipitation and snowfall in inches; wind movement in miles per hour; and relative humidity in percent.

Sky cover is expressed in a range of 0 for no clouds or obscuring phenomena to 10 for complete sky cover. The number of clear days is based on average cloudiness 0-3 tenths; partly cloudy days on 4-7 tenths and cloudy days on 8-10 tenths. Monthly degree day totals are the sums of the negative departures of average daily temperatures from 65° F. Sleet was included in snowfall totals beginning with July 1948.

Data for earlier years may be obtained by contacting the Weather Bureau Office for which this summary was issued.

Heavy fog in the Means and Extremes Table also includes data referred to at various times in the past as "Dense" or "Thick". The upper visibility limit for heavy fog is 1/4 mile.

Below zero temperatures are preceded by a minus sign.

Sale Price: 15 cents per copy. Checks and money orders should be made payable to the Superintendent of Documents. Remittances and correspondence regarding this summary should be sent to the Superintendent of Documents, Government Printing Office, Washington 25, D. C.

AVERAGE TEMPERATURE

Year	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	An'l.
1906	36.6	28.4	32.0	54.7	62.8	71.0	73.2	75.7	70.0	53.0	42.4	32.7	52.7
1907	33.5	27.2	46.8	43.0	55.7	67.2	73.9	71.0	66.7	50.0	40.2	34.6	50.8
1908	30.0	29.6	44.5	51.7	64.0	70.6	75.6	72.7	70.4	55.6	42.8	34.3	53.5
1909	32.8	36.2	38.1	50.1	59.8	71.8	72.0	73.4	64.1	49.9	49.6	25.8	52.0
1910	28.2	26.2	50.1	52.6	57.1	68.4	75.2	73.4	67.6	57.8	37.0	26.5	51.7
1911	33.5	35.4	38.3	49.0	68.6	72.8	75.7	73.9	68.8	54.1	38.2	37.4	53.8
1912	19.2	23.4	34.2	53.4	64.2	68.2	74.9	70.7	68.0	56.2	42.6	34.4	50.8
1913	36.6	27.2	40.8	50.4	61.7	71.8	76.5	75.4	65.4	54.4	45.8	35.5	53.5
1914	34.0	23.6	36.5	50.7	63.6	72.6	75.9	74.0	64.8	57.7	42.9	27.6	52.0
1915	27.8	36.0	34.0	56.3	58.2	66.0	73.0	68.2	68.0	56.7	44.4	31.0	51.8
1916	36.0	26.8	35.5	49.9	62.6	65.9	78.6	75.8	63.9	55.2	43.4	30.4	52.0
1917	29.8	25.8	40.4	49.6	55.2	67.8	73.6	72.4	63.8	47.5	40.3	21.3	49.0
1918	15.8	31.2	45.2	49.4	68.2	69.7	72.3	77.6	59.0	58.0	42.8	40.4	52.5
1919	33.1	32.6	41.8	50.8	59.2	75.6	76.5	70.6	68.1	60.4	40.6	26.0	52.9
1920	22.2	28.8	42.8	46.0	59.4	69.2	71.0	70.2	67.2	59.8	40.6	34.0	50.9
1921	33.8	34.6	49.8	55.0	62.9	74.4	78.7	71.7	70.6	54.2	43.2	33.8	55.2
1922	27.2	33.8	43.2	53.1	65.6	71.6	73.7	71.4	69.6	57.4	44.4	33.6	53.7
1923	33.8	26.4	37.8	49.6	59.8	71.9	74.8	71.8	60.9	52.2	42.4	42.0	52.4
1924	26.0	30.2	36.4	51.2	55.4	69.0	71.4	73.6	61.4	58.7	41.3	29.0	50.3
1925	28.8	37.6	42.8	55.2	57.4	74.6	73.0	73.5	71.8	47.0	40.8	29.7	52.7
1926	28.2	32.4	33.4	44.6	61.7	67.4	74.2	75.2	67.5	54.0	40.3	30.4	50.8
1927	29.9	37.8	44.0	50.9	60.4	65.6	73.7	67.7	70.3	59.0	47.1	32.8	53.3
1928	29.6	32.0	38.2	47.4	60.6	65.7	75.0	75.1	62.5	58.2	44.0	36.4	52.1
1929	27.7	26.2	47.4	54.4	59.7	68.7	74.2	69.5	66.3	53.1	39.9	33.1	51.7
1930	29.2	40.4	38.6	54.2	64.4	72.0	77.0	73.0	69.0	53.8	43.6	31.6	53.9
1931	33.0	36.8	37.3	51.7	59.8	72.5	78.6	72.8	71.4	58.2	51.4	41.1	55.4
1932	40.3	39.6	34.6	50.0	63.3	72.5	74.8	74.6	67.8	54.6	41.0	32.6	53.8
1933	38.6	32.4	39.4	51.9	64.7	76.1	76.5	73.4	70.4	53.6	40.6	35.6	54.4
1934	34.0	22.0	37.2	51.0	66.2	77.9	80.2	73.2	68.6	56.0	47.0	31.6	53.7
1935	31.2	32.4	47.8	49.1	58.0	69.2	78.2	74.4	66.0	55.0	43.9	26.2	52.6
1936	23.8	24.4	43.8	47.2	65.5	72.0	79.0	77.8	70.8	55.5	38.3	37.0	53.0
1937	37.8	32.3	36.8	51.1	62.5	71.6	75.1	76.6	65.0	52.6	40.2	31.0	52.7
1938	31.2	38.1	47.0	54.4	63.2	70.3	76.5	76.8	67.4	57.1	45.4	34.8	55.2
1939	35.3	33.9	42.4	48.4	66.2	74.4	74.6	74.9	72.4	57.2	42.1	35.8	54.8
1940	17.8	31.9	37.6	47.8	59.2	72.8	76.4	76.1	64.8	56.9	41.8	38.8	51.7
1941	31.6	28.6	34.8	58.0	65.2	73.0	77.4	73.9	70.6	59.7	45.1	38.2	54.7
1942	29.4	28.6	43.6	55.5	64.2	73.3	76.8	73.2	66.4	57.4	44.8	29.1	53.5
1943	32.0	33.0	38.1	47.0	63.4	76.6	76.2	74.6	64.6	54.3	40.6	30.6	52.6
1944	33.9	35.2	37.6	50.8	69.4	75.5	77.3	76.0	67.0	55.3	43.8	27.8	54.1
1945	23.7	32.8	51.4	54.3	58.0	69.9	73.5	73.8	70.0	54.0	44.2	26.6	52.7
1946	32.2	34.2	52.6	52.2	60.2	70.6	75.0	69.1	67.6	59.6	48.0	36.8	54.8
1947	35.8	24.0	34.6	52.0	59.4	70.4	71.0	78.4	68.0	64.2	40.5	33.4	52.7
1948	22.2	32.6	44.2	56.0	61.0	72.4	77.0	74.4	68.9	52.6	48.0	36.5	53.8
1949	37.4	38.0	42.0	50.6	64.6	76.0	79.3	75.4	62.3	60.9	43.8	37.1	55.6
1950	40.4	33.6	37.5	46.4	64.4	70.4	73.9	72.1	66.0	59.9	38.1	26.7	52.5
1951	32.4	32.3	41.0	50.1	64.8	70.8	74.2	71.8	64.4	56.8	36.7	31.8	52.3
1952	35.1	34.5	39.9	51.9	60.8	74.5	76.9	73.2	65.7	48.6	43.3	35.3	53.3
1953	35.3	35.7	42.5	48.2	64.9	73.7	75.5	73.3	66.2	57.2	43.6	34.0	54.2
1954	31.8	39.0	37.8	57.4	57.7	72.9	74.4	72.3	69.0	56.1	42.5	32.3	53.6
1955	28.5	32.1	41.4	56.8	63.8	66.9	79.0	77.2	69.2	55.2	39.9	30.1	53.3
1956	27.7	35.0	39.5	48.5	61.3	71.1	73.8	73.3	63.4	59.4	43.1	40.8	53.1
1957	25.8	36.5	40.7	54.1	62.7	72.3	75.1	73.3	66.3	51.5	43.4	36.6	53.2
RECORD MEAN TEMP	30.0	31.4	40.3	51.3	62.2	71.3	76.4	74.2	67.9	56.1	43.0	33.2	52.7
MAX	37.2	39.1	49.0	61.0	72.3	81.3	86.7	84.3	78.3	66.0	50.9	40.1	61.8
MIN	22.7	23.6	31.5	41.5	52.0	61.2	66.0	64.0	57.4	46.1	35.0	26.2	43.6

TOTAL PRECIPITATION

Year	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	Annual
1906	1.98	1.08	4.59	1.16	2.47	1.44	5.27	6.15	1.59	2.07	2.57	3.33	33.70
1907	5.73	.43	5.21	3.27	3.35	3.39	6.07	2.74	2.27	1.59	1.68	1.85	37.58
1908	1.40	3.66	6.03	2.75	4.04	2.13	3.74	2.34	.42	1.20	.84	1.59	30.14
1909	2.52	4.97	2.68	3.20	4.65	3.88	3.34	2.53	1.81	2.77	1.66	2.58	38.59
1910	5.11	5.05	.28	2.52	4.10	2.93	2.40	.42	3.66	5.22	.79	2.31	34.79
1911	4.48	1.71	2.36	4.37	1.15	4.04	3.29	3.62	5.98	5.21	2.71	4.53	43.43
1912	1.58	1.53	4.56	4.20	2.65	1.48	3.50	2.25	2.83	1.71	1.01	2.34	29.64
1913	6.63	2.09	8.09	3.91	2.60	1.56	2.88	2.10	3.28	2.05	4.56	1.13	40.88
1914	2.21	3.70	2.46	2.48	1.28	2.03	1.64	4.78	1.26	4.44	1.99	2.91	31.18
1915	3.30	1.52	1.19	.95	2.57	5.06	6.85	7.01	4.43	.94	1.97	4.15	39.94
1916	5.02	1.47	4.88	2.33	4.81	3.49	.66	3.22	1.54	1.84	1.58	3.59	34.43
1917	3.74	1.09	3.59	3.15	2.80	6.29	4.09	3.10	.55	3.05	.18	1.31	32.94
1918	3.51	2.55	1.85	2.80	4.30	1.25	2.50	4.42	3.19	2.09	1.24	3.23	32.93
1919	1.28	1.27	4.58	2.26	5.19	1.76	4.93	3.71	1.15	5.33	4.23	2.26	37.95
1920	2.64	1.12	3.32	4.51	2.00	3.79	5.18	4.09	2.29	1.61	3.45	1.60	35.60
1921	2.19	1.90	7.66	4.11	2.67	2.06	2.20	4.55	2.05	1.86	5.50	3.74	40.49
1922	1.80	1.58	4.54	3.05	5.26	3.14	2.29	1.94	2.68	1.56	1.57	2.69	32.10
1923	3.46	2.51	3.04	2.12	2.42	5.10	4.09	4.97	3.39	1.66	2.30	6.12	41.18
1924	3.95	1.73	4.28	3.12	3.43	5.37	2.98	.33	3.77	.10	1.86	2.35	33.27
1925	1.48	1.77	2.25	1.68	2.33	1.67	3.27	2.37	2.21	3.71	3.40	.73	26.87
1926	2.52	2.93	2.16	2.56	1.42	.96	4.47	6.13	5.77	4.50	2.22	3.05	38.69
1927	3.93	1.62	3.97	3.80	6.48	3.63	3.86	1.91	2.06	1.18	6.19	3.55	42.18
1928	1.44	2.28	2.79	2.92	1.51	6.94	6.27	1.05	.80	2.85	2.32	2.07	33.24
1929	3.54	3.36	1.76	3.71	4.55	4.76	2.91	3.78	2.28	4.09	3.72	3.81	42.27
1930	4.66	3.45	2.13	1.28	1.04	1.25	1.28	2.45	1.05	.70	1.31	.80	21.60
1931	.88	1.54	1.34	4.38	2.96	2.30	4.66	4.80	3.34	2.41	3.64	3.29	35.54
1932	4.38	.88	2.20	1.75	.79	4.38	4.31	.71	1.52	3.99	2.07	3.05	30.03
1933	1.36	1.37	5.44	3.15	6.95	1.71	.85	2.07	4.45	1.20	1.02	2.45	32.02
1934	1.16	1.18	2.20	1.24	.60	2.74	2.94	4.13	3.36	.26	.92	1.30	22.03
1935	1.50	1.39	2.81	1.88	4.71	2.40	4.83	6.61	2.80	1.26	2.84	2.32	35.35
1936	1.26	2.87	3.10	3.17	2.34	2.23	1.80	4.94	3.29	3.41	2.95	1.79	33.15
1937	10.71	.87	1.56	3.16	3.28	6.71	3.91	2.89	2.45	2.41	1.02	3.00	41.97
1938	1.11	2.73	4.32	3.07	4.82	1.98	5.22	3.32	5.40	.70	1.94	1.04	35.65
1939	3.17	3.92	2.41	4.62	.33	6.06	2.55	.49	1.39	3.17	1.06	1.20	30.37
1940	1.44	2.98	2.65	6.31	4.65	3.79	.49	2.27	1.55	1.38	3.64	2.72	33.87
1941	2.05	.64	.61	2.32	2.47	6.20	2.65	2.96	1.82	4.66	1.31	1.78	29.47
1942	1.24	2.11	2.59	2.24	3.41	2.95	2.57	2.97	2.83	1.06	3.08	3.28	30.33
1943	2.03	1.48	4.57	2.69	4.33	4.19	7.05	1.78	.95	.95	1.15	.91	32.08
1944	.50	2.05	5.41	3.79	4.51	2.30	1.99	3.68	1.35	.25	1.54	3.53	30.90
1945	.93	3.14	8.08	5.52	3.86	3.24	4.84	1.87	2.71	2.11	3.01	1.66	40.97
1946	.78	3.21	3.31	1.72	4.31	7.25	3.95	3.58	1.02	2.20	2.76	1.98	36.07
1947	4.73	.51	1.07	5.30	4.94	4.14	6.34	3.64	3.67	1.59	1.76	1.44	39.13
1948	2.16	2.75	3.86	4.39	3.34	3.94	5.25	2.31	2.25	1.98	3.99	2.24	38.46
1949	6.74	2.53	3.03	2.84	2.05	4.47	3.24	2.37	2.86	1.31	1.21	2.37	35.02
1950	6.86	3.22	1.31	3.80	1.15	1.95	2.96	3.52	2.31	1.58	4.92	3.04	38.62
1951	3.80	3.05	3.92	2.94	3.71	4.43	.99	.58	3.00	1.36	4.42	5.07	37.27
1952	5.99	2.31	3.95	4.07	4.33	2.75	3.01	1.87	2.49	.81	1.09	2.58	35.25
1953	4.83	1.41	1.49	2.32	3.05	1.79	4.89	2.62	2.12	.59	1.43	1.88	28.42
1954	2.92	1.78	2.82	3.77	2.56	3.27	3.26	5.18	.74	5.24	1.08	1.69	34.31
1955	1.44	3.73	3.63	2.09	3.56	2.63	4.77	2.38	2.69	1.95	3.38	0.46	32.71
1956	2.13	4.33	3.49	4.06	4.85	3.22	4.92	2.23	1.37	1.15	0.97	3.21	35.93
1957	1.94	1.87	1.61	5.70	5.41	5.20	2.98	0.89	2.99	1.27	3.23	4.68	37.77
RECORD MEAN	3.01	2.55	3.32	3.07	3.50	3.46	3.61	3.13	2.51	2.35	2.03	2.63	35.77

MONTHLY AND SEASONAL DEGREE DAYS

Season	July	Aug.	Sept.	Oct.	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May	June	Total
1905-1906	0	0	40	354	732	954	880	1024	1024	318	170	15	5511
1906-1907	0	0	22	376	679	1001	977	1058	580	664	306	56	5719
1907-1908	2	4	64	474	746	941	1086	1026	638	410	168	26	5585
1908-1909	0	0	44	300	664	952	998	806	834	453	200	4	5255
1909-1910	7	0	112	468	462	1213	1138	1085	484	378	266	89	5702
1910-1911	0	1	35	272	840	1193	975	830	828	481	110	2	5570
1911-1912	5	8	27	338	802	856	1420	1205	954	359	102	28	6104
1912-1913	0	17	81	294	672	948	872	1060	746	448	154	47	5341
1913-1914	0	2	103	358	576	915	960	1162	882	453	145	19	5575
1914-1915	1	0	101	248	663	1100	1150	812	960	318	216	26	5655
1915-1916	2	35	70	265	618	1052	898	1108	918	460	144	46	5616
1916-1917	0	2	124	330	646	1074	1090	1096	762	481	341	38	5984
1917-1918	1	8	78	546	741	1354	1523	948	614	470	64	18	6365
1918-1919	10	0	202	226	666	762	989	908	721	432	230	0	5146
1919-1920	0	10	38	198	732	1208	1326	1052	689	572	209	46	6080
1920-1921	5	14	44	203	732	960	968	849	478	320	162	14	4749
1921-1922	0	4	16	334	652	966	1172	874	678	370	56	4	5126
1922-1923	4	9	38	286	618	972	969	1080	842	462	208	35	5523
1923-1924	0	19	71	400	678	714	1209	1008	887	412	312	33	5743
1924-1925	14	4	146	208	710	1114	1121	768	690	320	270	4	5369
1925-1926	2	0	28	557	726	1094	1140	914	981	610	161	40	6253
1926-1927	4	0	64	378	741	1074	1088	761	652	432	193	76	5463
1927-1928	3	24	50	220	538	998	1097	958	829	536	180	63	5496
1928-1929	0	0	159	259	630	886	1158	1086	555	343	234	44	5354
1929-1930	4	18	96	368	752	988	1112	688	818	361	124	24	5353
1930-1931	3	4	38	382	644	1034	991	790	858	407	207	27	5385
1931-1932	0	2	37	241	414	741	766	736	944	454	128	6	4469
1932-1933	0	2	58	324	721	1007	820	914	793	401	94	27	5161
1933-1934	0	0	22	357	734	912	960	1202	864	420	90	0	5561
1934-1935	0	8	36	282	539	1034	1046	914	532	480	236	38	5145
1935-1936	0	20	61	320	632	1201	1278	1178	656	545	104	6	6001
1936-1937	0	1	53	308	800	870	842	916	882	416	150	6	5244
1937-1938	2	0	110	408	744	1054	1046	754	562	338	136	2	5156
1938-1939	0	0	57	258	592	934	920	871	704	506	109	9	4960
1939-1940	2	0	21	288	686	904	1464	960	850	514	226	15	5930
1940-1941	8	6	104	263	694	812	1036	1018	936	247	119	7	5250
1941-1942	0	3	26	214	594	833	1103	1018	661	325	138	17	4932
1942-1943	0	8	133	246	607	1109	1026	897	833	540	126	8	5533
1943-1944	1	4	111	339	734	1066	962	864	854	428	49	10	5422
1944-1945	0	6	53	316	637	1155	1283	903	432	332	241	56	5414
1945-1946	7	3	31	347	623	1192	1016	862	390	390	169	28	5058
1946-1947	0	23	35	187	516	881	904	1147	943	392	218	23	5269
1947-1948	7	0	112	88	733	983	1324	939	648	306	155	3	5298
1948-1949	0	2	45	380	510	885	855	758	711	429	111	7	4693
1949-1950	0	1	130	187	636	867	761	882	894	555	94	19	5026
1950-1951	0	3	68	182	806	1176	1003	909	734	464	99	13	5457
1951-1952	0	8	94	297	843	1023	918	878	771	398	156	7	5393
1952-1953	0	5	67	505	645	913	915	817	691	500	79	9	5146
1953-1954	0	3	75	254	638	955	1024	724	836	261	258	23	5051
1954-1955	0	4	33	346	668	1008	1127	915	728	243	105	40	5217
1955-1956	0	0	29	302	748	1075	1152	845	786	495	186	42	5660
1956-1957	0	8	114	173	651	745	1211	792	745	379	138	9	4965
1957-1958	0	0	73	413	640	867							

