DOCKET NO. 50-39

CURTISS-WRIGHT CORPORATION
RESEARCH DIVISION

QUEHANNA, PENNSYLVANIA

AMHERST 3-4711

January 16, 1958

U. S. Atomic Energy Commission
1901 Constitution Avenue
Washington 25, D. C.

Attn: Mr. H. L. Price, Director
Division of Licensing and Regulations

Gentlemen:

Transmitted herewith, 6 copies of Curtiss-Wright, Research Division Report CWR-400-2, "Hazards Evaluation Report Curtiss-Wright Research Reactor."

Very truly yours,

CURTISS-WRIGHT CORPORATION
RESEARCH DIVISION

Richard S. Greene

Richard S. Greene
Chief, Technical Services Division
Nuclear Power Department

RSG/jz

DOCKETED USAEC Division of Civilian Application JAN 17 1958 Docket Officer

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DOCKET NO. 50-39

HAZARDS EVALUATION REPORT

CURTISS-WRIGHT RESEARCH REACTOR

December 16, 1957

CURTISS WRIGHT

CURTISS-WRIGHT CORPORATION
RESEARCH DIVISION
CLIFTON, NEW JERSEY, U.S.A.
QUEHANNA, PA., U.S.A.

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CURTISS WRIGHT

Docketed 1/17/58 by R. J. Frederick

CURTISS-WRIGHT CORPORATION
RESEARCH DIVISION
CLIFTON, NEW JERSEY
QUEHANNA, PENNSYLVANIA

CW-400-2

HAZARDS EVALUATION REPORT

CURTISS-WRIGHT RESEARCH REACTOR

December 16, 1957

Prepared by

RESEARCH REACTOR and PHYSICS DIVISIONS
NUCLEAR POWER DEPARTMENT

Approved by:

B. J. Mezger
Manager
Nuclear Power Department

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INTRODUCTION

This report has been prepared for presentation to the Advisory Committee on Reactor Safeguards and the Reactor Hazards Evaluation Staff of the Atomic Energy Commission in support of an application being made by Curtiss-Wright Corporation for a license to operate a nuclear reactor.