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MONTHLY AND SEASONAL SNOWFALL

Season	July	Aug.	Sept.	Oct.	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May	June	Total
1905-1906	0	0	0	0	T	1.1	5.4	5.0	25.3	0	T	0	36.8
1906-1907	0	0	0	0.2	0.3	6.5	5.2	3.5	2.5	1.5	0	0	19.7
1907-1908	0	0	0	T	T	2.5	3.0	4.3	1.6	2.4	0	0	13.8
1908-1909	0	0	0	0	T	0.2	13.4	1.3	0.9	0.4	T	T	16.2
1909-1910	0	0	0	T	0.1	11.0	24.3	29.2	T	3.2	0	0	67.8
1910-1911	0	0	0	0.5	0.6	7.3	13.7	9.4	3.4	2.5	T	0	37.4
1911-1912	0	0	0	0	3.0	4.4	8.9	10.1	7.4	0.8	0	0	34.6
1912-1913	0	0	0	0	T	1.3	5.9	6.6	1.9	T	0	0	15.7
1913-1914	0	0	0	T	7.5	0.5	7.6	19.6	4.2	0.5	0	0	39.9
1914-1915	0	0	0	0	T	6.0	17.8	1.2	0.6	T	0	0	25.6
1915-1916	0	0	0	0	0.7	5.1	1.8	7.1	10.8	2.0	0	0	27.5
1916-1917	0	0	0	0	0.3	13.5	15.4	4.0	4.2	0.3	0	0	37.7
1917-1918	0	0	0	0.6	0.6	14.5	25.4	1.1	0.2	1.5	0	0	43.9
1918-1919	0	0	0	0	T	0.8	0.4	1.8	0.2	T	0	0	3.2
1919-1920	0	0	0	0	T	5.4	12.5	5.4	1.8	5.4	0	0	30.5
1920-1921	0	0	0	0	8.2	3.3	1.3	4.2	T	0.3	0	0	17.3
1921-1922	0	0	0	T	1.8	4.1	3.4	2.5	3.1	T	0	0	14.9
1922-1923	0	0	0	0	1.0	2.1	0.9	9.0	3.5	T	0.3	0	16.8
1923-1924	0	0	0	0	T	0.3	2.8	5.3	2.4	1.0	0	0	11.8
1924-1925	0	0	0	0	4.0	2.0	11.3	0.4	1.1	T	0	0	18.8
1925-1926	0	0	0	3.0	T	1.2	11.9	6.1	6.5	1.6	0	0	30.3
1926-1927	0	0	0	T	1.4	3.4	3.7	4.5	T	T	0	0	13.0
1927-1928	0	0	0	0	2.4	0.5	1.7	3.7	1.8	0.1	0	0	10.2
1928-1929	0	0	0	0	T	T	7.7	10.3	0.4	T	T	0	18.4
1929-1930	0	0	0	T	T	10.7	4.9	0.7	4.1	T	0	0	20.4
1930-1931	0	0	0	T	1.6	2.0	1.4	1.9	5.3	T	0	0	12.2
1931-1932	0	0	0	0	3.1	T	T	0.4	1.7	T	0	0	5.2
1932-1933	0	0	0	0	T	6.8	0.2	2.7	3.3	T	0	0	13.0
1933-1934	0	0	0	0	2.7	7.8	2.1	13.2	6.8	0.8	0	0	33.4
1934-1935	0	0	0	0	T	3.0	2.6	1.7	T	0.5	0	0	7.8
1935-1936	0	0	0	0	1.1	9.8	12.1	9.6	2.8	0.2	0	0	35.6
1936-1937	0	0	0	0	3.6	3.8	4.2	1.3	2.3	T	0	0	15.2
1937-1938	0	0	0	T	2.4	1.9	2.9	3.4	2.1	T	0	0	12.7
1938-1939	0	0	0	0	1.7	0.3	6.9	9.2	T	0.2	0	0	18.3
1939-1940	0	0	0	T	T	3.1	8.5	10.4	4.3	0.4	T	0	26.7
1940-1941	0	0	0	0	0.3	0.1	6.4	3.2	0.6	0	0	0	10.6
1941-1942	0	0	0	0	0.3	T	2.8	0.6	1.2	T	0	0	4.9
1942-1943	0	0	0	T	2.0	8.0	5.5	1.1	4.4	0.8	0	0	21.8
1943-1944	0	0	0	T	1.2	0.3	T	8.8	3.2	0.1	0	0	13.6
1944-1945	0	0	0	0	T	4.2	6.0	0.6	T	T	0	0	10.8
1945-1946	0	0	0	0	0.4	7.7	1.1	0.5	T	0.2	0	0	9.9
1946-1947	0	0	0	0	0	2.4	1.7	5.4	3.3	0	T	0	12.8
1947-1948	0	0	0	0	0.4	6.2	11.2	6.7	0.8	0	0	0	25.3
1948-1949	0	0	0	0	T	3.5	4.5	0.1	1.7	T	0	0	9.8
1949-1950	0	0	0	0	T	0.3	0.1	2.0	0.7	0.2	T	0	3.3
1950-1951	0	0	0	0	14.3	7.1	3.8	4.6	3.4	T	0	0	33.2
1951-1952	0	0	0	T	1.9	13.7	2.7	4.1	1.3	T	0	0	23.7
1952-1953	0	0	0	T	2.3	3.6	6.7	0.8	3.9	4.5	0	0	21.8
1953-1954	0	0	0	0	3.0	5.3	4.5	5.7	5.5	0.9	T	0	24.9
1954-1955	0	0	0	0.2	1.9	1.5	7.1	7.7	3.9	T	0	0	22.3
1955-1956	0	0	0	T	5.1	2.9	11.4	0.8	7.6	0.6	0	0	28.4
1956-1957	0	0	0	0	2.6	5.0	10.2	3.0	3.4	0.8	T	0	25.2
1957-1958	0	0	0	T	T	8.5							

The horizontal lines drawn on the Average Temperature, Total Precipitation, Monthly and Seasonal Degree Days, and Monthly and Seasonal Snowfall tables separate the data according to station location (see Station Location table).

STATION LOCATION

Location	Occupied from	Occupied to	Airline distance and direction from previous location	Latitude	Longitude	Elevation above: Sea level: Ground	Elevation above: Sea level: Actual barometer elevation (H_s)	Elevation above: Ground: Wind instruments	Extreme thermometers	Psychrometer	Telepsychrometer	Tipping bucket rain gage	Weighing rain gage	8" rain gage	Sunshine Switch		
						CITY OFFICE											
Irving House, between 3d & 4th on E. Broad Street.	7- 1-78	7-15-78		39° 58' N	83° 00' W				25								Location was temporary. No wind instr., rain gage on roof. Thermometers in N window.
Huntington Bank Bldg., 4th Floor, Broad & High Streets	7-15-78	5- 1-89	1 1/2 Blk W	39° 58' N	83° 00' W	755	805	84	52					70			On 10-12-84 thermometers moved to 9 ft. above roof & 78 ft. above ground.
Board of Trade Bldg., 40 E. Broad Street	5- 1-89	2- 1-93	1/2 blk E	39° 58' N	83° 00' W		837	100	102	102		96		96			On 9-16-91 anem. was raised to 15 ft. above roof - 108 ft. above ground.
Wheeler Bldg., 5 W. Broad Street	2- 1-93	11- 1-94	1/2 blk W	39° 58' N	83° 00' W	759	868	132	126	126		120		120			
Eberly Bldg., 215 So. High Street	11- 1-94	6- 1-02	1/4 mi. S	39° 58' N	83° 00' W	770	824	93	87	87		81		81			On 10-7-96 anemometer raised to 100 ft. above ground.
New Haydon Bldg., 16 E. Broad Street	6- 1-02	7- 1-30	1/4 mi. N	39° 58' N	83° 00' W	759	918	190	173	173		171		171			Erection of taller bldg. to SW caused interference with wind record 4-3-06 to 9-22-06 at which time wind instruments were moved to the Capitol Trust Bldg. & mounted 222 ft. above ground.
8 East Broad Bldg.	7- 1-30	2- 1-35	1st bldg. W	39° 58' N	83° 00' W	759	947	230	216	216		209		209			Move made because of erection of taller bldg. in 1927 1 blk SW, interfering with wind records.
New Post Office Bldg., 85 Marconi Blvd.	2- 1-35	Present	1/4 mi. NW	39° 58' N	83° 00' W	724	782	110	#90	90		*89	*89	*89			*Tipping bucket gage moved to airport 5-11-1951 and replaced by weighing gage at same roof location. On 7-10-1951 this gage was moved to ground location 100 ft. S of Post Office Bldg. On 12-1-1954 it was returned to the roof location. #On 7-10-1951 an extra set of thermometers was installed in CR shelter in a ground location 100 ft. south of PO Bldg. and on 12-1-1954 this installation was abandoned.
						AIRPORT STATION											
Port Columbus Airport Administration Bldg., 2d Floor, 7.3 miles ENE of Post Office	5- 7-30	Present		39°59'48"	82°52'48"	815	833	46	32	32		30		28	31		Expansion of administration building required placing instrumental equipment on roof 6-21-1955. The roof is covered with gravel set in tar which results in slightly different temperature extremes than were usually observed when a grass plot was used as the exposure.

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3.5 CLIMATOLOGY INFORMATION

The site area is in a temperate, continental climate zone. Local climatology data for the year 1957 with comparative data from previous years are given in Table IV.

Climatology highlights

Temperature	The mean daily temperature for June, July, and August is 73.3°F, and for December, January, and February is 31.2°F.
Winds	Figures 11 through 14 show surface wind roses taken at Port Columbus, approximately 7 miles due east of the reactor site. Winds are predominately from the west and southwest. The data for these plots were derived from "Climatology of the United States No. 30-33, Columbus, Ohio," Superintendent of Documents, U. S. Government Printing Office, Washington 25, D. C.
Tornados	There have been 4 tornados recorded since 1931.
Precipitation	The yearly average moisture precipitation is 2.96 inches per month. The season between April and August averages 3.5 inches per month. The average winter yields 22 inches of snow with an average of 2-3 inches per storm.

Low-Level Temperature Inversions

Although actual measurements of the height of low-level temperature inversions are not available for the Columbus area there are certain meteorological factors which may be used to verify the presence of such inversions. First, the normal diurnal variation in temperature is a key to the presence of inversions. This variation from the highest daily temperature to the lowest averages 20 to 25 degrees daily throughout the year. At the same time, the diurnal variation of the temperature in the layer above surface friction is very small, except in those cases where a front passes over a given area, resulting in a change of air mass. Frontal passages take place on an average of every 2 to 3 days during the winter months and 3 to 4 days during the summer months, hence a daily range of 68° to 45°, for example, indicates the presence of a temperature inversion during the time of lowest temperature.

A second criterion in the Columbus area is the high frequency of calm or very low wind speeds during the night and early morning hours from May through September. Very light wind speeds and periods of calm indicate a lack of mixing with the free moving air above the layer of surface friction. The above factors, combined with the irregular topographic features around Columbus, result in the presence of one or more temperature inversions in the lowest layers of the atmosphere. The irregular topographic features lead to air drainage, that is, the cooler pockets of air forming close to the ground tend to drain toward the lower contours of the land.