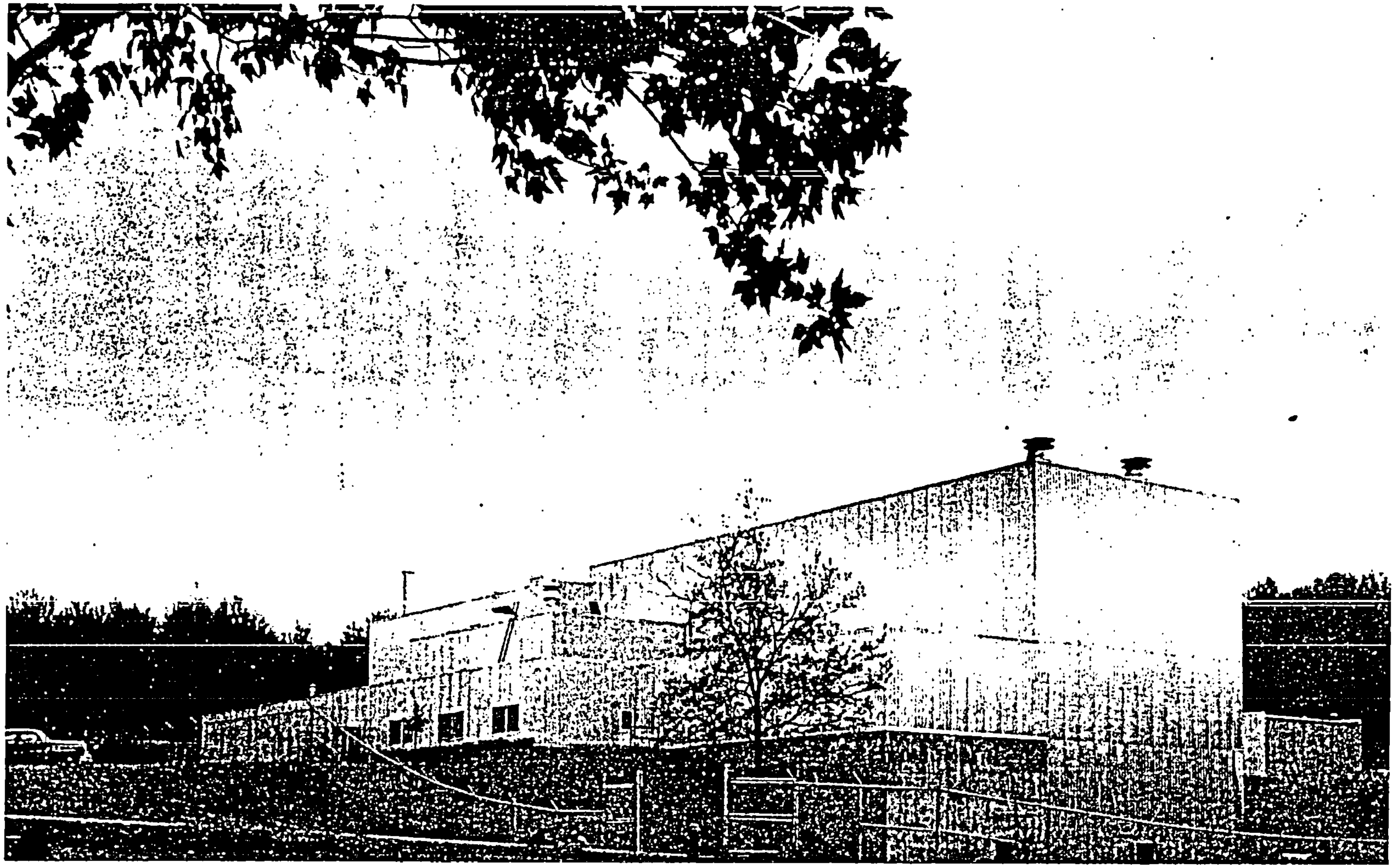
The reactor is of the light water moderated and cooled, solid fuel type often referred to as a swimming pool reactor. The aluminum-uranium alloy fuel elements are similar in construction to those used in the Bulk Shielding Reactor (BSR) and the Materials Testing Reactor (MTR). Cooling water is circulated through the core by free convection at power levels up to 100 kw. From 100 kw to full rated capacity of 1000 kw, cooling is by forced circulation of water. Most of the necessary shielding is supplied by the water.

The reactor is located at the Curtiss-Wright Research and Development Center at Quehanna, Pennsylvania. This center is located on an 80 sq mi tract of land surrounded by a low population density area in the north central portion of the state. One of several research facilities at this site is the Research Reactor and Radioactive Materials Laboratory, the exterior of which is shown in Figure 1. This building houses the reactor under discussion. The Ralph M. Parsons Company, Los Angeles, was the Architect-Engineer.

The experimental program for which the reactor is to be utilized will include shielding studies, reactor component and instrument development, investigations of radiation damage, and neutron physics. In addition, the reactor will be used for radioisotope production, activation analysis and training purposes.

Clearly, a neutron flux higher than that obtainable from a swimming pool reactor at 1000 kw would be desirable for much of this work. From the standpoint of experimental desirability, the proposed conservative operating level of 1000 kw, therefore, represents a sacrifice. When more extensive operating experience has been gained, and the operating characteristics of the reactor are well established, it will be possible to go to higher power. It is felt that at a 1000 kw rating, the devices reduce the possibility of an incident of minor, let alone major, proportions to an exceedingly small value.

In Sections I-B and I-C, supporting facilities and the site are described in somewhat more detail. However, since this particular reactor type has become relatively well standardized, many details have been purposely omitted from the description. The matter of operational procedures and safety devices is considered in Section II.



Research Reactor and Radioactive Materials Laboratory

Figure 1

The effects of human and instrument errors, particularly in combination with the failure of safety devices are examined in Section III.

Finally, in the last section the possible consequences of the release of radioactivity to the general environment is considered without specific reference to the mode of escape.

I. THE REACTOR; ITS SUPPORTING FACILITIES AND SITE

A. Reactor

Some of the more important characteristics of the reactor are tabulated in Table I.