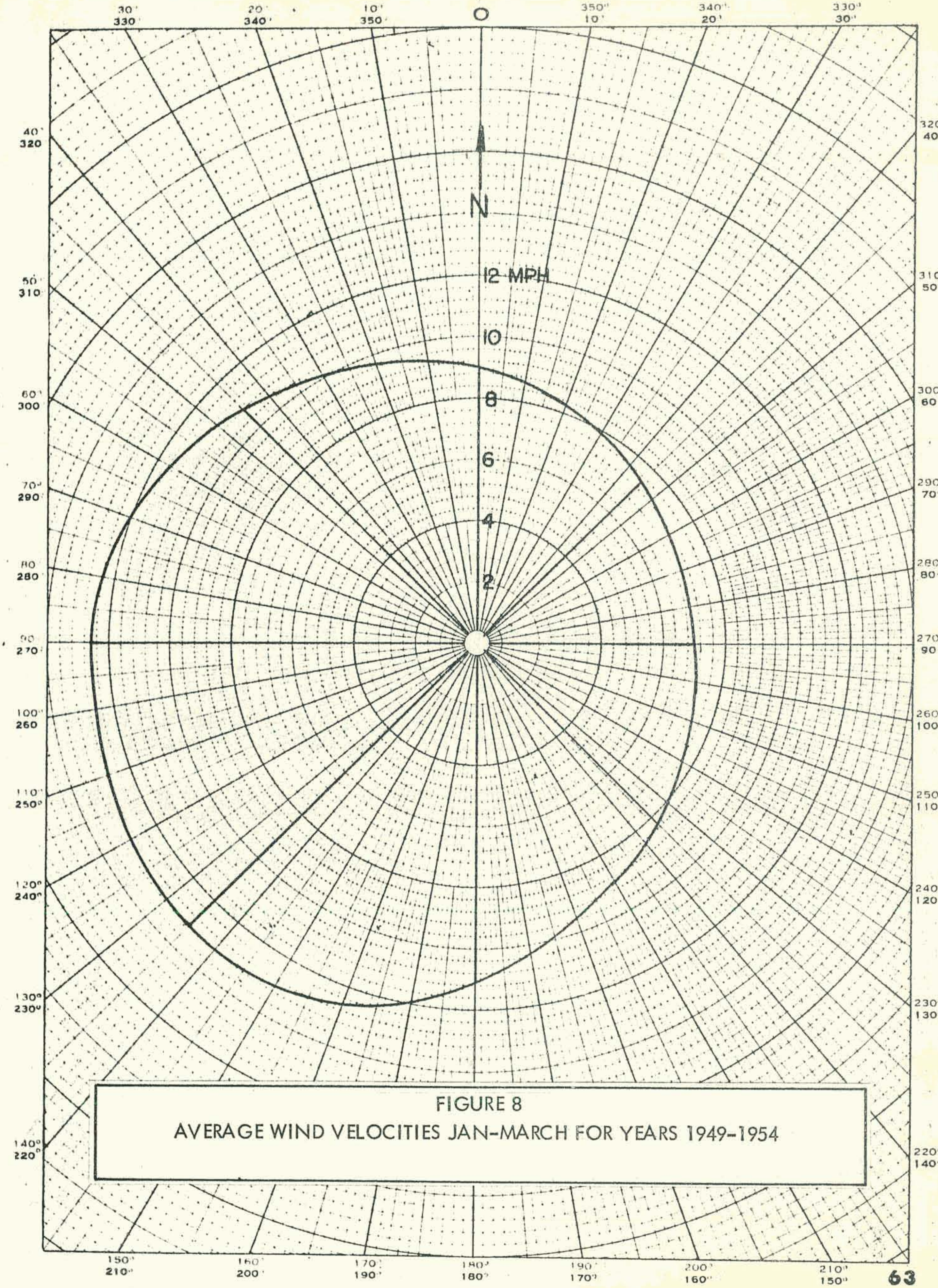


FIGURE 8
AVERAGE WIND VELOCITIES JAN-MARCH FOR YEARS 1949-1954

418 71

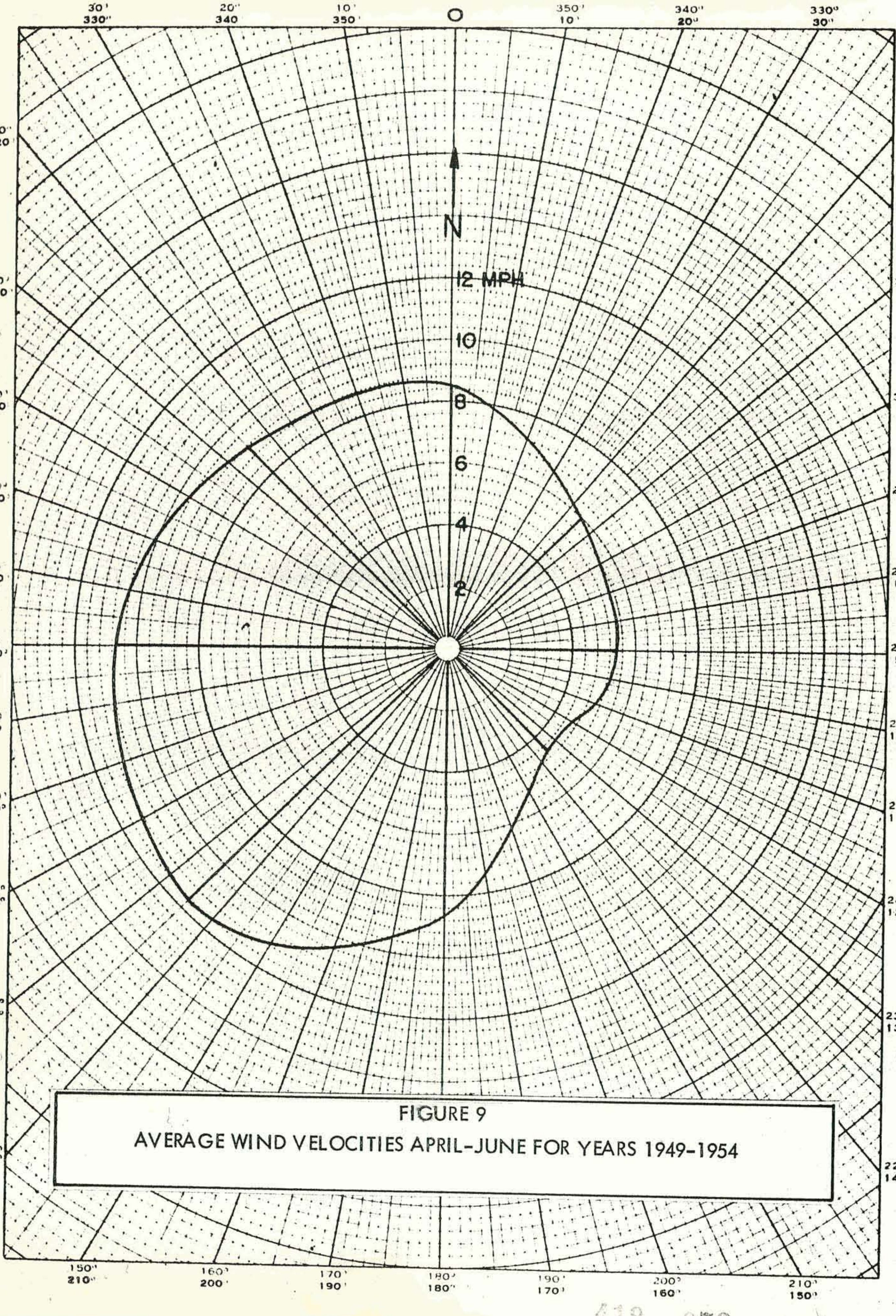


FIGURE 9

AVERAGE WIND VELOCITIES APRIL–JUNE FOR YEARS 1949–1954

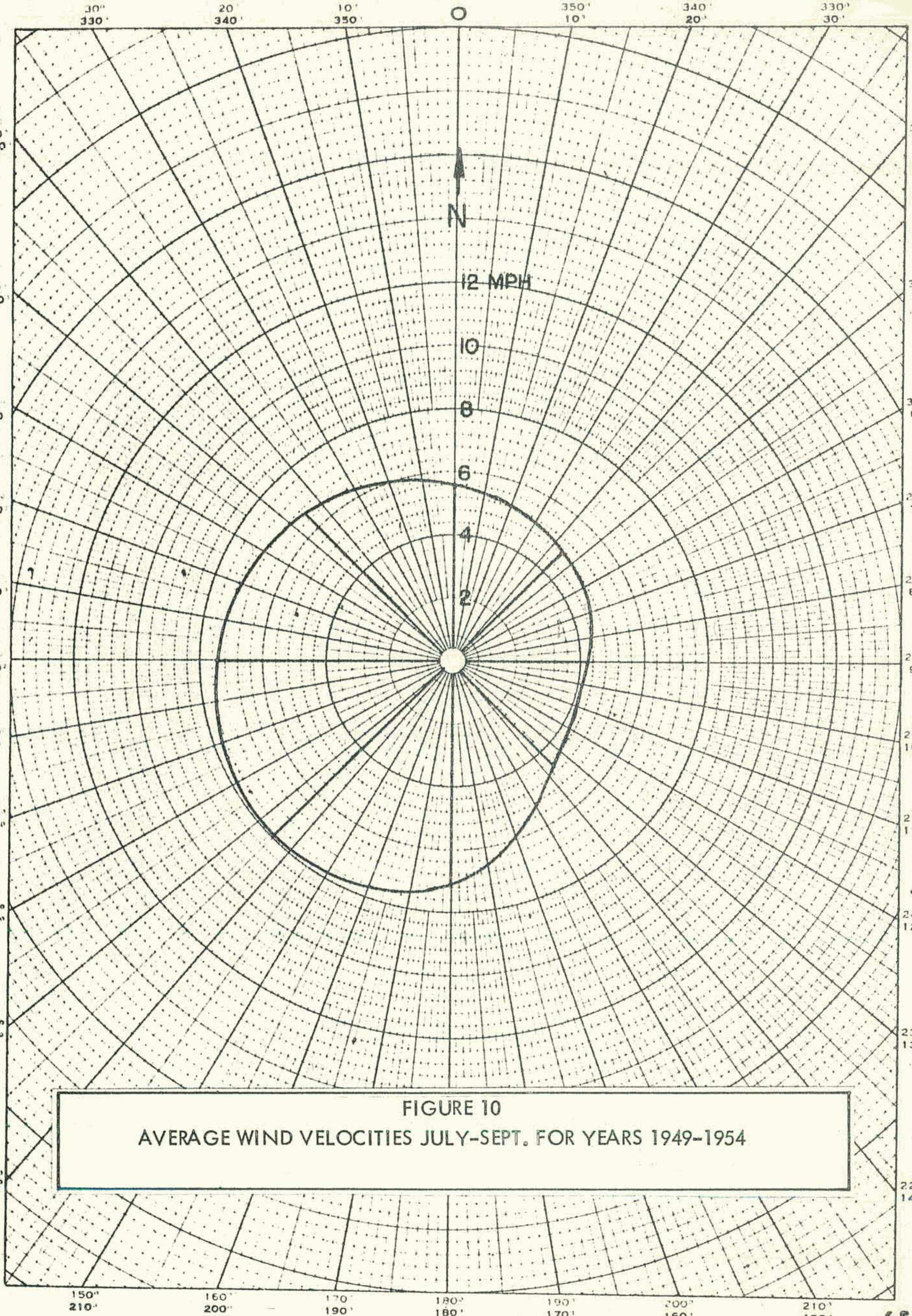


FIGURE 10

AVERAGE WIND VELOCITIES JULY-SEPT. FOR YEARS 1949-1954

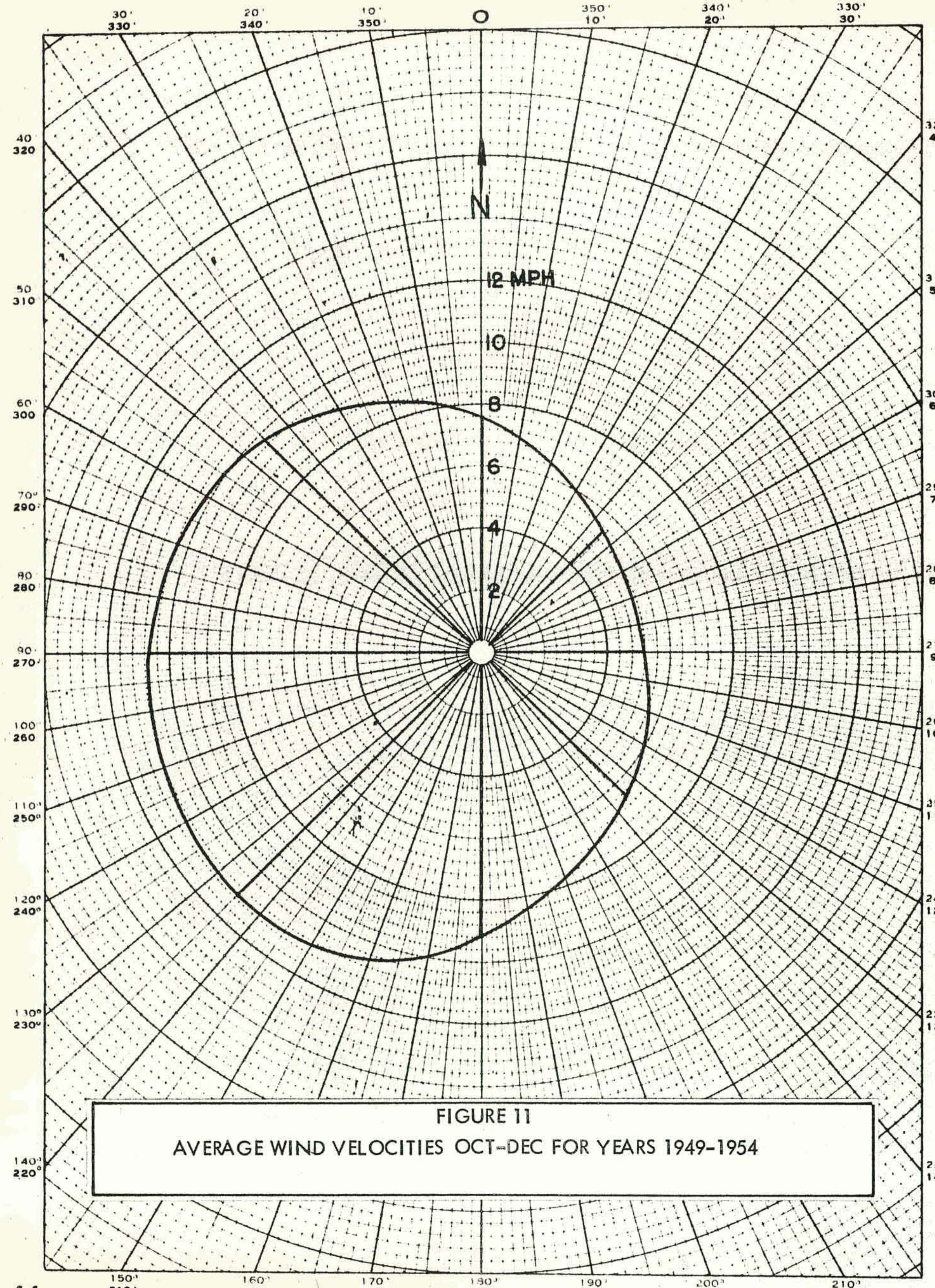


FIGURE 11

AVERAGE WIND VELOCITIES OCT-DEC FOR YEARS 1949-1954

3.6 SITE GEOLOGICAL CHARACTERISTICS

The sedimentary rock strata at the reactor site are Delaware limestone, a mixture of argillaceous cherty blue limestones and calcareous brown shales. These strata are covered by glacial drift which is predominantly clay. A boring analysis taken at a point about 500 feet southwest of the reactor site gave the information shown in Table V.

TABLE V

Strata	Depth (ft.) From	Depth (ft.) To
Clay	0	60
Slab rock	60	63
Hard clay	63	81
Rock	81	85
Hard clay and gravel	85	109
Hard rock	109	115
Clay	115	138
Rock	138	142
Soft clay	142	158
Limestone	158	190

Damage from earthquakes is considered remote based on the earthquake information presented in Figure 12 and the type of structure in which the reactor will be housed.

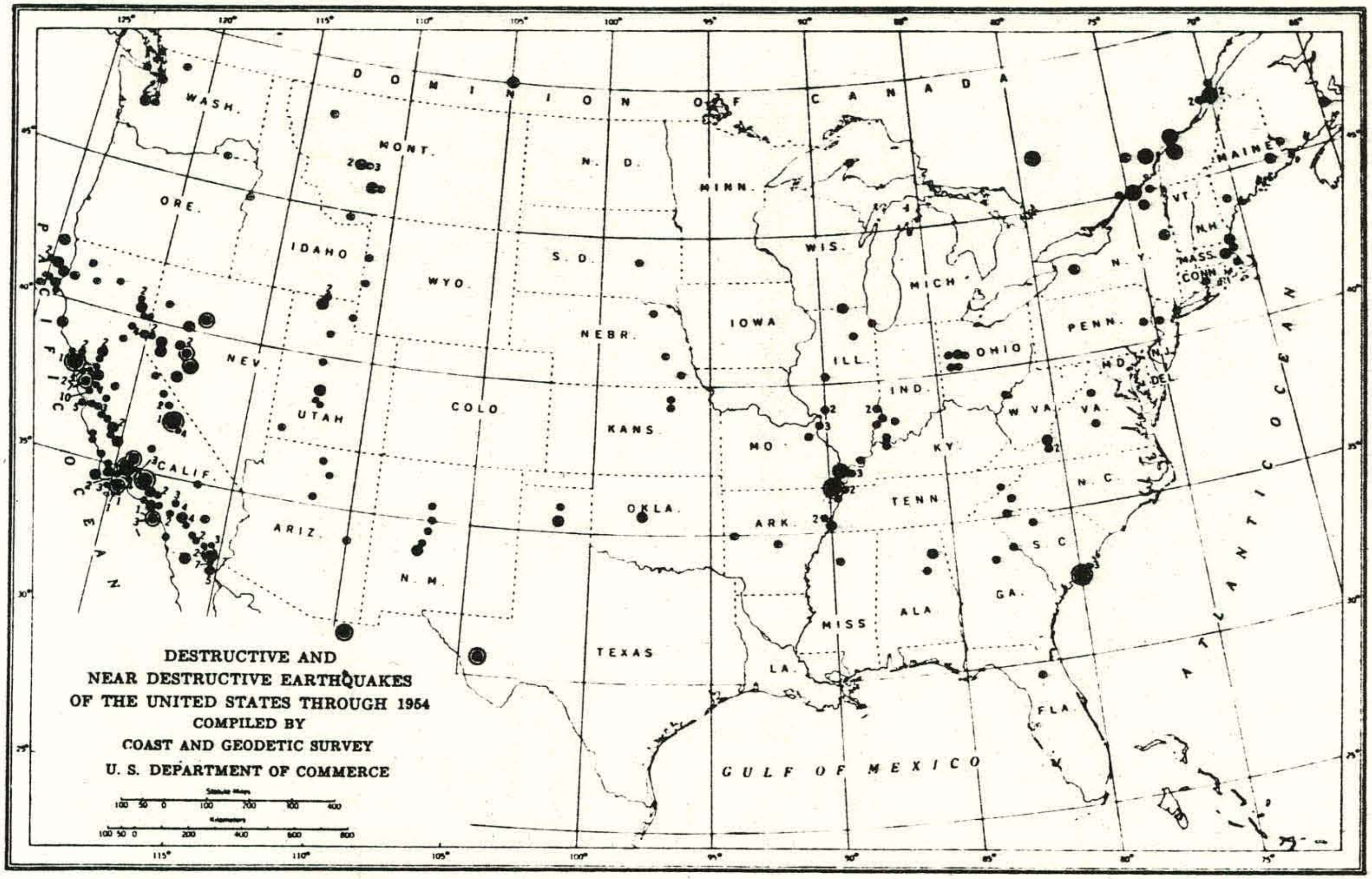


FIGURE 12 EARTHQUAKE LOCATIONS

4. HAZARDS

In presenting the hazards associated with the proposed reactor, both the hazards associated with normal operation and the hazards associated with abnormal conditions or situations will be covered. Because the hazards associated with abnormal conditions vary widely, a special section is included to cover the maximum credible accident.

The Borax programs have helped to confirm that water moderated and cooled reactors are among the safest that can be constructed, and they have demonstrated the shutdown mechanisms by which these reactors protect themselves. Although the safety of these reactors is based upon their ability to absorb reactivity additions internally by moderator changes (specifically by the negative void and temperature coefficients characteristic of these types), these shutdown mechanisms are not completely understood. However, certain conclusions may be drawn with regard to their actions. Extrapolation of experimental data obtained from the Borax I and II experiments has been used successfully to predict the performance of several boiling reactors and is considered to be a valid approach. Since all water reactors possess the ability to function as boilers, these same methods can be used to analyze the behavior of normally nonboiling reactors when subjected to unusual operating conditions or power levels. Therefore, this approach will be used in subsequent paragraphs to analyze the effect of large reactivity additions to the Ohio State reactor using the data of the Borax experiments.

4.1 HAZARDS ASSOCIATED WITH NORMAL OPERATIONS

Hazards associated with normal reactor operations are those resulting from normal exposure, exposure as a result of fuel handling, and failure of fuel element cladding.

4.1.1 Normal Radiation Exposure

Calculations were made to determine the radiation intensities to be expected at various locations in the reactor facility. The results of the calculations for a power of 10 kw are summarized in Table **III.**

TABLE III

RADIATION INTENSITIES

Position	Combined Dose Rate-mrem/hr
Outer Surface of Concrete Shield	0.25
Surface of Water in Reactor Pool	2.50
Surface of Water in Shielding Pool	0.25

These dose rates are equal to, or a factor of, ten less than the recently proposed tolerance levels.

Any overexposures of personnel to radiation through the use of the experimental facilities will result not because of shortcomings in the design of the facility, but rather from negligence or from disregard for procedures. Occurrences of this nature will be avoided by strict adherence to normal operating procedures and policies.

An important potential hazard to operating personnel will be exposure to Argon-41 originating by neutron bombardment of air within the experimental facilities. Therefore, the air will be sealed in the experimental facilities to prevent its escape to the reactor building. Also, a neutron shield provided on the thermal column face prevents air activation in the space between the shield door.

In the air pockets exposed to the highest neutron flux, the air will be activated to a level approximately 10^5 times the maximum permissible concentration under stagnant conditions. Procedures will prohibit opening of the various ports after shutdown until the activated air has had time to decay to a point within acceptable tolerances for direct inhalation.

4.1.2 Radiation Exposure Resulting From Fuel Handling

During normal operations -- except for initial loading and final removal -- fuel handling will be performed with the fuel remaining deeply submerged in the pool. Under these conditions, there is virtually no danger from radiation exposure. When a full core of fuel is stored in the fuel storage pit and the cover is in place, there will be a radiation field of less than 25/mr/hr at the surface of the cover.

As a consequence of the heavy cladding of the fuel elements, their life expectancy is 6 to 8 years. At the end of this time, a suitable cask may be prepared for shipping the elements to a refabricator; this operation can be performed without excessive exposure of personnel. No shipping cask is provided in the initial facility equipment.

4.1.3 Fuel Element Cladding Failure

A fuel element cladding failure may occur in the course of normal operation as a result of corrosion action over a number of years. In this event, some radioactive fission products will escape to the water. However, previous works have shown that a hole in the fuel-plate cladding for an MTR type element has to be several square centimeters in size before the radioactivity can be detected in the pool water.[3] It is highly improbable that holes of this size will occur during normal operation. Operating procedures will prohibit removing the pool water until adequate tests of the water have been run to ensure that the water does not contain harmful amounts of radioactive contaminants. Inadvertent pool drainage is impossible, since no drains or siphon lines are provided. Pool drainage must be accomplished by means of auxiliary pumping or siphoning.

As stated in the description of the fuel, the corrosion rate of the cladding will be less than one mil per year in demineralized water at 50°C. In addition, the 0.036-inch cladding, which will be thicker than is normally found on similar type fuel elements, will ensure longer life. A life of 6 to 8 years may be expected for these elements with continuous temperature control of the demineralized water and maintenance of the resistivity greater than 330,000 ohm-cm.

4.2 HAZARDS ASSOCIATED WITH ABNORMAL CONDITIONS

The hazards associated with abnormal conditions and situations are somewhat more severe for this reactor than are the normal operation hazards. In view of this, several hazard potentialities and possible consequences will be discussed first, then the maximum credible accident and its possible consequences will be discussed. Abnornal conditions and situations include component malfunction or failure, acts of God, acts of sabotage, and acts of negligence.

4.2.1 Component Malfunction

Control and Instrumentation System Failure - Although the inherent safety of the Ohio State reactor can be shown to be good under the worst conditions to be expected for a reactor of this type, dependence on this inherent safety will be limited to those situations where control provisions have failed. The control provisions include a system of scram interlocks to protect the reactor against equipment and instrumentation malfunctions and errors committed in operations. Suitable operating procedures and interlocks will be included to ensure that each instrument is in the correct range and is functioning properly prior to startup.

Descriptions of the instrumentation and control systems to be used in the Ohio State Reactor have been given previously in the description of the facility. The use of duplicate channels and overlapping ranges will provide complete backup for the entire instrumentation system. Incorporation of fail safe features into the design will further protect the reactor in the event of failure of a component or loss of instrument power. These features of the system will make complete loss of all scram circuitry virtually impossible. However, to determine the effect of the more hazardous accidents which might occur in the facility, a startup accident will be postulated in which all but the over power trips fail to scram the reactor. This accident will be described in some detail in Section 4.4.

Included in the instrumentation system will be two period trip circuits (one deriving its signal from the compensated ion chamber and initiating a slow scram) and two over-power trip circuits. The addition of 0.038% k_{eff}/sec will cause both period trips to scram the reactor before serious over-power can be reached. If these trips fail to operate, which is a requisite for the startup accident, the over-power trips will initiate a scram when the power level reaches a value of 1.2 times full power, but the excursion will not reach its peak until the power level has reached approximately 2.1 times full power (This accident does not constitute a maximum credible accident).

Fuel Element Cladding Failure - Two types of cladding failure will be considered. In one case a hole develops in the cladding because of some abrasive action. This type of failure will be discovered by periodic test of the pool water. The other type, caused by local or general overheating of the fuel plates, and is usually considered to be the more serious. Since the temperature of the hottest fuel elements will not exceed 115^{o}F during operation without temperature control, the latter type of failure will not be possible during normal operation. Only during a power excursion will the local heating problem arise; in this situation, other factors will have to be evaluated to determine the most serious aspect of the excursion. This situation is discussed in section 4.3.

Flooding of Beam Ports and Rabbit Tube - Since the beam ports and the rabbit tube are all on the same side of the core, an improbable but credible accident could result in these facilities being replaced suddenly by water. The accident could occur if a heavy object dropped by the side of the core breaks off the beam ports and the rabbit tube. Such an accident would bring about the sudden insertion of about 1%Δk into the reactor. The control rods, in this instance, would be scrammed by this accident and there would be no damage to the core. However, should the scram mechanism fail, the power excursion would be controlled by the formation of steam voids; the accident would be one of lower severity than the maximum credible accident.

The water leakage rate from the broken beam ports would be slow, because they are filled with beam plugs or other equipment. Consequently, with the pool fill valve turned on, the fuel elements can be placed in the fuel element storage pool before the water above the core drops to a hazardous level.

4.2.2 Acts of God, Sabotage or Negligence

Severe Storms, Floods and Earthquakes - Severe storms, such as tornadoes and other high winds, might be expected to cause considerable damage to the reactor facility, but this damage will not be intensified because of the nature of the nuclear facility. A study of the tornado history in the United States over a 35 year period to 1956 shows that 111 tornadoes occurred in Ohio.[3] The largest percentage of these storms occurred in the northern and western portions of the state. Only four tornadoes have been recorded in the Columbus area since 1931. The maximum recorded wind speed in the Columbus area since approximately 1880 was 84 miles per hour, in July, 1916.

As seen from Figures 3, 4, and 5, the terrain around the proposed site is relatively flat. This situation might be conducive to flooding in the event of heavy precipitation. Climatological data for the Columbus area show that the heaviest rainfall recorded since

1880 for any 24-hour period is 3.91 inches, and the heaviest snowfall for any 24-hour period is 11.9 inches. These occurred in September, 1938, and in January, 1910, respectively.

In the event of a flood, the reactor pool is not expected to be damaged. There should be no danger of the flood water mixing with the pool water, since the top of the pool wall will be 20 feet above ground level. Therefore, any radioactivity existing in the pool water will not be dispersed.

According to Heck there have been 11 earthquakes reported having epicenters in Ohio to 1947, including 5 important ones.[4] One of these five, with its epicenter in Columbus, occurred in September, 1884, with an intensity of 6 on the Rossi-Forel scale. The last of the five occurred at Anna and Sidney in March, 1937. The center of this region of earthquakes is 20 to 30 miles west of Columbus. Although there might be future earthquake activity in the Columbus area, the reactor facility should not suffer damage which would result in a serious hazard. Cracking of the shield would probably be the most serious damage. Should the pool liner be ruptured also, pool water would escape from the pool leaving the reactor unshielded in the vertical direction (The reactor would be subcritical because of the absence of moderator). This would result in a radiation level of approximately 10 r/hr at the top of the pool.

<u>Sabotage</u> - The probability of sabotage in connection with the Ohio State Reactor is deemed negligible, since there would be no political or military advantage to be gained. Should a demented person attempt to destroy the reactor, he would find it difficult to commit an act resulting in a reactor excursion as serious as that described in a subsequent section entitled Maximum Credible Accident. Even through disassembly and reassembly of the core in another configuration, it would be virtually impossible to induce a reactor excursion more serious than that described under Maximum Credible Accident. Key switches and building security will be employed to limit access to the reactor and its controls.

Negligence - The history of accidents in almost every industry including the atomic energy industry has shown that negligence has been one of the largest contributing factors to accidents. Promoting continued safety consciousness even though no accidents occur is one of the most difficult tasks in preventing negligence. Negligence will be a problem to be faced even though a proper attitude toward safety consciousness be established. This problem is especially hard to evaluate, because individual personalities form the basis for the negligent actions.

Strict adherence to standard operating procedures will be required and observance of operating restrictions imposed by nuclear safety considerations will be mandatory. Check lists of operations to be followed during periods of startup and shutdown will be used, and log books will be maintained to record meter readings periodically during operation.

The following accident is postulated to occur by a combination of negligence, violation of standard procedures, and failure of the instrumentation system: the glory hole element has been removed. After criticality has been attained and operation at approximately source level has been established, a new fuel element is being moved over the core. The fuel element is then dropped by carelessness and falls by gravity into the glory hole position supplying approximately 2.1% excess k_{eff} in a ramp addition. While the flux and power are increasing, the period trips fail to scram the pile.

4.3 MAXIMUM CREDIBLE ACCIDENT AND ASSOCIATED HAZARDS

4.3.1 Selection of Maximum Credible Accident

From a study of accident potentialities, the maximum credible accident is postulated to occur as a result of a fuel element being dropped into the central or "glory hole" position when the reactor is assumed to be critical at source level. Simultaneously with this occurrence, an accompanying failure of the period trips and level trips in the instrumentation and control system is postulated as previously described. This accident was chosen as the maximum credible, since the period of the excursion is expected to be shorter than for any other credible accident.

4.3.2 Consequences of Maximum Credible Accident

The Borax programs have demonstrated that water cooled and moderated reactors exhibit substantial self-protection against the effects of sudden additions of reactivity, even in the absence of corrective action by the reactor control system. This self-protection is manifest in the negative steam-void coefficient of reactivity and the negative temperature coefficient of reactivity, both of which can effect large reductions in reactivity as reactor power increases. In the next paragraphs, analyses are made of the behavior of the reactor following additions of reactivity.

Characteristics of the Ohio State Reactor that determine its behavior during power transients resulting from large reactivity additions are similar to those of the Borax I reactor. Consequently, transient behavior of the Ohio State Reactor can be predicted by adjusting the Borax data to account for the differences in design of the two reactors. The significant quantitative characteristics of the two reactors are compared in Table IV.

Extension of the Borax results to the Ohio State Reactor is made on the basis that the exponential period determines the total energy release and the fuel temperatures attained during an excursion. As determined from the Borax experiments, the excess reactivity and neutron lifetime have effects only as they jointly determine the period.

TABLE IV

OHIO STATE REACTOR AND BORAX I CHARACTERISTICS

Characteristics	Ohio State	Borax I
Fuel plate "meat"	14.1 w/o U-Al alloy (fully enriched)	18 w/o U-Al alloy (fully enriched)
Fuel plate cladding	1100 aluminum	1100 aluminum
"Meat" thickness	0.036 in.	0.020 in.
Cladding thickness	0.036 in.	0.020 in.
Ratio $\frac{\text{Aluminum volume in core}}{\text{Water volume in core}}$	0.722	0.626
Coolant channel thickness	0.192 in.	0.117 in.
Core volume (approx.)	91 liters	106 liters
Void coefficient of reactivity	-0.28% k/% coolant void	-0.24%k/% coolant void
Temperature coefficient of reactivity (room temperature)	-0.021% k/^{o}C	-0.01% k/^{o}C
Effective neutron lifetime	7.0 x 10^{-5} sec.	6.5 x 10^{-5} sec.
Power ratio in core, max./av.	2.23	1.82

The more important characteristics of the Ohio State Reactor and Borax I will be considered. They include the following:

	Ohio State	Borax I
Aluminum to-water ratio	0.722	0.626
Coolant channel thickness	0.192 in.	0.117 in.
Plate thickness	0.108 in.	0.060 in.
Void coefficient	-0.28% k/% void	-0.24% k/% void
Power ratio	2.23	1.82

Differences in the behavior of the two reactors, based on the above characteristics, are discussed in the following paragraphs, each of which is headed by the name of the characteristic discussed.

Aluminum-to-Water Ratio - If the data presented in Reference 8 for Borax I and Borax II (specifically, figures 14 and 17b) are compared, the maximum fuel plate temperature rise for Borax II is seen to be higher than that for Borax I for a given period of the excursion. However, the aluminum-to-water ratio for Borax II is seen to be lower than that for Borax I. Therefore, the assumption is made that, since the aluminum-to-water ratio for Borax I is lower than that of the Ohio State Reactor, the maximum temperature rise in the hottest fuel plates of the Ohio State core will not exceed that of Borax I for a given period on the basis of the aluminum-to-water ratio difference.

Plate Thickness - Since the Ohio State fuel plates are thicker than those of Borax I, they will have a higher total heat capacity and a higher central metal temperature for a given sheath temperature. However, the difference in the central metal temperature and the sheath temperature will be small.

During the excursion, the transfer of heat into the water removes heat from the fuel plate and limits its temperature rise. The important characteristic of the plate during the excursion is the heat flux which it can supply to the water for a given temperature difference between the water and the plate surface. The plate thickness affects this characteristic only in that the meat temperature must be slightly higher to compensate for the thickness, but the surface temperature should not be altered appreciably by this effect. The important consideration is, of course, the temperature of the plate surface as this temperature determines when the cladding will melt and when the aluminum-water chemical reaction is probable.

Coolant Channel Thickness - The fuel plate temperature rise for Borax I elements was higher than that of the plates in Borax I elements for a given period. The coolant channel thickness for Borax II elements was also greater than that of Borax I elements. Since the coolant channel thickness of the Ohio State plates is between that of Borax I and II, on this basis, the fuel plate temperature rise during an excursion is expected to be between that of Borax I and Borax II for any given period. Figure 13 indicates that this effect is small.

Void Coefficient - The calculated void coefficient of reactivity for Borax I was higher than that of Borax II in the ratio:

$$\frac{-0.24\%\ k_{eff}/\%\ \text{void}}{-0.10\%\ k_{eff}/\%\ \text{void}} = 2.4$$

This factor would be expected to cause a lower energy release per fuel plate in Borax I which occurrence was observed. The measurements made with subcooled water at periods down to 23 milliseconds showed that the energy release per fuel plate in Borax II was between 1.7 and 2.0 times that of Borax I. Furthermore, with the fuel plates for both reactors being the same, the fuel plates in Borax II were found to experience higher temperature rises for a given period than those of Borax I. Since a similar situation exists in the Ohio State reactor, similar behavior is expected. The void coefficient of the Ohio State core is slightly greater than that of Borax I. Therefore, the temperature rise in the OSU fuel plates on this basis is expected to be slightly lower or nearly the same as that in Borax I.

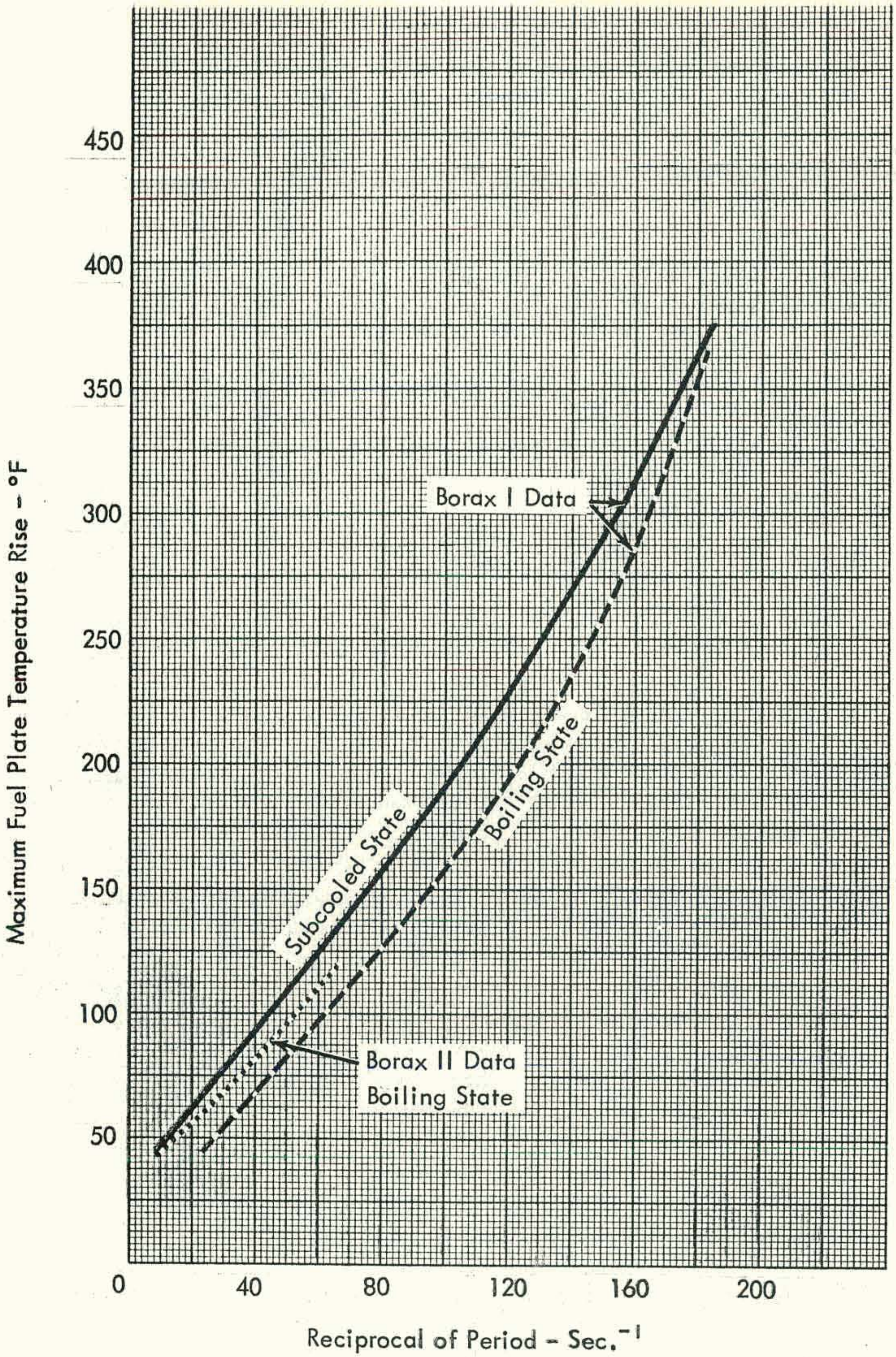


FIGURE 13

MAXIMUM FUEL PLATE TEMPERATURE RISE VS RECIPROCAL PERIOD

Power Ratio - The remaining difference between the Ohio State Reactor and Borax I is in gross maximum to average power ratio for the two reactors. Since the hottest point in the hottest fuel plate is the most important consideration, this difference must be taken into account. With the power ratios being taken into consideration, the maximum temperature rise in the hottest fuel plate in the Ohio State Reactor is expected to exceed that of Borax I by a factor corresponding to the ratio 2.23/1.82 = 1.23.

4.3.3 Conclusion

Results of the foregoing analysis for the excursion in which the total excess k_{eff} added is 2.1%, indicate that the maximum fuel plate temperature rise to be expected in the Ohio State core will be only slightly greater than 1.23 times that experienced in Borax I for the same Δk_{eff}. In order to arrive at an estimate of the temperature which will be experienced, several other factors will be considered.

If the data from Figures 13 and 14 in Reference 8 are replotted to show the maximum temperature rises for the excursions started from both subcooled and boiling conditions, the curves shown in Figure 13 are obtained. The curve for the excursions starting from the subcooled state has been extrapolated on the basis of the relation of the two curves in the region of the experimental data; viz., the ratio of the two curves decreases at a constant rate. That the two curves tend to merge in the region characterized by short periods is to be expected based on the following assumption.

As periods experienced become shorter, the heat liberated in an excursion increases. Therefore, the quantity of heat required to raise the temperature of the subcooled water to the saturation point becomes a smaller fraction of the heat liberated. Consequently the maximum temperatures attained will tend to become independent of the subcooling as shown in Figure 13.

A second item for consideration in evaluating the maximum temperature rise of the fuel plate is the rate of fall of the fuel element into the reactor as compared to rate of fall of the poison rod out of the Borax reactor. Since these rates are controlled by hydrodynamic properties it is difficult to determine the exact rate of fall. It is, however, evident that the fuel element will fall at a slower rate than the rod and will, therefore, produce a longer period. Finally, in order to determine the temperature rise from Figure 13, a period must be selected. The relation for the asymptotic period

$$\tau = \frac{1^*}{k_{ex}(1-B)-B}$$

yields a value of 5.25 milliseconds for the value $k_{ex}=.021$ and a lifetime of $1^*=7\times10^{-5}$ seconds. The actual period may be somewhat longer than this as a result of the excursion possibly being terminated prior to the complete insertion of the element but, in the interest of conservatism, a reciprocal period of 190 sec^{-1} will be used. The curve at this point indicates for zero subcooling a value of 390°F temperature rise. Assuming, pessimistically, that the ratio for subcooling over saturated at long periods applies at the shorter periods this value becomes 390° x 1.44 = 560°. Then applying the power ratio factor of 1.23 the indicated temperature rise is 690°. When added to the saturation temperature the maximum expected plate surface temperature is approximately 900°. This temperature is well below the melting temperature of approximately 1200°F for 1100 aluminum.

In view of the experimental results from Borax I, this temperature is considered to be highly pessimistic.[6] A power excursion from a power level of one watt in Borax I with a period as short as 0.005 second (2.1% excess k_{eff} added) resulted in a power level of 2600 megawatts; however, the maximum temperature in the hottest fuel plate never exceeded 640°F.

4.4 STARTUP ACCIDENT EXCURSION AND DROPPED FUEL ELEMENT EXCURSION

Two accidents which are less severe but somewhat more probable than the maximum credible accident are described in this section. The first is the so-called startup accident, in which both period trip circuits fail and the reactor is shut down by the action of the power level safetics. It was assumed for this analysis that although not provided for in the control system, all rods were being withdrawn simultaneously. The second accident is one in which a fuel element is dropped into the glory hole position when the reactor is critical at a very low power (i.e. source level) and the excursion is again stopped by the action of the control rods. In both cases, it was assumed that the rods were in their fully withdrawn, and therefore least effective, position when the scram action begins.

The relations given below, which are attributed to H. W. Newson, were used in the transient analyses.

(1) $\ln \frac{N}{N_o} = \frac{Rt^2}{2}$ (Equation for flux prior to scram, delayed neutron effect neglected)

(2) $\frac{1}{\tau} = \frac{Rt}{\ell^{*}-2}$

(3) $N = N_s e^{\left(\frac{t}{\tau_s} - \frac{2.87\,S}{\ell^{*}}\right) t^3}$ (Equation for flux after scram, t = time after scram)

(4) $X = 1/2(17.22)t^2$ (The assumed time displacement of the rods and fuel element)

Where:

N = Neutron flux, neutrons/cm^2-sec

N_o = Flux level of start, neutrons/cm^2-sec

N_s = Flux at scram, neutron/cm^2-sec

ℓ^* = Mean time which elapses from the time neutrons are produced in fission until they initiate another fission or are lost to the reaction, sec.