TABLE I

Characteristics of the Curtiss-Wright Research Reactor

Type	swimming pool (modified BSR-type)
Core	heterogeneous - uranium, aluminum, water
Al/H_2O volume ratio	0.37
Moderator	light water
Reflector	light water, graphite or beryllium oxide
Coolant	light water, free convection flow at 0 to 100 kw, forced circulation above 100 kw.
Biological shield	light water, normal and high density concrete
Critical mass	2.7 to 3.6 kg U-235 depending on configuration (water reflected)
Power level	up to 1000 kw
Average thermal flux	8×10^{12} n/cm^2/sec at 1000 kw with an H_2O reflector

Additional information regarding the Curtiss-Wright Research Reactor is described under the appropriate headings which follow.

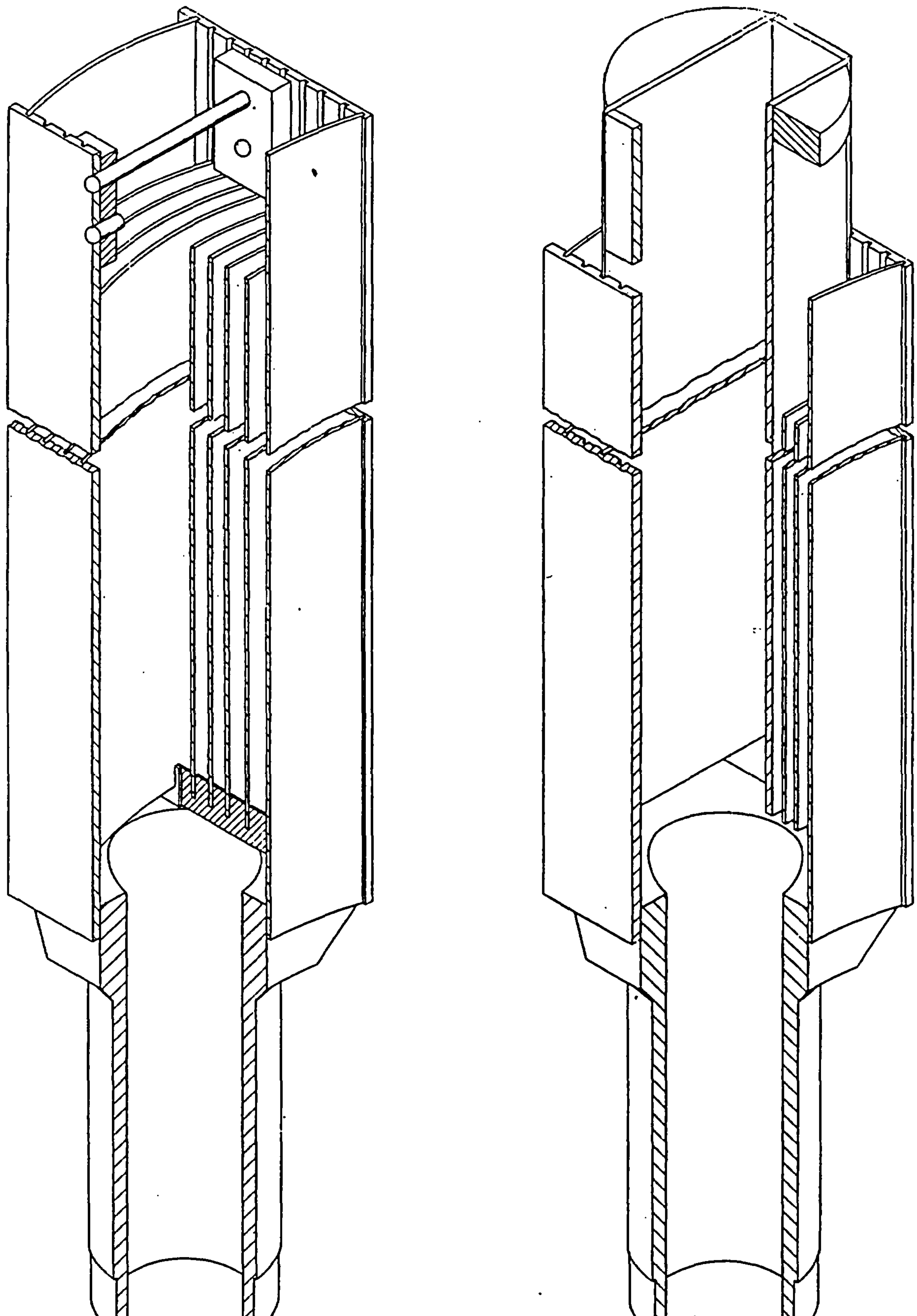
1. Fuel Elements

The reactor core will be made up of eighteen or more Curtiss-Wright designed, MTR-type fuel elements. The standard elements will contain ten fuel bearing plates. Each plate is a sandwich consisting of an 0.020 in. thick layer of aluminum-uranium alloy covered on each side by an 0.020 in. thick layer of aluminum. This thickness of aluminum is sufficient to contain all fission fragments under normal circumstances. The uranium is enriched to greater than 90% in the 235 isotope. The alloy layer will measure approximately 2.5 in. in width and will contain 17 grams of U-235. The finished plate will be approximately 3 in. wide, 24 in. long and 0.060 in. thick.

The fuel plates are fastened into groups of ten with aluminum side plates so that the finished element has an almost square cross-section measuring 3 in. by 3 in. At one end a male guide section of circular cross-section is attached, bringing the over-all length of an element to about 3 ft. This guide piece is inserted into a hole in the grid plate which supports the entire fuel element array or core.

The elements and grid plate are designed so that the fuel bearing plates are uniformly spaced throughout the core. Both ends of the elements are open so that cooling water may flow up or down between the fuel plates. The tolerances are set so that if all dimensions are off in the same direction there will be only a 20% reduction in coolant flow through any channel. The outer surfaces of the elements are cooled by water which passes through a funnel formed at the intersection of four elements and through an auxiliary coolant hole in the grid plate.

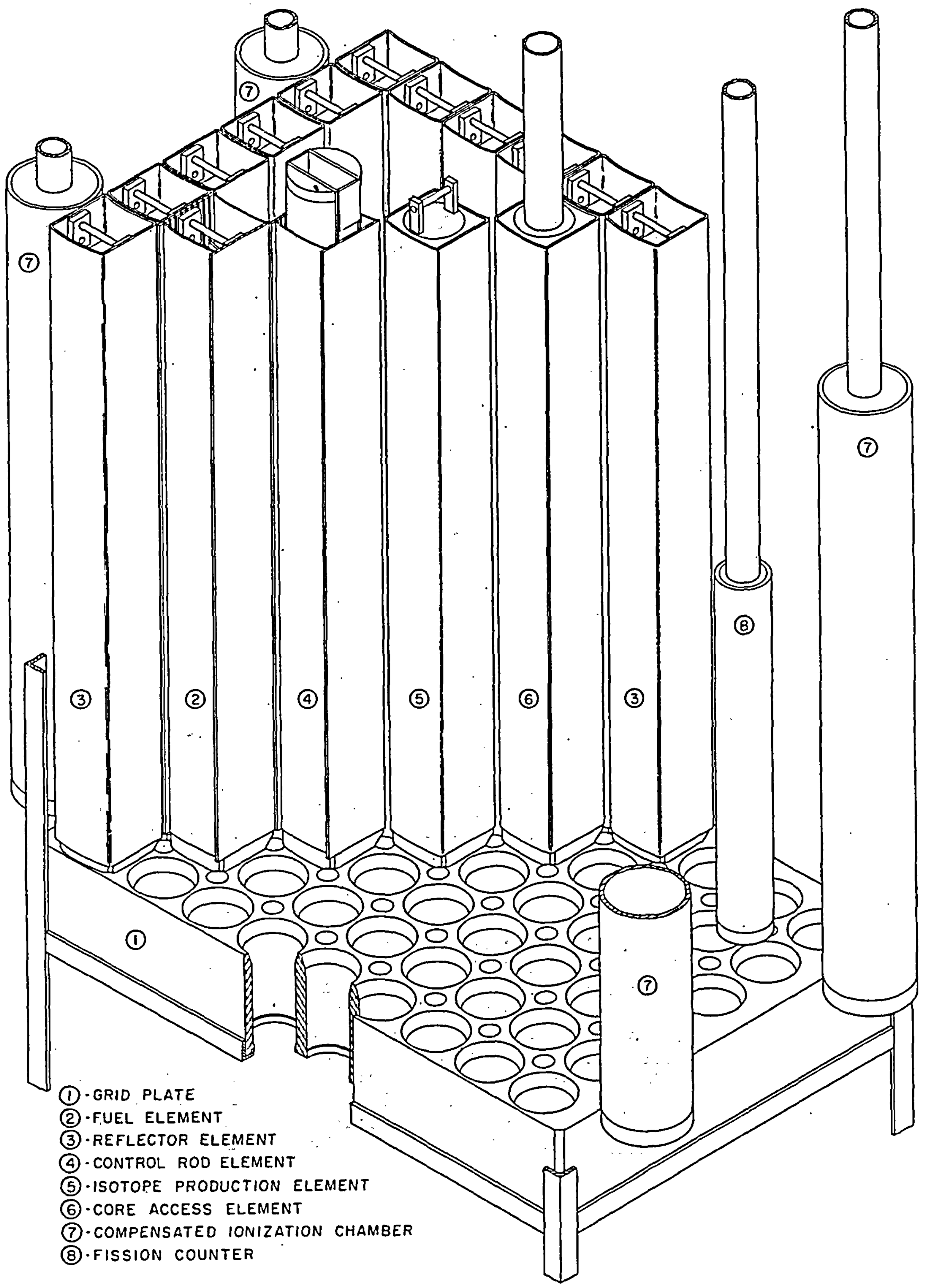
The standard fuel element designed by Curtiss-Wright is shown in Figure 2. The method of inserting the element in the grid plate is indicated by Figure 3. In addition to the standard elements, there will be partial elements and rod elements. The partial elements are identical to standard elements, except that fuel bearing plates are replaced by solid aluminum plates as per the schedule in Table II. The rod elements have the central four plates removed to accommodate the rod. The remainder