R = Reactivity insertion rate, Δ k/sec

t = time, sec

τ_s = Period at scram, sec

S = Sensitivity of rod Δ k/ft

X = Displacement of rod, or dropped element, ft

Data used in calculations:

N_f = 0.89 x 10^{11} neutrons/cm^2-sec (Full power flux level)

N_t = 1.2 N_f = 1.07 x 10^{11} neutrons/cm^2-sec (Trip flux level)

N_o = 10^3 neutrons/cm^2-sec (Source level)

ℓ^* = 7 x 10^{-5} sec

S = 0.06 Δ k/ft

10 MS = time lapse from time neutron trip level is reached until control rods are released.

Two time histories of N/N_f illustrating an accident caused by withdrawing the rods at 3 in/min until the reactor reached trip level, and an accident caused by dropping a fuel element into the critical reactor, were plotted respectively in Figure 14 and Figure 15. In both cases, the rods were released at scram level or 25 ms after trip level. However, in the first case, Figure 14, R was a constant,

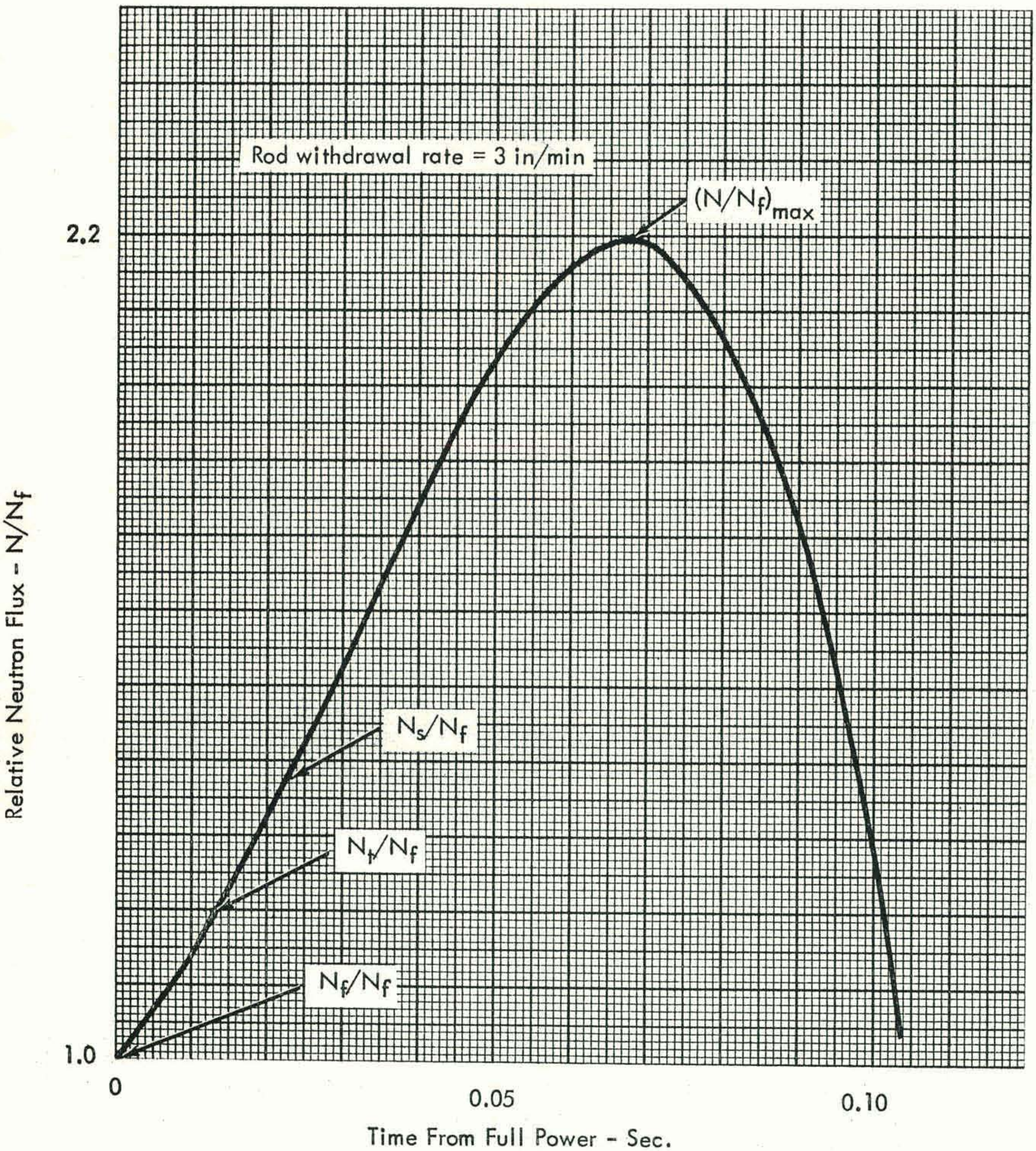


FIGURE 14

N/N_f VS TIME FOR 10 KW TRAINING REACTOR DUE TO RUNAWAY ACCIDENT

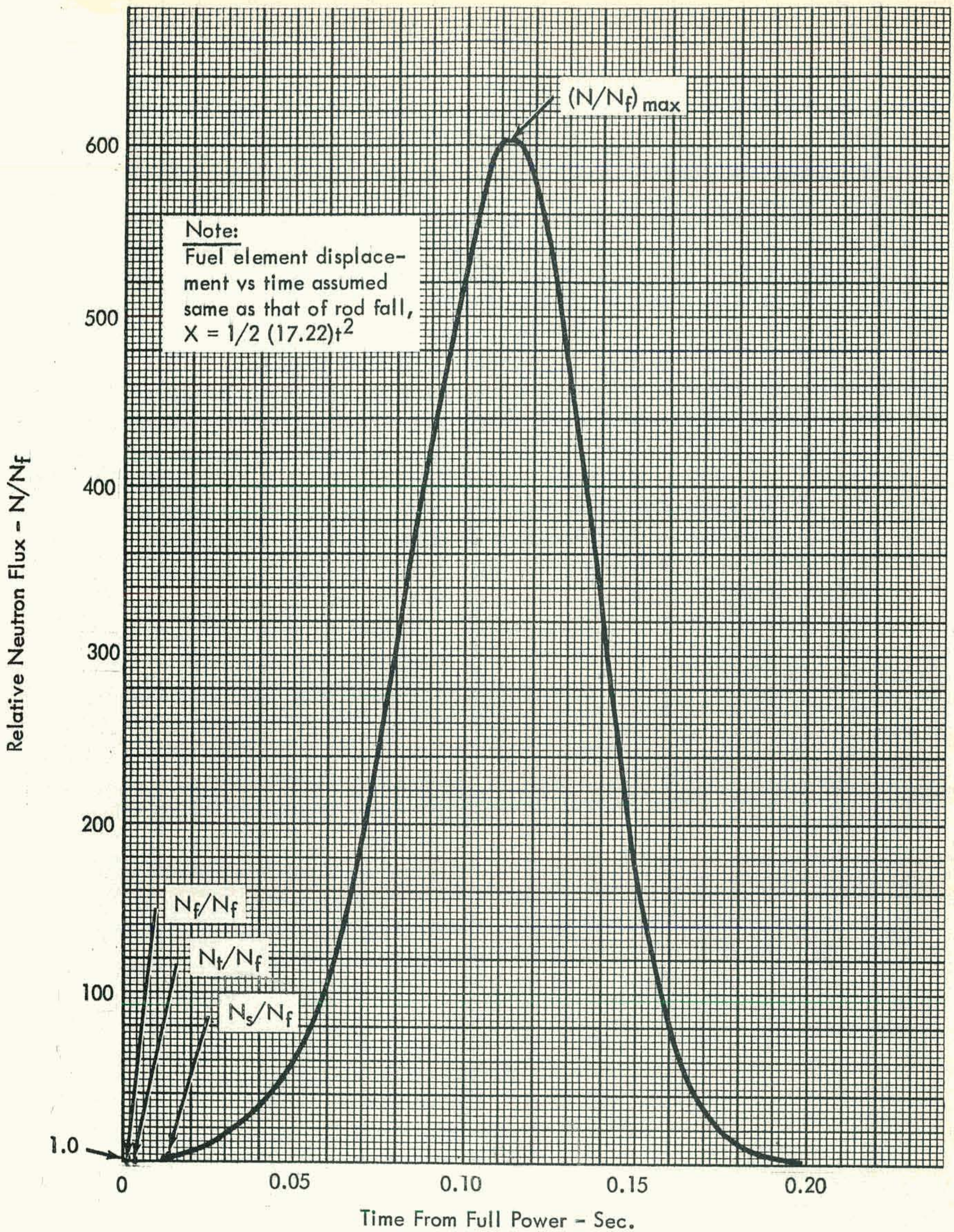


FIGURE 15 N/N_f VS TIME FOR 10 KW TRAINING REACTOR DUE TO ACCIDENTAL FUEL ELEMENT DROP while in the second case, Figure 15, R = R(t). It was assumed the time displacement of the fuel element was equal to that of the rods in a free fall through water.

The following is a tabulation of excursion data for the two cases.

EXCURSION DATA

	Rod Withdrawal @ 3 in/min	Dropped Fuel Element
Rate of Δ k insertion, Δ k/sec	0.000381	R = 0.0197t
Time to reach N_{max}, sec	2.660	0.6196
Period at scram, sec	0.0702	0.0132
Maximum Power, kw	21.00	6,060
Relative flux overshoot, N_{max}/N_f	2.10	606
Maximum Temp. Rise, ^{o}F	1	124

References

1. Gorker, G. E., "Nuclear Power Plant Control," Notes for an introductory course, DC 56-12-93, p. 10-25, 1956

2. Schultz, M. A., Control of Nuclear Reactors and Power Plants. New York: McGraw-Hill Book Co., Inc. (1955)

3. Chastain, J. W., Redmond, R. F., Klickman, A., Anno, J., and Fawcell, S. L. "Hazards Summary Report for the Battelle Research Reactor," BMI-ACRS-601 (Rev.) (1955)

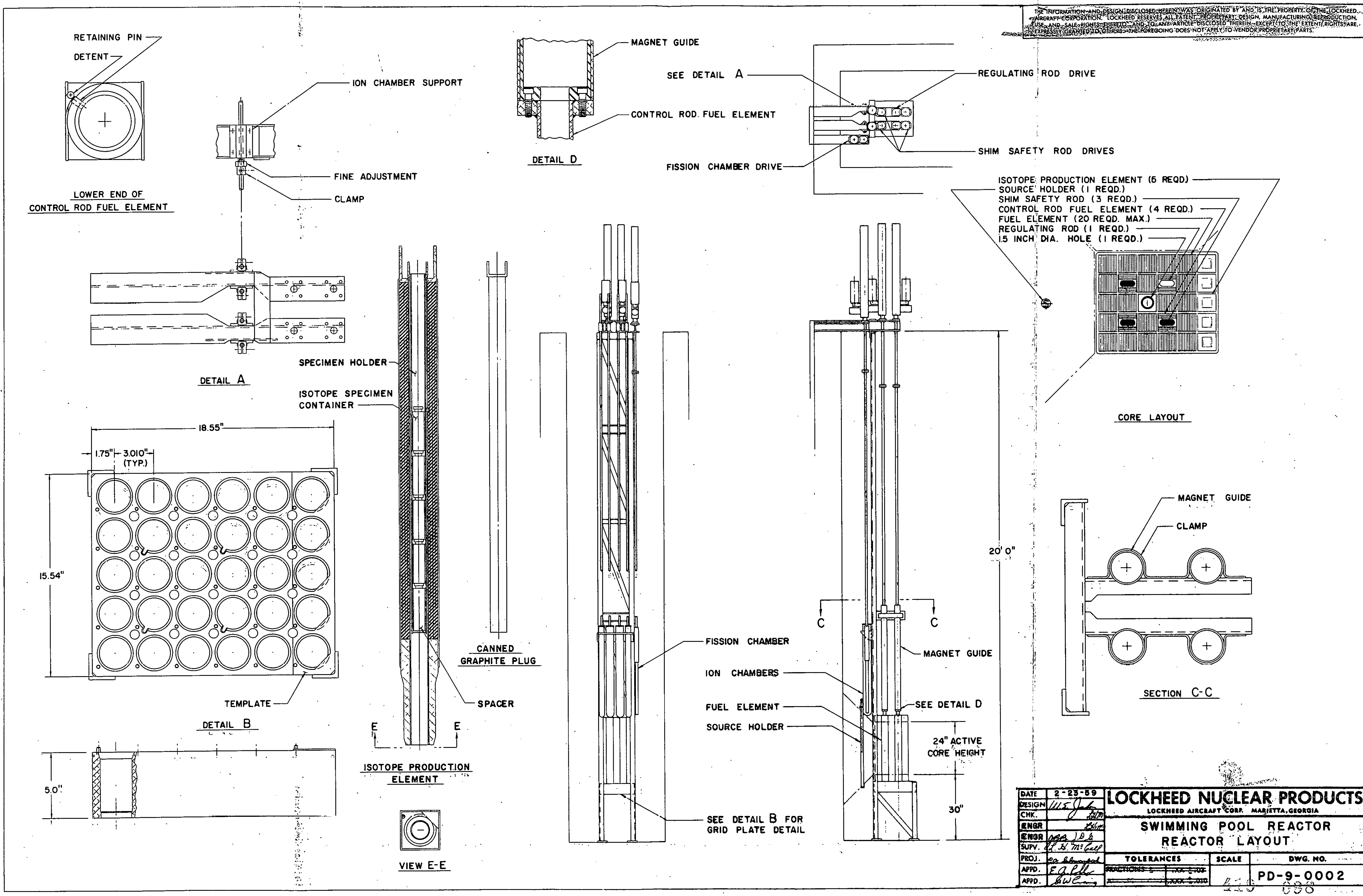
4. Heck, N. H., "Earthquake History of the United States (Exclusive of California and Western Nevada) and Alaska," U. S. Dept. of Commerce, Coast and Geodetic Survey, Serial No. 609

5. U. S. Department of Commerce, National Bureau of Standards Handbook 52, March 1953, "Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water," Superintendent of Documents, Washington, D. C.

6. Dietrich, J. R., and Layman, D. C. "Transient and Steady State Characteristics of a Boiling Reactor. The Borax Experiments, 1953," AECD-3840 (1954)

7. Dietrich, J. R., "Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor," AECD-3668 (1954)

8. Dietrich, J. R., "Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors," International Conference on the Peaceful Uses of Atomic Energy (Geneva), Vol. 13, paper 481, pp. 88-101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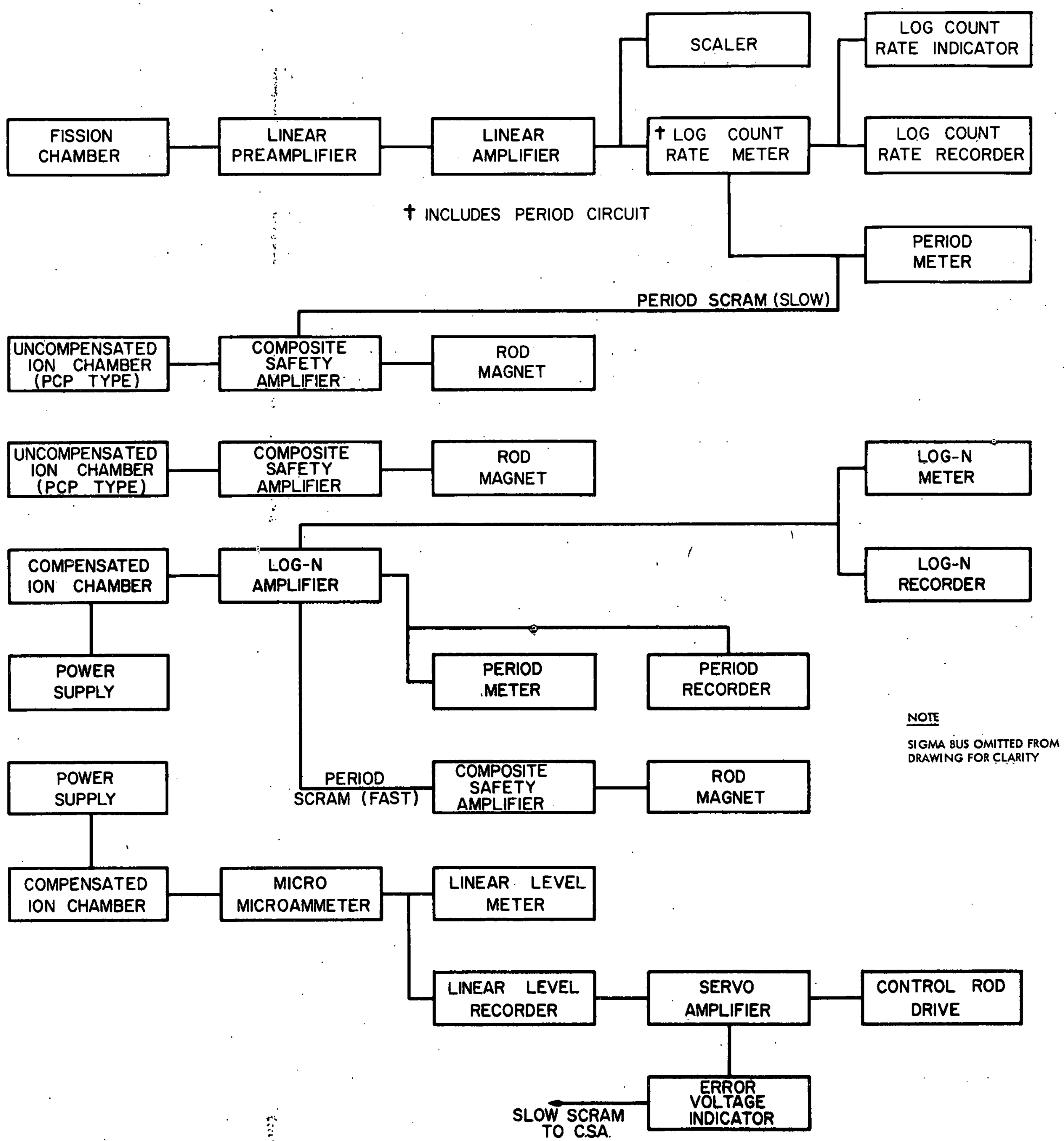
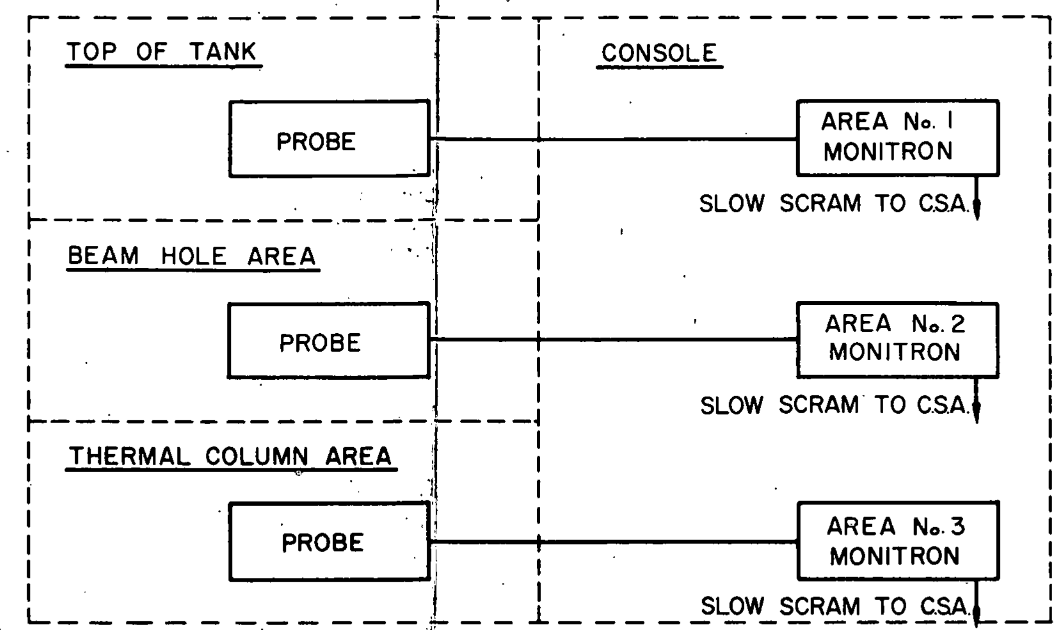
RETAINING PIN
DETENT
LOWER END OF CONTROL ROD FUEL ELEMENT
ION CHAMBER SUPPORT
FINE ADJUSTMENT
CLAMP
DETAIL A
MAGNET GUIDE
CONTROL ROD FUEL ELEMENT
DETAIL D
SEE DETAIL A
REGULATING ROD DRIVE
SHIM SAFETY ROD DRIVES
FISSION CHAMBER DRIVE
ISOTOPE PRODUCTION ELEMENT (5 REQD.)
SOURCE HOLDER (1 REQD.)
SHIM SAFETY ROD (3 REQD.)
CONTROL ROD FUEL ELEMENT (4 REQD.)
FUEL ELEMENT (20 REQD. MAX.)
REGULATING ROD (1 REQD.)
1.5 INCH DIA. HOLE (1 REQD.)
CORE LAYOUT
SPECIMEN HOLDER
ISOTOPE SPECIMEN CONTAINER
SPACER
E
E
ISOTOPE PRODUCTION ELEMENT
CANNED GRAPHITE PLUG
VIEW E-E
18.55"
1.75"
3.010" (TYP.)
15.54"
TEMPLATE
DETAIL B
5.0"
FISSION CHAMBER
ION CHAMBERS
FUEL ELEMENT
SOURCE HOLDER
SEE DETAIL B FOR GRID PLATE DETAIL
C
C
MAGNET GUIDE
SEE DETAIL D
20' 0"
24" ACTIVE CORE HEIGHT
30"
MAGNET GUIDE
CLAMP
SECTION C-C
DATE 2-23-59
DESIGN
CHK.
ENGR
ENGR
SUPV.
PROJ.
APPD.
APPD.
LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA
SWIMMING POOL REACTOR
REACTOR LAYOUT
TOLERANCES
SCALE
DWG. NO.
PD-9-0002