of the plates are spaced so that they have .9, .9, .6, ...6, .9, .9 times the cooling area of a plate in a standard element. The reduction in cooling area is justified since a control or safety-shim rod will never be placed in the center of the reactor for full power operation. The design of the rod element is also shown in Figure 2.



FUEL ELEMENT

CONTROL ROD ELEMENT

Figure 2



REACTOR CORE

Figure 3

TABLE II

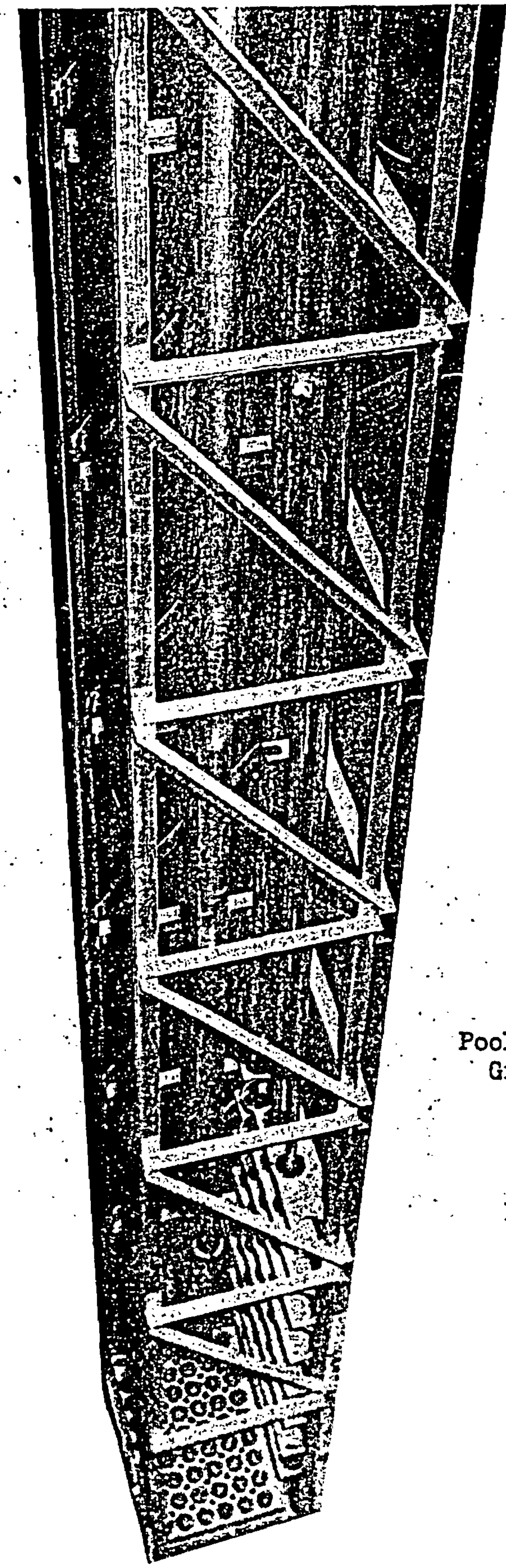
Initial Inventory of Fuel Elements for the
Curtiss-Wright Research Reactor