BLOCK DIAGRAM - NUCLEAR INSTRUMENTATION - ALTERNATE NO. I



BLOCK DIAGRAM - AREA MONITORS

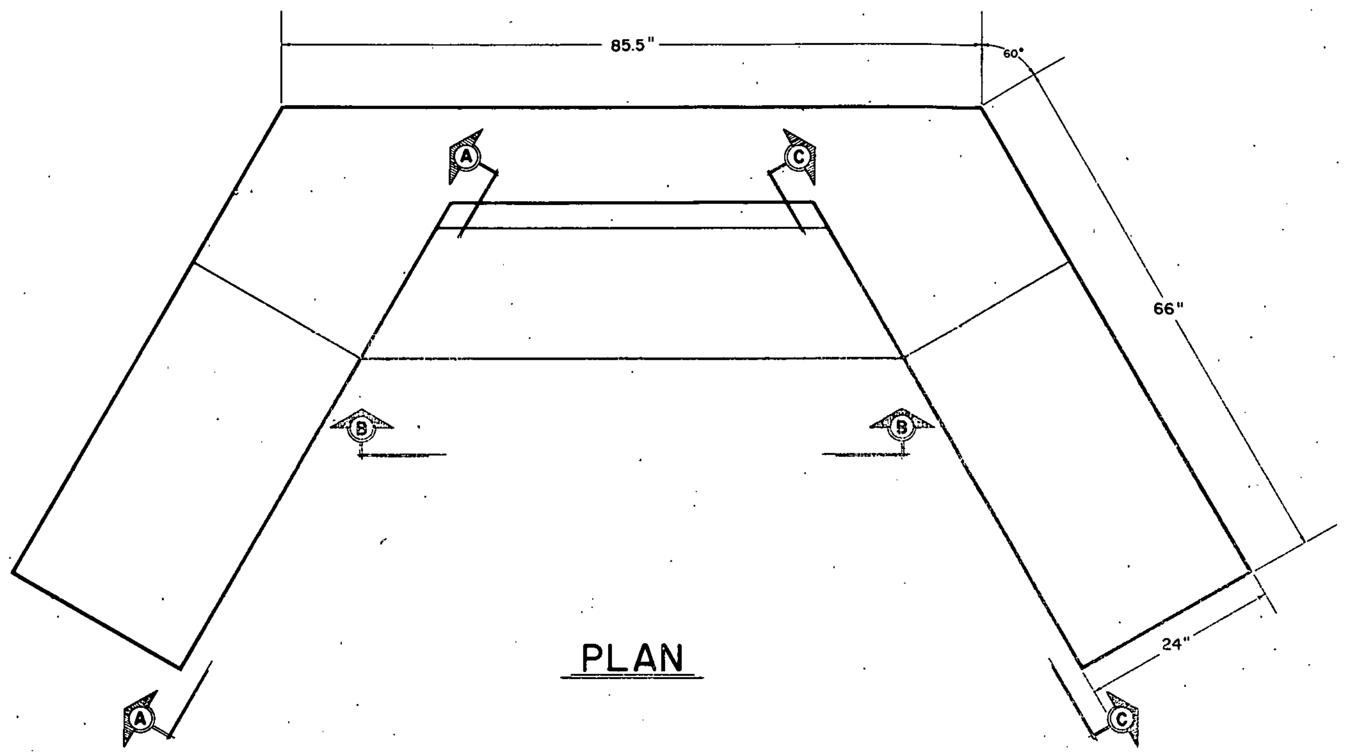
419 099

DATE	2-23-59
DESIGN	Anderson
CHK.	WD
ENGR.	WD
ENGR.	[illegible]
SUPV.	[illegible]
PROJ.	[illegible]
APPD.	[illegible]
APPD.	[illegible]

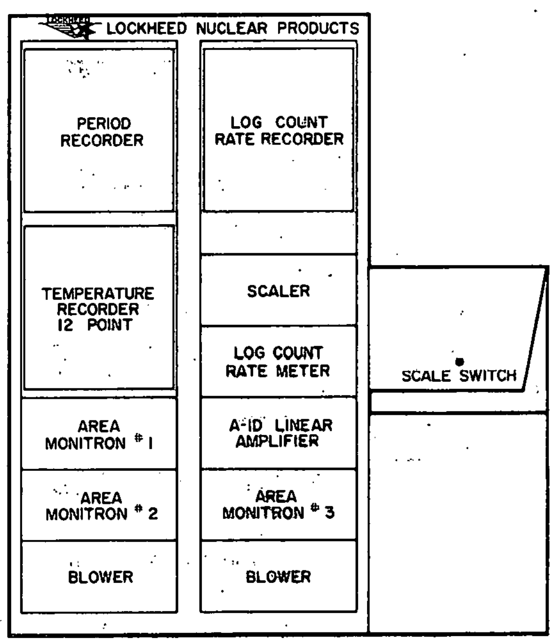
LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

SWIMMING POOL REACTOR INSTRUMENTATION BLOCK DIAGRAM ALTERNATE NO. I

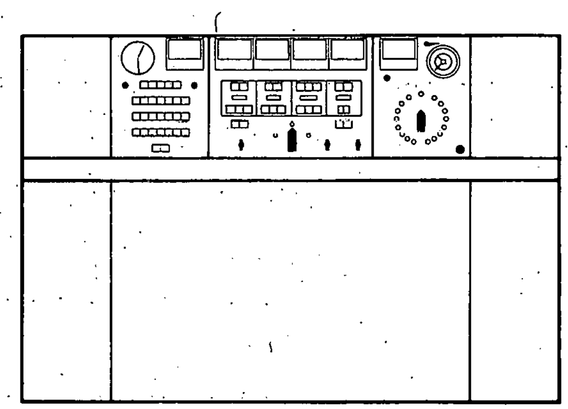
TOLERANCES	SCALE	DWG. NO.
		PD-9-0003



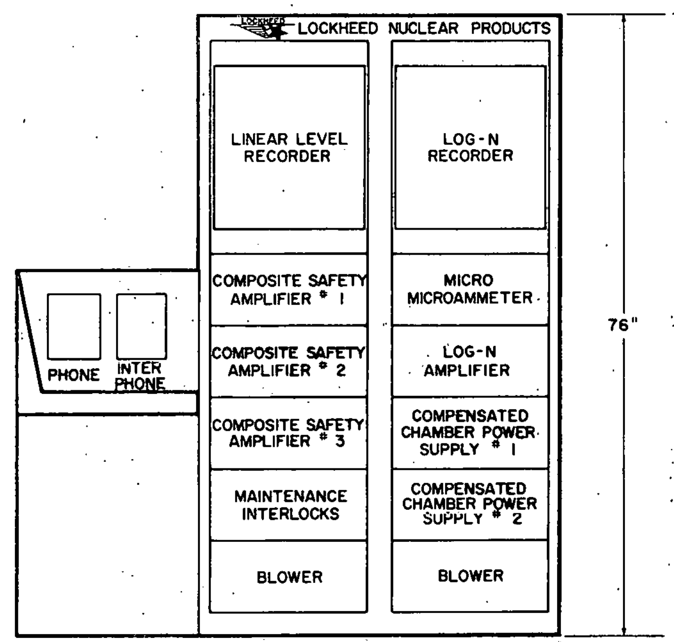
PLAN



ELEVATION A-A



ELEVATION B-B

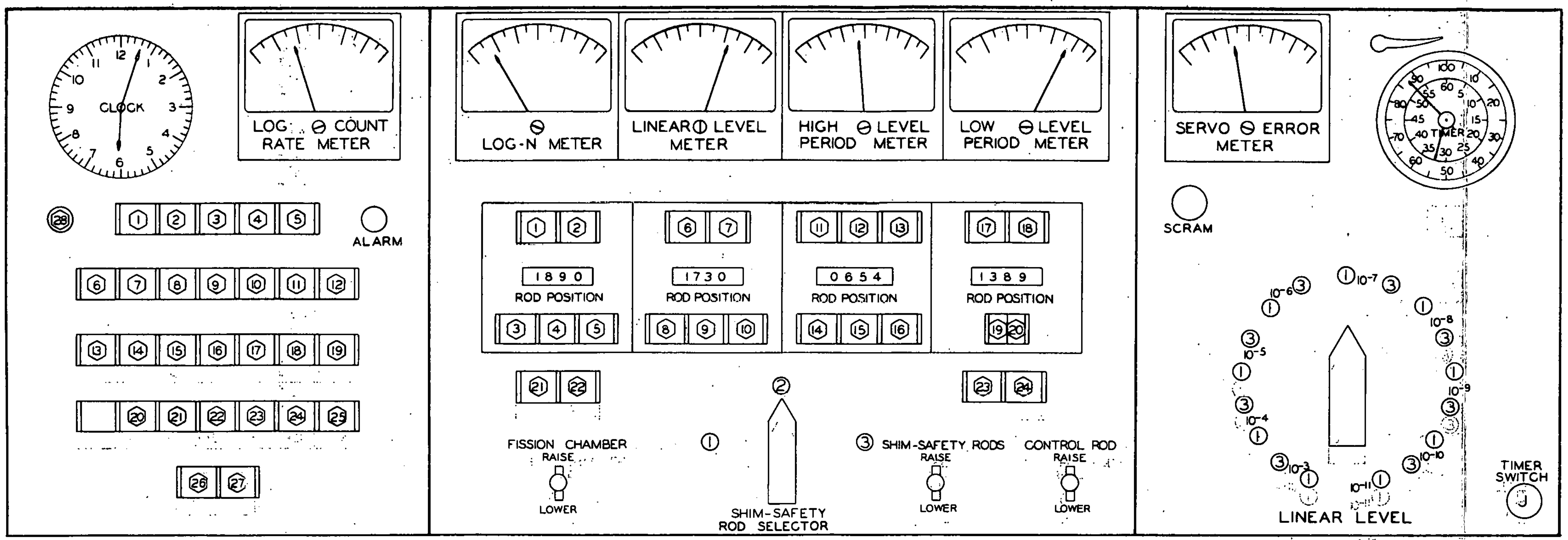


ELEVATION C-C

SCALE
0 4 8 12 16 20 24 INCHES

419 100

DATE	2-23-59	**LOCKHEED NUCLEAR PRODUCTS** LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA
DESIGN	Anderson, R.	
CHK.	WO	SWIMMING POOL REACTOR CONTROL CONSOLE
ENGR.	WO	
ENGR.	[illegible]	
SUPV.	[illegible]	
PROJ.	[illegible]	TOLERANCES / SCALE / DWG. NO.
APPD.	[illegible]	
APPD.	[illegible]	PD-9-0004



* KEY

1. INTERLOCKS SHORTED
2. CONTROL POWER
3. INSTRUMENT POWER
4. MAGNET POWER
5. PUMP ON
6. SCRAM CONSOLE
7. SCRAM BULK SHIELDING AREA
8. SCRAM ROD DRIVE AREA
9. SCRAM THERMAL COLUMN AREA
10. SCRAM BEAM HOLE AREA
11. LEVEL SCRAM
12. PERIOD SCRAM
13. SAFETY AMPLIFIER NO. 1
14. SAFETY AMPLIFIER NO. 2
15. SAFETY AMPLIFIER NO. 3
16. CHAMBER POWER SUPPLY NO. 1
17. CHAMBER POWER SUPPLY NO. 2
18. SOURCE MISSING
19. EXPERIMENT NO. 1
20. EXPERIMENT NO. 2
21. HIGH LEVEL ROD DRIVE AREA
22. HIGH LEVEL THERMAL COLUMN AREA
23. HIGH LEVEL BEAM HOLE AREA
24. SERVO TROUBLE
25. SCRAM LOW LEVEL PERIOD
26. ANNUNCIATOR TEST
27. ANNUNCIATOR ACKNOWLEDGE
28. MAGNET POWER KEY SWITCH

* NOMENCLATURE INDICATED BY KEY WILL BE ENGRAVED ON ANNUNCIATORS.

* KEY

1. CLUTCH
2. UPPER LIMIT
3. JAM
4. DRIVE
5. LOWER LIMIT
6. CLUTCH
7. UPPER LIMIT
8. JAM
9. DRIVE
10. LOWER LIMIT
11. CLUTCH
12. SHIM RANGE
13. UPPER LIMIT
14. JAM
15. DRIVE
16. LOWER LIMIT
17. UPPER LIMIT
18. LOWER LIMIT
19. SERVO
20. SERVO PERMIT
21. UPPER LIMIT
22. LOWER LIMIT
23. GANG LOWER
24. SCRAM RESET

* NOMENCLATURE INDICATED BY KEY WILL BE ENGRAVED ON SWITCHES

SCALE

0 1 2 3 4 5 6 INCHES

419 101

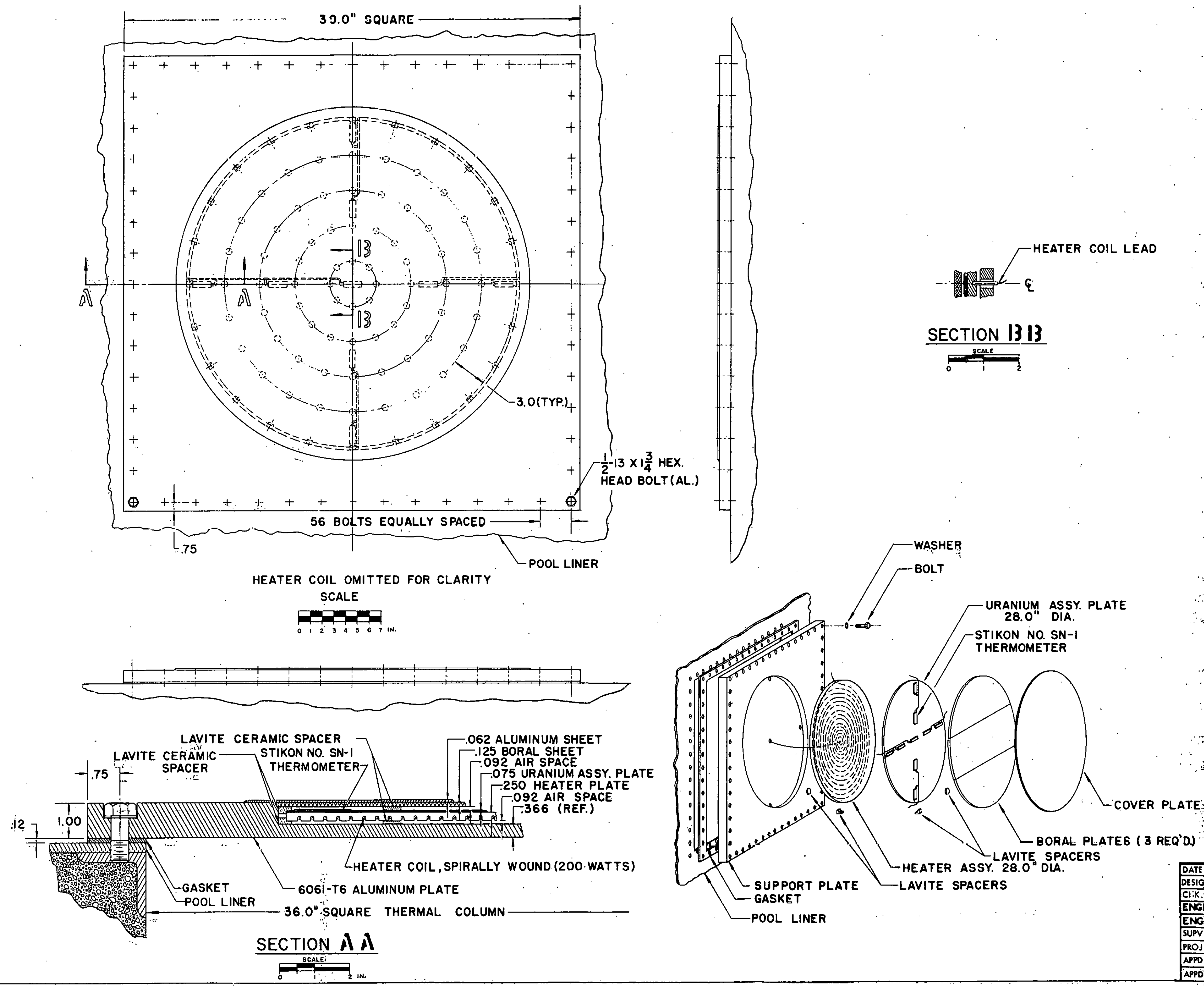
DATE	2-23-59
DESIGN	Anderson
CHK.	
ENGR.	
ENGR.	
SUPV.	
PROJ.	
APPD.	
APPD.	