Type	Number of Fuel Elements	Grams of U-235 per Element	Total Grams U-235	Number of Fuel Bearing Plates per Element
Full	20	170	3400	10
Partial				
80%	2	136	272	8
60%	2	102	204	6
40%	2	68	136	4
20%	2	34	68	2
Rod element	4	102	408	6
	32		4488	

2. Supporting Structure

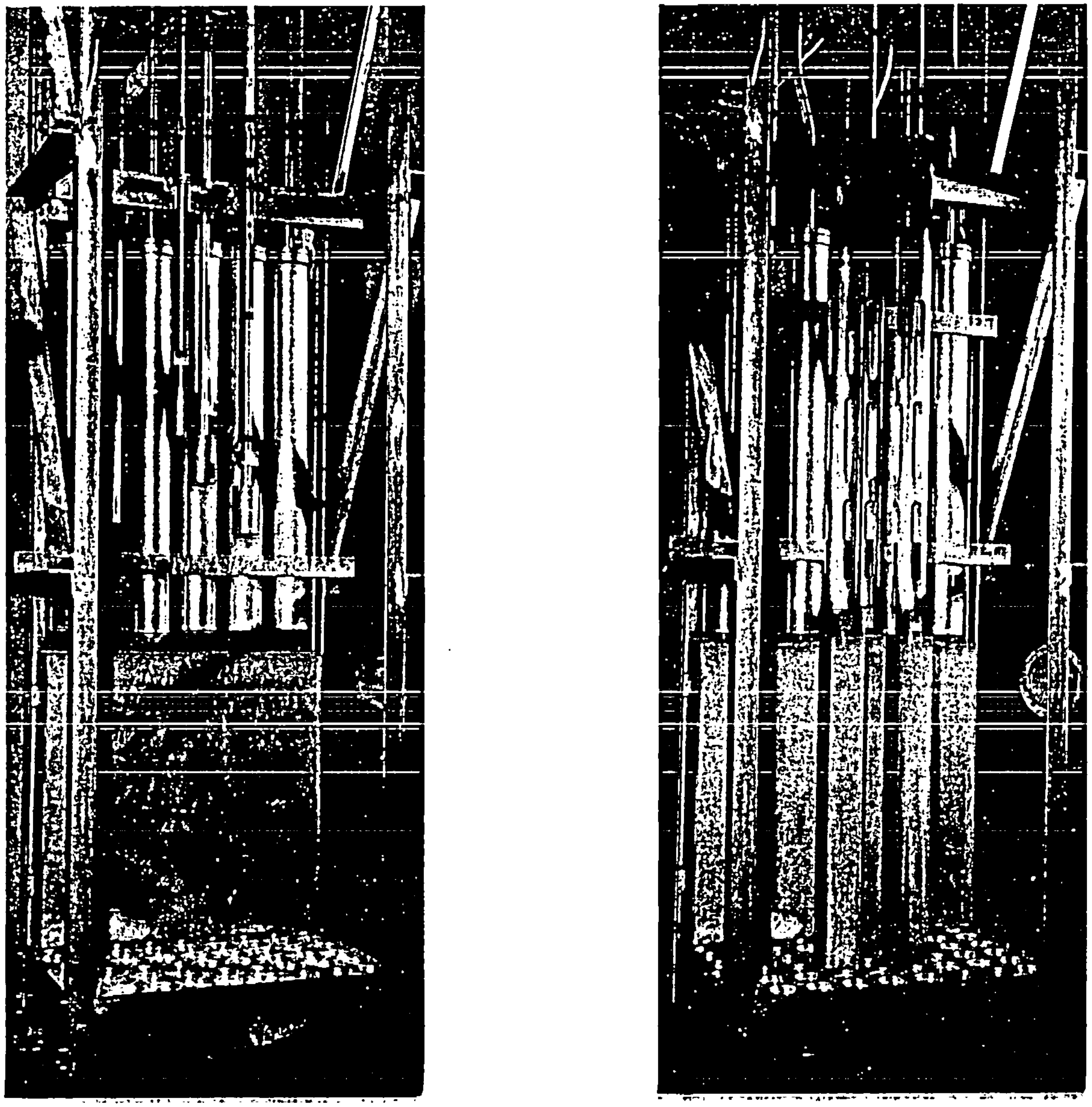
The fuel elements are supported by a grid plate which is attached to a rigid tower suspended from the underside of the reactor bridge. The grid plate and the tower have been fabricated from 2S aluminum to reduce the radiation hazard during the maintenance work. The general arrangement is shown in Figure 4. The grid plate has 54 holes, in a 9 x 6 array, into which the mating part of the fuel elements will fit. A picture of a grid plate from a similar reactor is shown unloaded and partially loaded in Figure 5. Additional holes are provided in the grid plate to allow water to circulate around as well as through the fuel elements.