LOCKHEED NUCLEAR PRODUCTS

LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

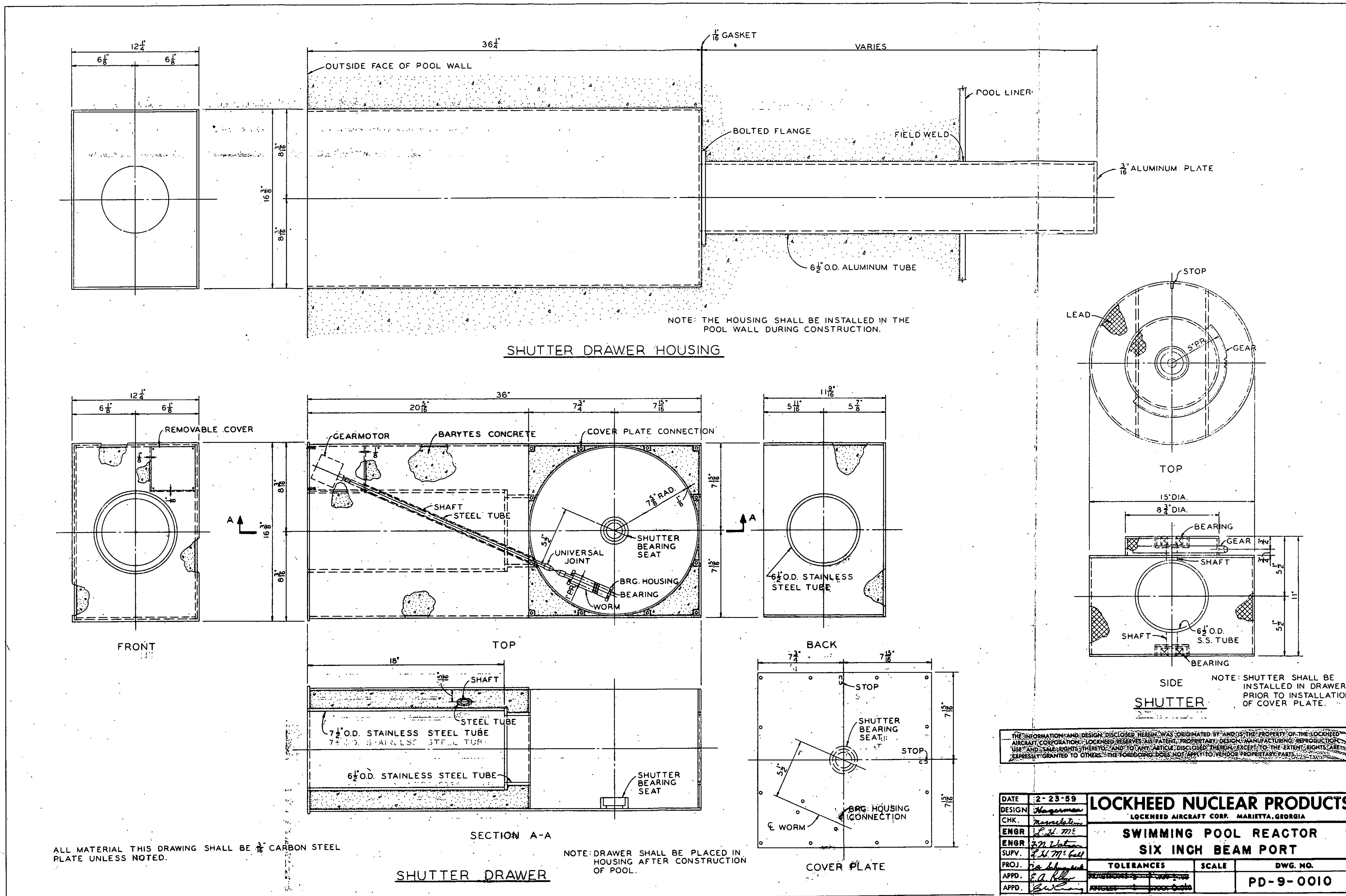
SWIMMING POOL REACTOR OPERATOR'S CONTROL PANEL

TOLERANCES	SCALE	DWG. NO.
		PD-9-0005



419 102

DATE	2-23-59	LOCKHEED NUCLEAR PRODUCTS LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA	
DESIGN			
CHK.			
ENGR.		FISSION PLATE ASSEMBLY	
ENGR.			
SUPV.			
PROJ.		TOLERANCES	SCALE	DWG. NO.
APPD.		FRACTIONS ± .XX ± .03		PD-9-0009
APPD.		ANGLES ± .XXX ± .010		


SHUTTER DRAWER HOUSING
OUTSIDE FACE OF POOL WALL
$\frac{1}{16}$" GASKET
VARIES
POOL LINER
BOLTED FLANGE
FIELD WELD
$\frac{3}{16}$" ALUMINUM PLATE
$6\frac{1}{2}$" O.D. ALUMINUM TUBE
NOTE: THE HOUSING SHALL BE INSTALLED IN THE POOL WALL DURING CONSTRUCTION.
REMOVABLE COVER
GEARMOTOR
BARYTES CONCRETE
COVER PLATE CONNECTION
SHAFT
STEEL TUBE
UNIVERSAL JOINT
SHUTTER BEARING SEAT
BRG. HOUSING
BEARING
WORM
$6\frac{1}{2}$" O.D. STAINLESS STEEL TUBE
FRONT
TOP
BACK
SECTION A-A
$7\frac{1}{2}$" O.D. STAINLESS STEEL TUBE
$6\frac{1}{2}$" O.D. STAINLESS STEEL TUBE
SHUTTER BEARING SEAT
STOP
BRG. HOUSING CONNECTION
℄ WORM
COVER PLATE
SHUTTER DRAWER
ALL MATERIAL THIS DRAWING SHALL BE $\frac{3}{16}$" CARBON STEEL PLATE UNLESS NOTED.
NOTE: DRAWER SHALL BE PLACED IN HOUSING AFTER CONSTRUCTION OF POOL.
LEAD
GEAR
5" PR.
15" DIA.
$8\frac{3}{4}$" DIA.
BEARING
SHAFT
$6\frac{1}{2}$" O.D. S.S. TUBE
SIDE
SHUTTER
NOTE: SHUTTER SHALL BE INSTALLED IN DRAWER PRIOR TO INSTALLATION OF COVER PLATE.
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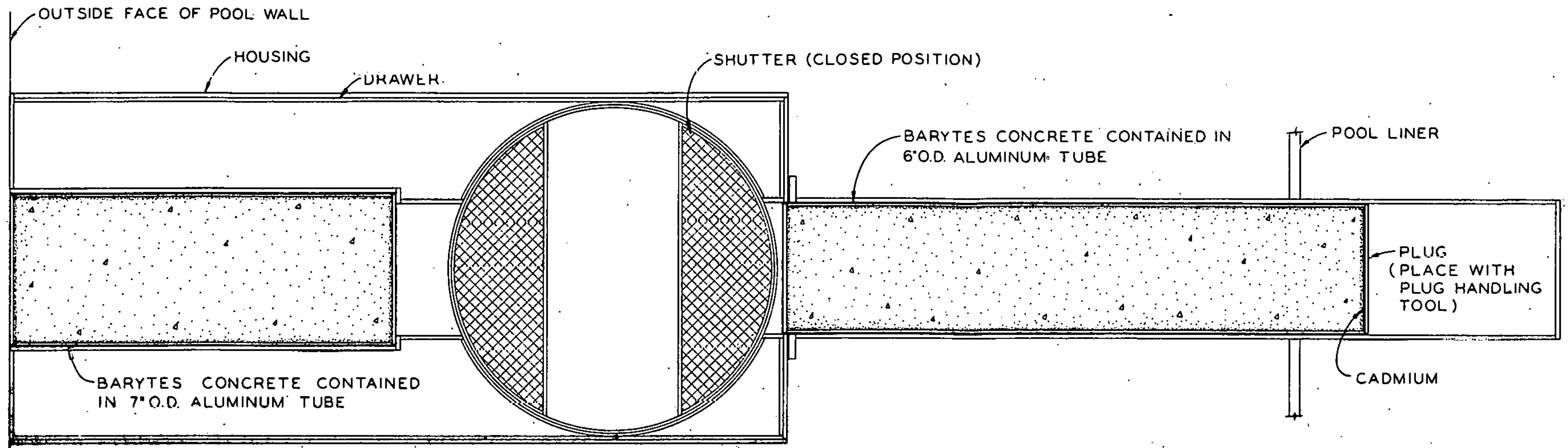
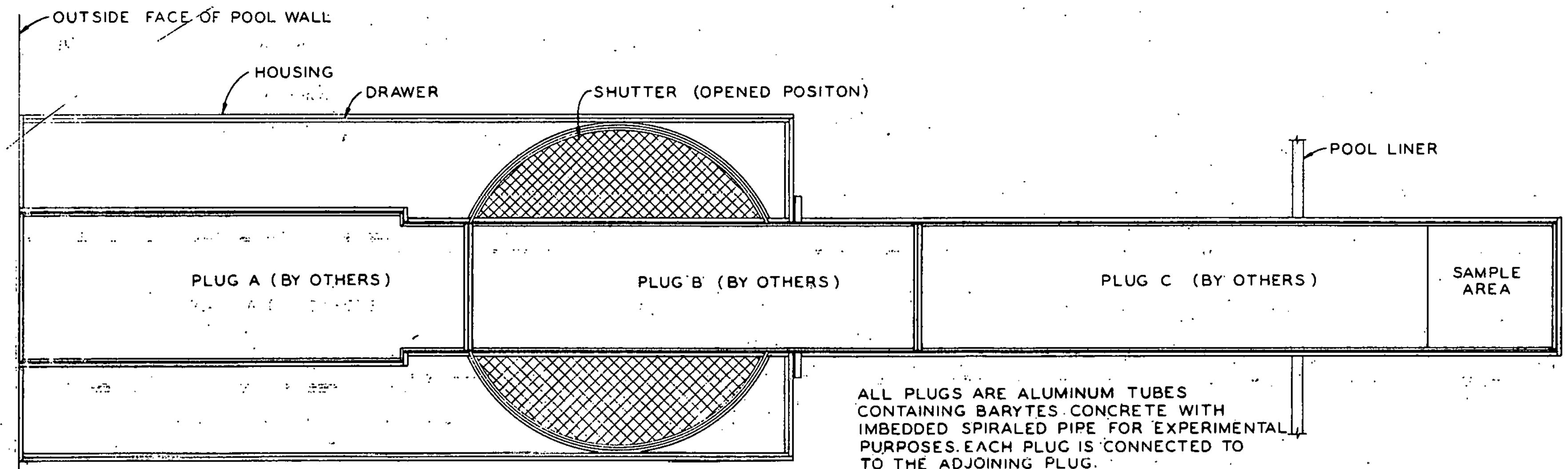
DATE 2-23-59
LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA
SWIMMING POOL REACTOR
SIX INCH BEAM PORT
TOLERANCES
SCALE
DWG. NO.
PD-9-0010

419 103



NOTE: 1. WITH REACTOR SHUT OFF, NO PLUGS ARE REQUIRED BUT SHUTTER IS TO BE AS SHOWN.
2. WITH REACTOR OPERATING BUT BEAM PORT NOT IN USE, PLUGS AND SHUTTER ARE TO BE AS SHOWN.

PLAN



NOTE: 1. WITH REACTOR OPERATING AND BEAM PORT IN USE, PLUGS AND SHUTTER TO BE AS SHOWN.
2. WITH EXPERIMENT COMPLETED AND REACTOR SHUT-OFF, PLUGS A AND B ARE REMOVED MANUALLY AND STORED FOR REUSE. CASK IS THEN PLACED IN FRONT OF PORT AND PLUG C WITHDRAWN INTO CASK WITH HANDLING TOOL. SHUTTER IS THEN CLOSED AND CASK REMOVED.

PLAN
(SUGGESTED USE OF SPECIALIZED PLUGS)

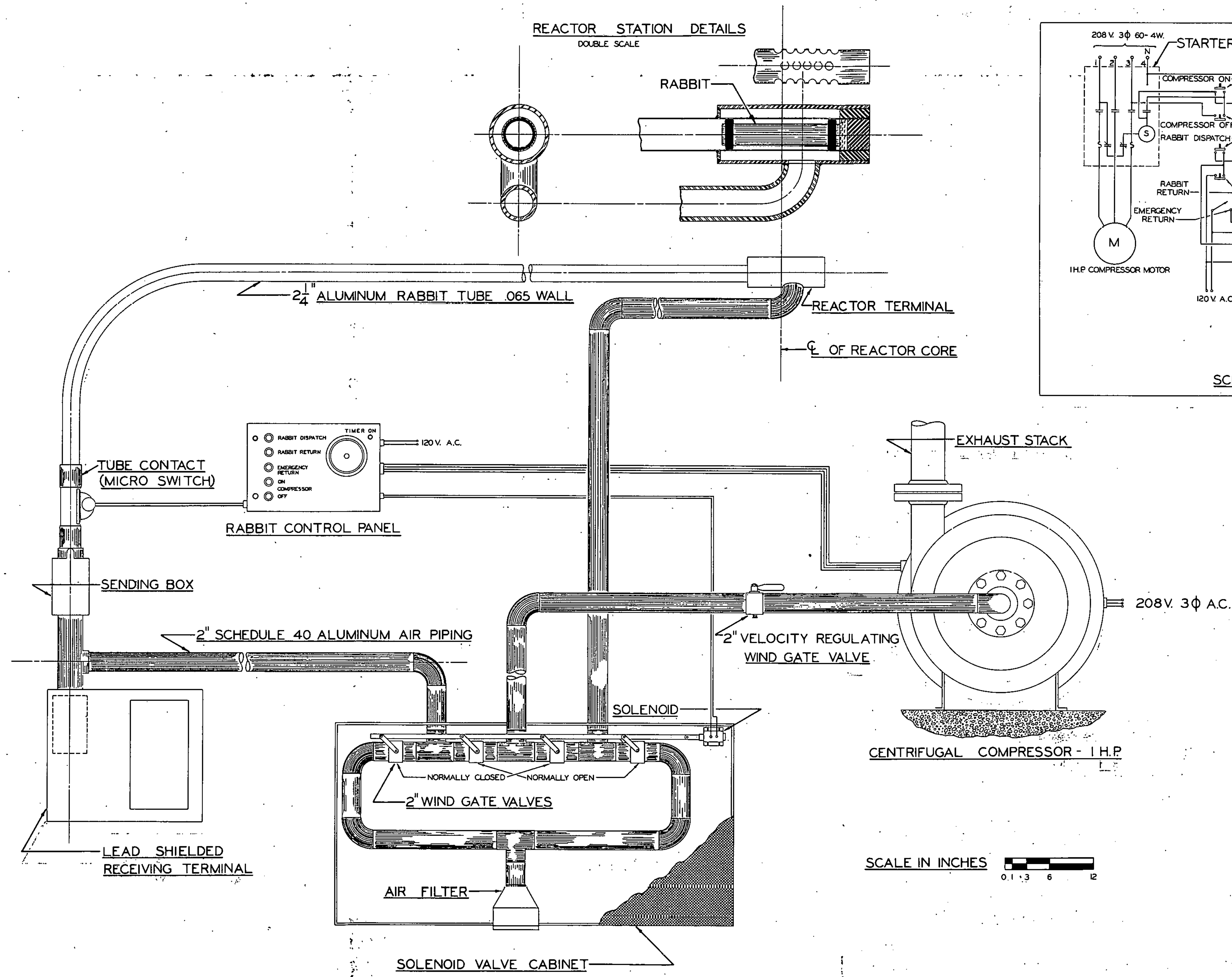
419 104

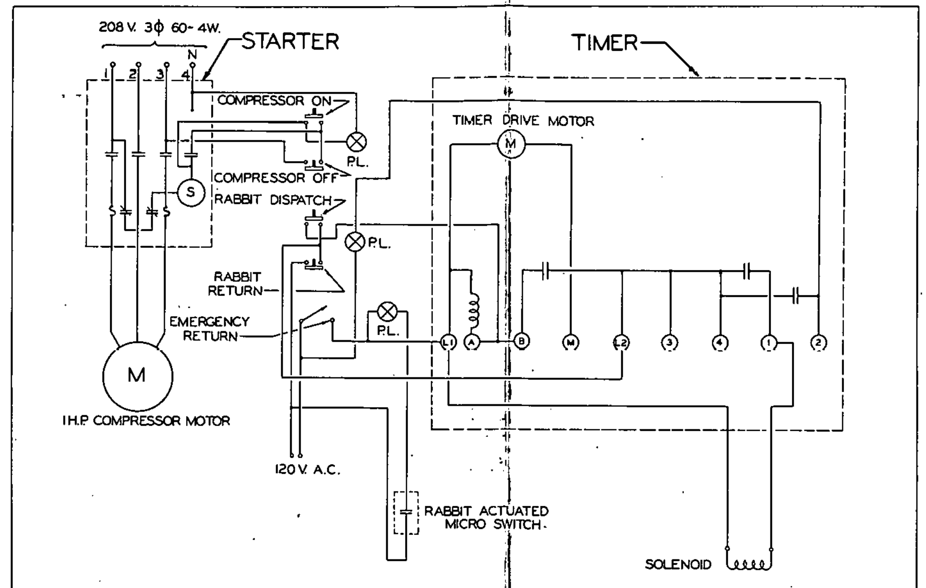
DATE	2-23-59
DESIGN	
CHK.	
ENGR	
ENGR	
SUPV.	
PROJ.	
APPD.	
APPD.	

LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

SWIMMING POOL REACTOR
SIX INCH BEAM PORT PLUGS

TOLERANCES	SCALE	DWG. NO.
		PD-9-0011





SCHEMATIC OF CONTROL SYSTEM

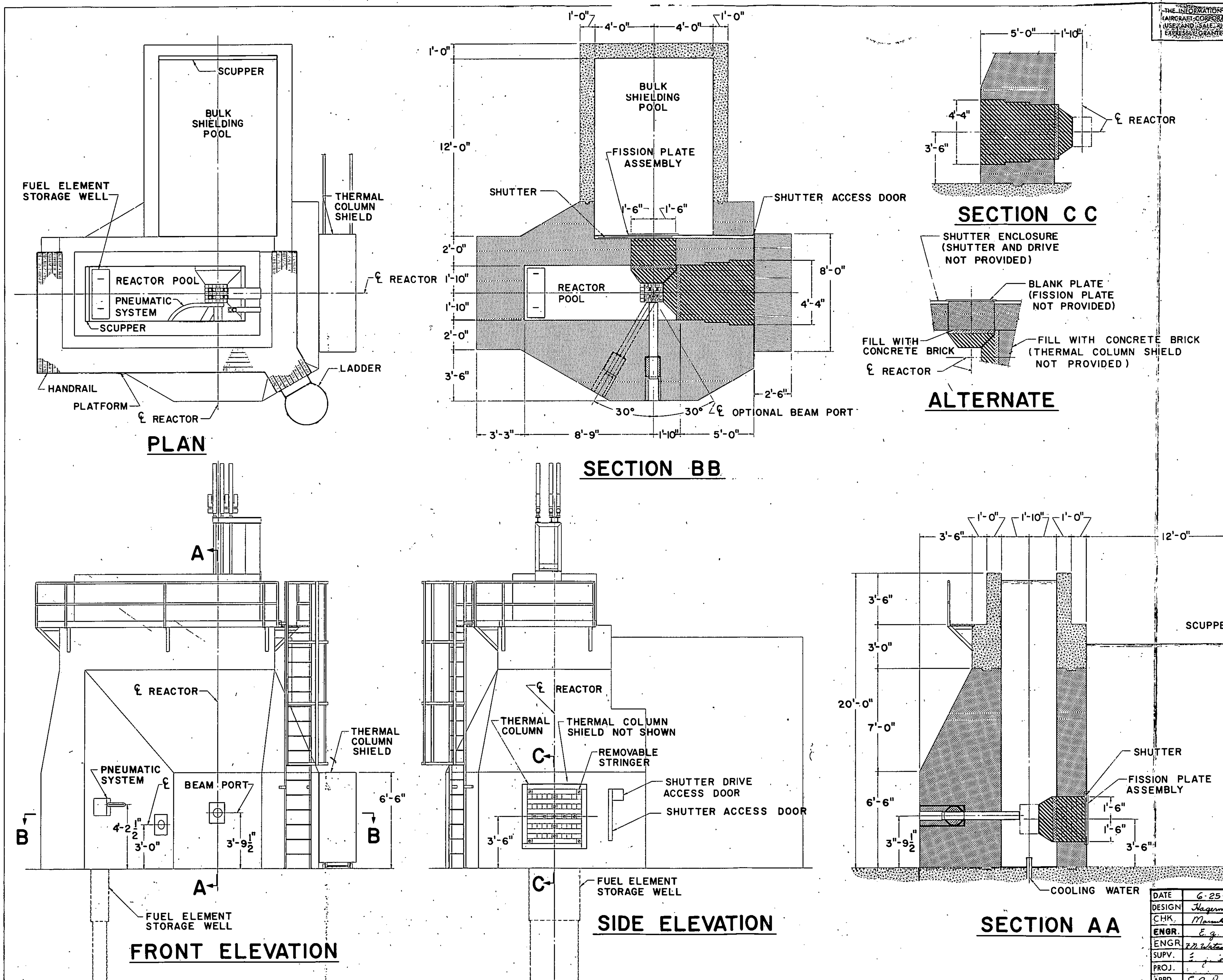
NOTES

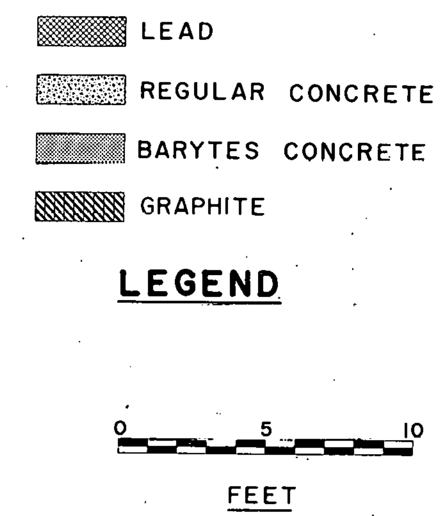
1. FOR RABBIT TUBE ROUTING SEE REACTOR POOL CONFIGURATION DRAWING.
2. SEALED FLANGES ARE USED AT ALL REACTOR POOL WALL PENETRATIONS.
3. VALVE GATE ARRANGEMENT EJECTS RABBIT TO RECEIVING TERMINAL IN THE EVENT OF SOLENOID FAILURE.

SCALE IN INCHES 0 1 3 6 12

419 105

DATE	2-23-59	**LOCKHEED NUCLEAR PRODUCTS** LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA
DESIGN	H.A. Llewellyn	
CHK.	[illegible]	SWIMMING POOL REACTOR
ENGR.	L.L. Mc	RABBIT TUBE - MECHANICAL DETAILS
ENGR.	F.N. Watson	
SUPV.	L.L. McC[illegible]	
PROJ.	C.O. [illegible]	TOLERANCES: ~~FRACTIONS ±~~ ~~.XX ±.03~~ ~~ANGLES ±~~ ~~.XXX ±.010~~ — SCALE — DWG. NO. PD-9-0012
APPD.	E.A. [illegible]	
APPD.	[illegible]	





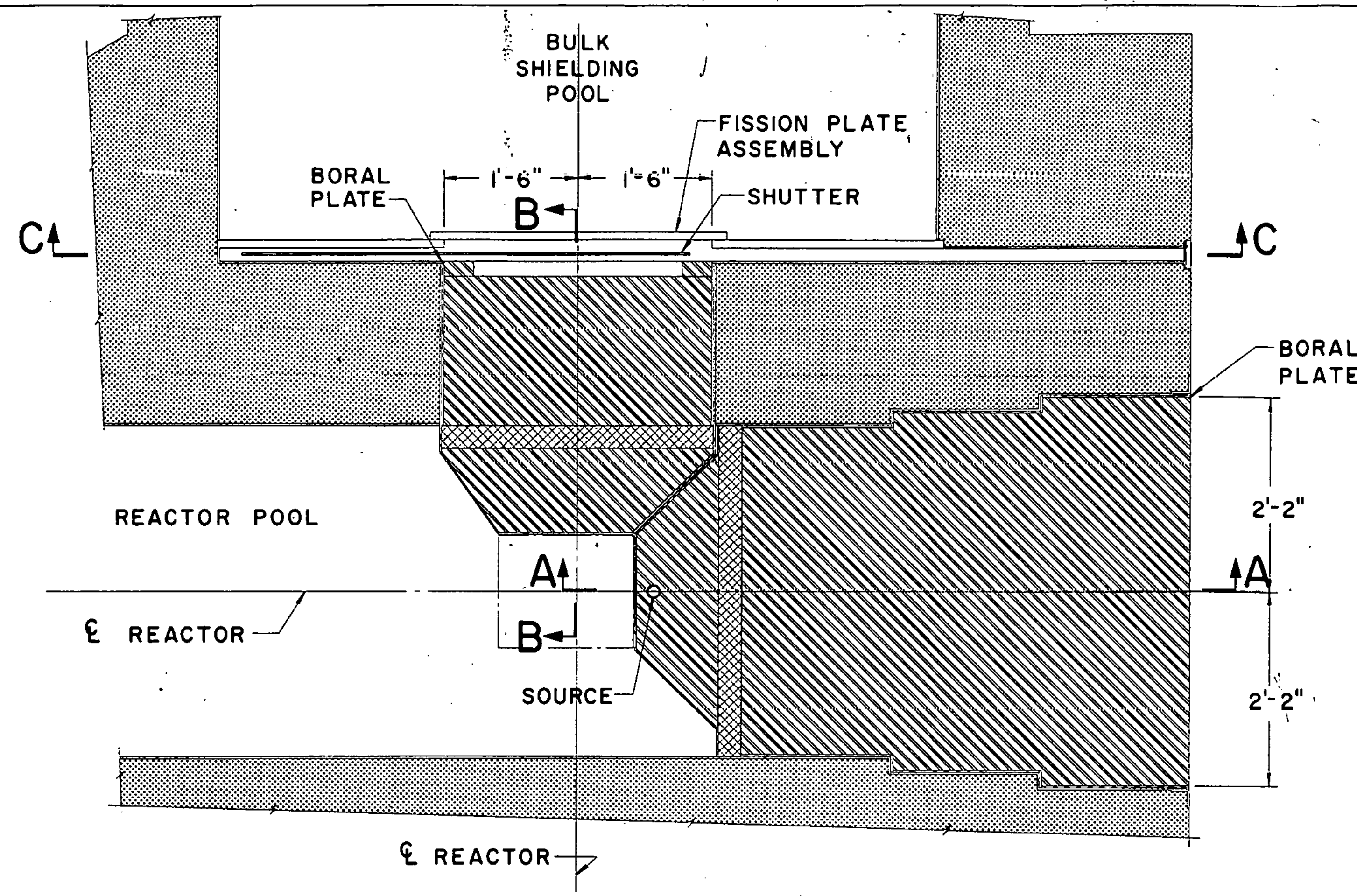
419 106

DATE	6-25-59
DESIGN	Hagerman
CHK.	Mannkin
ENGR.	E. g. E.
ENGR.	F. N. Watson
SUPV.	E. J. ...
PROJ.	
APPD.	E. A. Roller
APPD.	S. W. ...

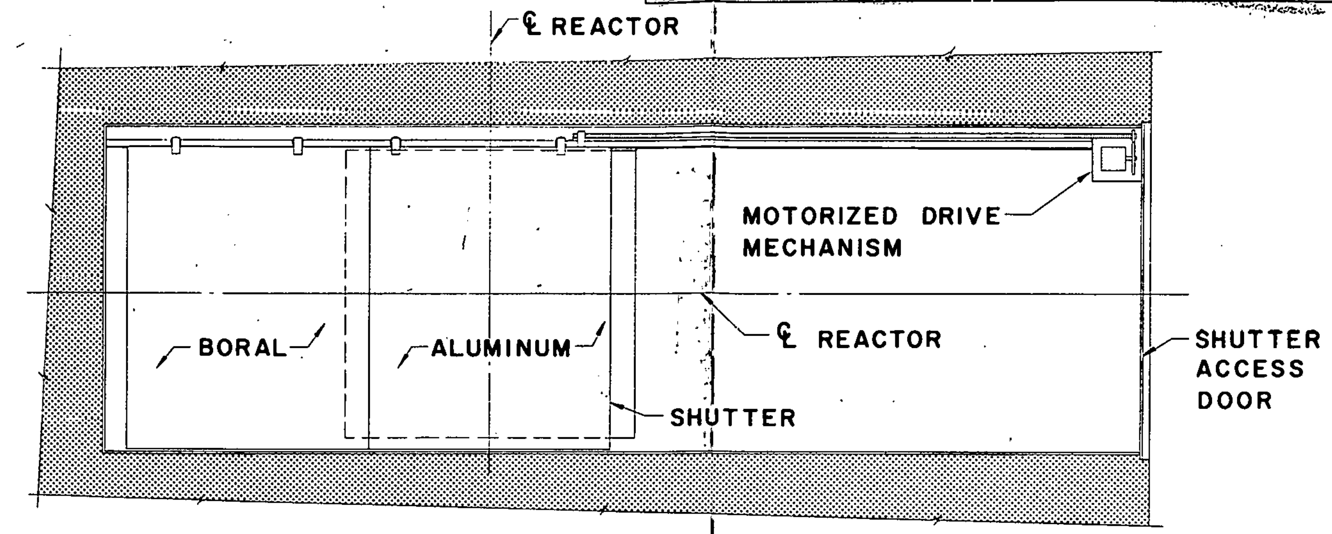
LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

SWIMMING POOL REACTOR
POOL CONFIGURATION

TOLERANCES	SCALE	DWG. NO.
		PD-10-0007



PLAN



SECTION C-C

LEAD

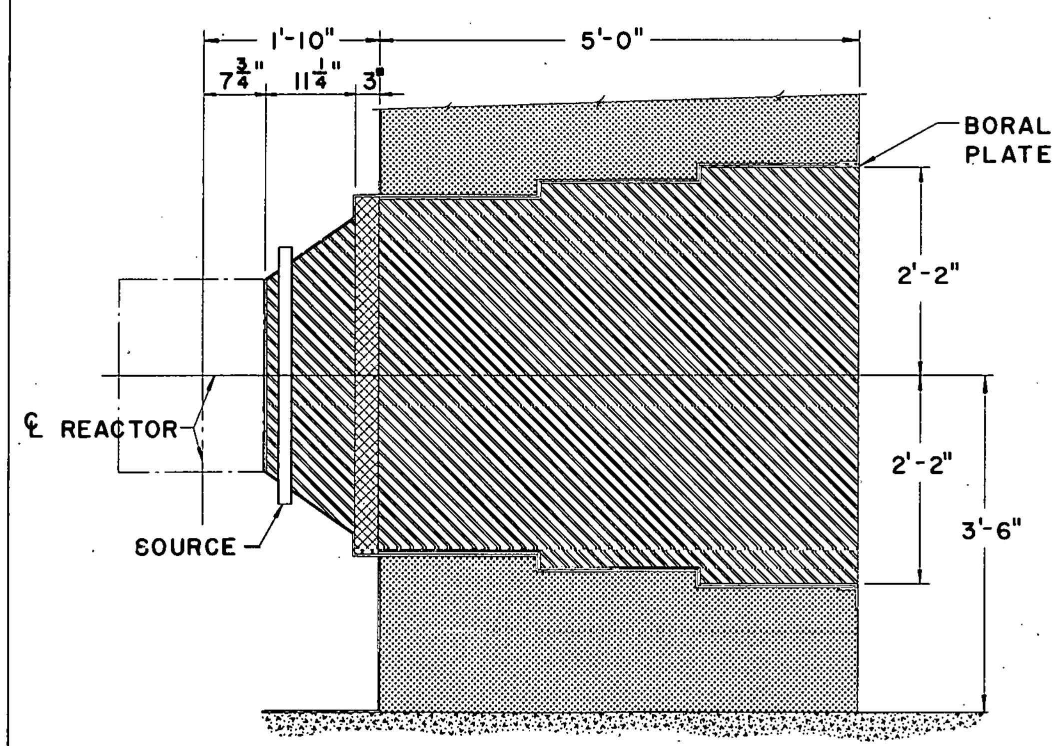
REGULAR CONCRETE

BARYTES CONCRETE

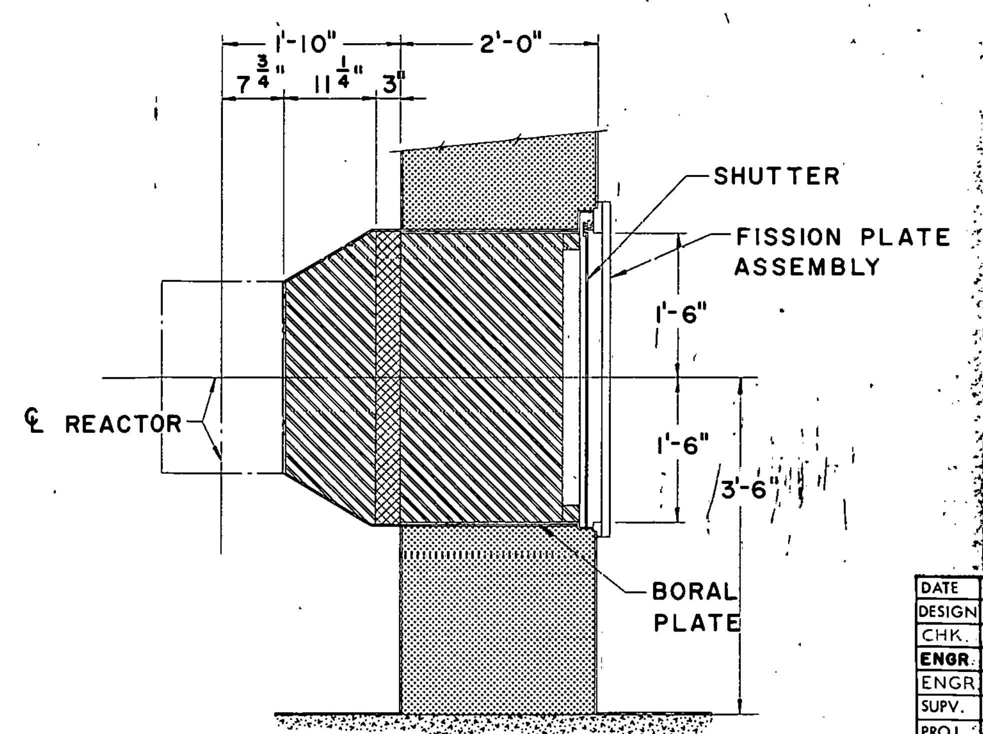
GRAPHITE

LEGEND

FEET

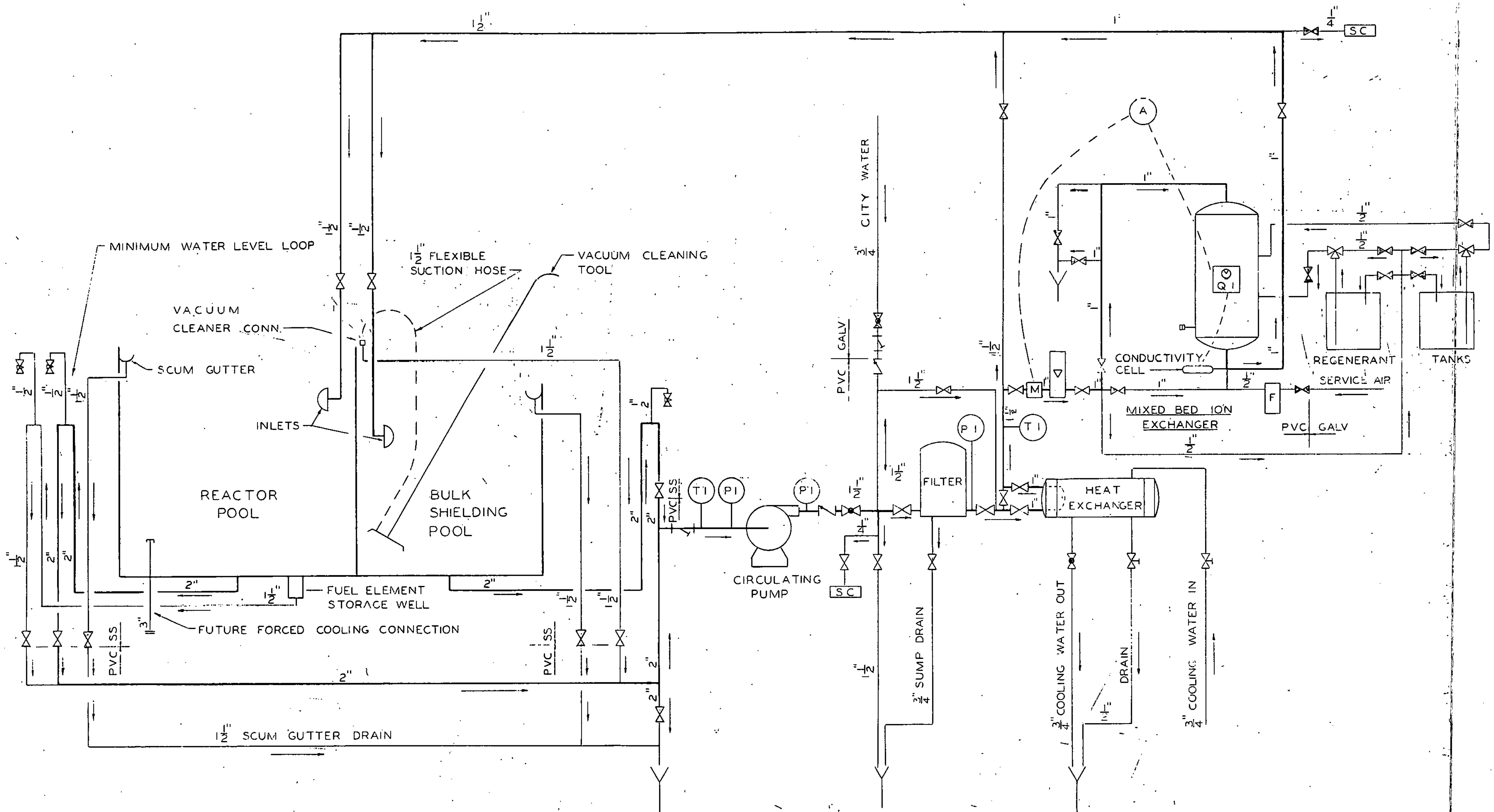


SECTION A-A



SECTION B-B

DATE	6-25-59	**LOCKHEED NUCLEAR PRODUCTS** LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA
DESIGN	Hageman	
CHK.	[illegible]	
ENGR.	E. g. [illegible]	SWIMMING POOL REACTOR
ENGR.	F. N. Watson	THERMAL COLUMN AND SHUTTER DETAILS
SUPV.	[illegible]	
PROJ.		TOLERANCES / SCALE / DWG. NO.
APPD.	E. A. Pollar	
APPD.	B W Craig	PD-10-0008



LEGEND

- BALL VALVE - NORMALLY OPEN
- BALL VALVE - NORMALLY CLOSED
- GATE VALVE
- GLOBE VALVE
- CHECK VALVE
- STRAINER
- EJECTOR
- (PI) PRESSURE INDICATOR
- (TI) TEMPERATURE INDICATOR
- (A) BELL ALARM
- [M] WATER METER
- [F] AIR FILTER
- [SC] SAMPLE CONNECTION
- FLOOR DRAIN (TO SEWER)
- FLOW INDICATOR
- (QI) QUALITY INDICATOR
- PLUG COCK WITH LOCK
- PIPE CAP

DATE	6-24-59
DESIGN	C. B. Waynick
CHK.	E. M. Burnett
ENGR	L. H. McC
ENGR	W. J. Judd
SUPV.	L. H. McCall
PROJ.	C. A. Schaufeh
APPD.	E. A. Bolly
APPD.	[illegible]

LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

SWIMMING POOL REACTOR PROCESS FLOW DIAGRAM

TOLERANCES		SCALE	DWG. NO.
FRACTIONS ±	.XX ± .03		PD-10-0013
ANGLES ±	.XXX ± .010		

Formally filed 11/2/59.