The grid plate is positioned so that the fuel is at least 4 ft above the pool floor. Neither the grid plate nor the tower supporting it is capable of motion relative to the reactor bridge. The bridge has been designed to support a static load of 4,000 lbs at its center and 6,000 lbs on the cantilevered side which supports the control console and reactor operators. The bridge is free to roll along rails mounted on top of the walls surrounding the pool so that the core may be positioned at any point along the pool center line. The core position relative to underwater components, such as the beam hole tubes, can easily be determined by noting the position of a pointer indicator which moves along a fixed metric tape. The bridge may be locked in position by



Pool Tower and Grid Plate

Figure 4



VIEW OF GRID AND ACTIVE LATICE PARTIALLY LOADED. FROM THE BSR.

Figure 5

two manually operated brakes, one on each side of the pool. A combination of fixed and movable stops are mounted on the rails to prevent the core being damaged by moving against the pool walls, beam tubes or other obstacles.

3. Control Rods and Drives

The significant characteristics of the regulating and safety-shim rods and their drives are summarized in Table III below:

TABLE III

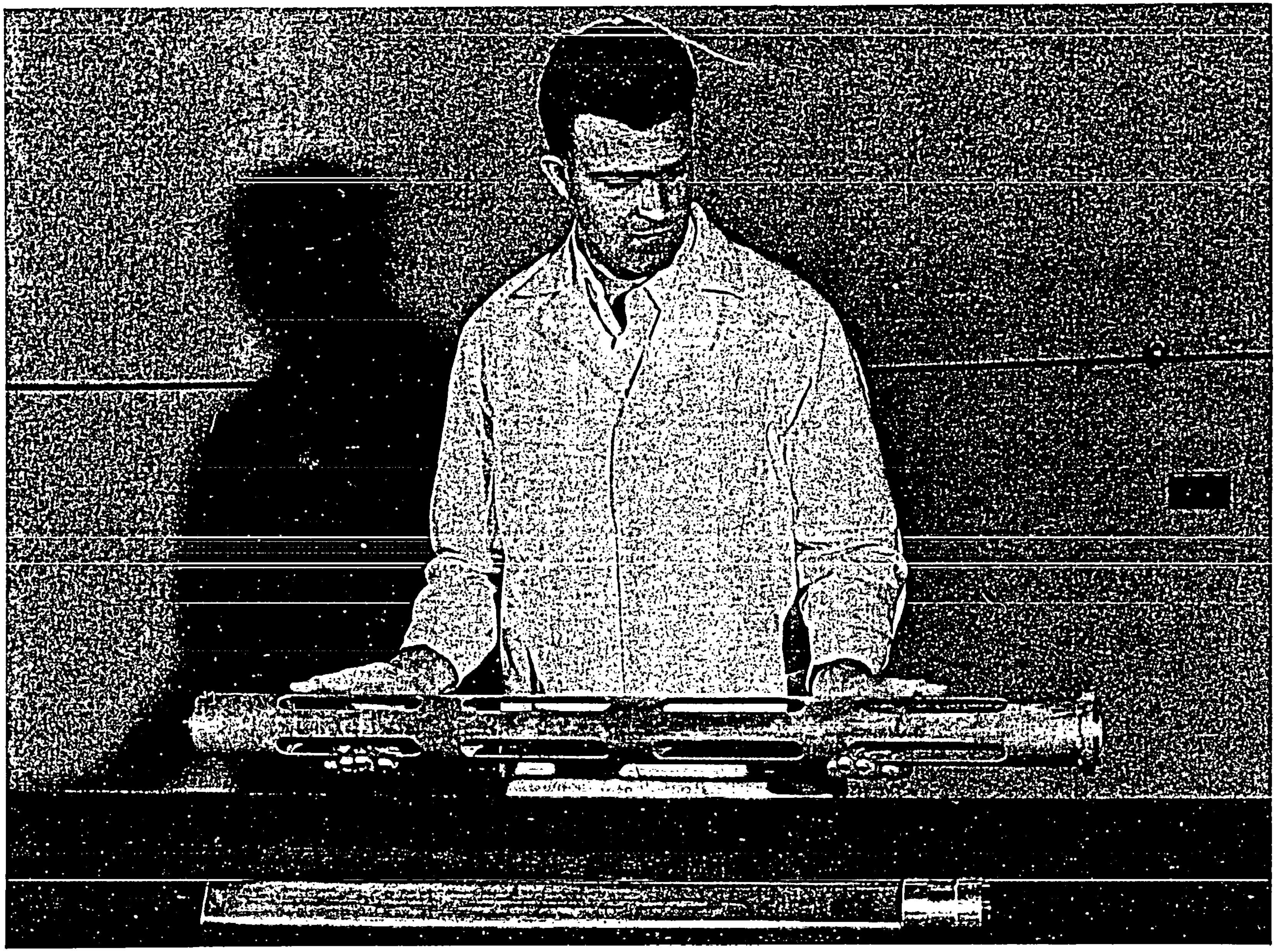
Characteristics of Control Rods for the Curtiss-Wright Research Reactor

Type	Material	Number Used	Reflector H_2O	Reflector Graphite	Rate of Drive in. min
Regulating	Stainless Steel	1	0.7%	1.2%	25
Safety-Shim	Cadmium-Boron Carbide	3	2.7% ea.	4.0% ea.	6 1/4

The regulating rod is a flattened 1/16 in. thick stainless steel tube with a 2.25 by 0.800 in. cross section. The safety-shim rods, with outside dimensions 2.25 by 0.875 in. are of laminated construction: a 0.065 in. stainless steel outer shell, a 0.032 in. cadmium inner shell, and the remaining cavity filled with boron carbide crystals to produce a minimum density of 1.5 gm/cm^3. Figure 6 shows a safety rod and its housing. The three safety-shim rods are magnetically coupled to their rod extensions. Upon power failure or receipt of a scram signal, the exciting current to the coupling magnets will be cut off and the rods will fall freely into the core. A piston attached to the safety rod passes through a close fitting cylindrical device when the safety rod nears its lower limit. The water forced upwards around the piston provides a hydraulic snubbing action which permits the safety rod to come to rest without damage.

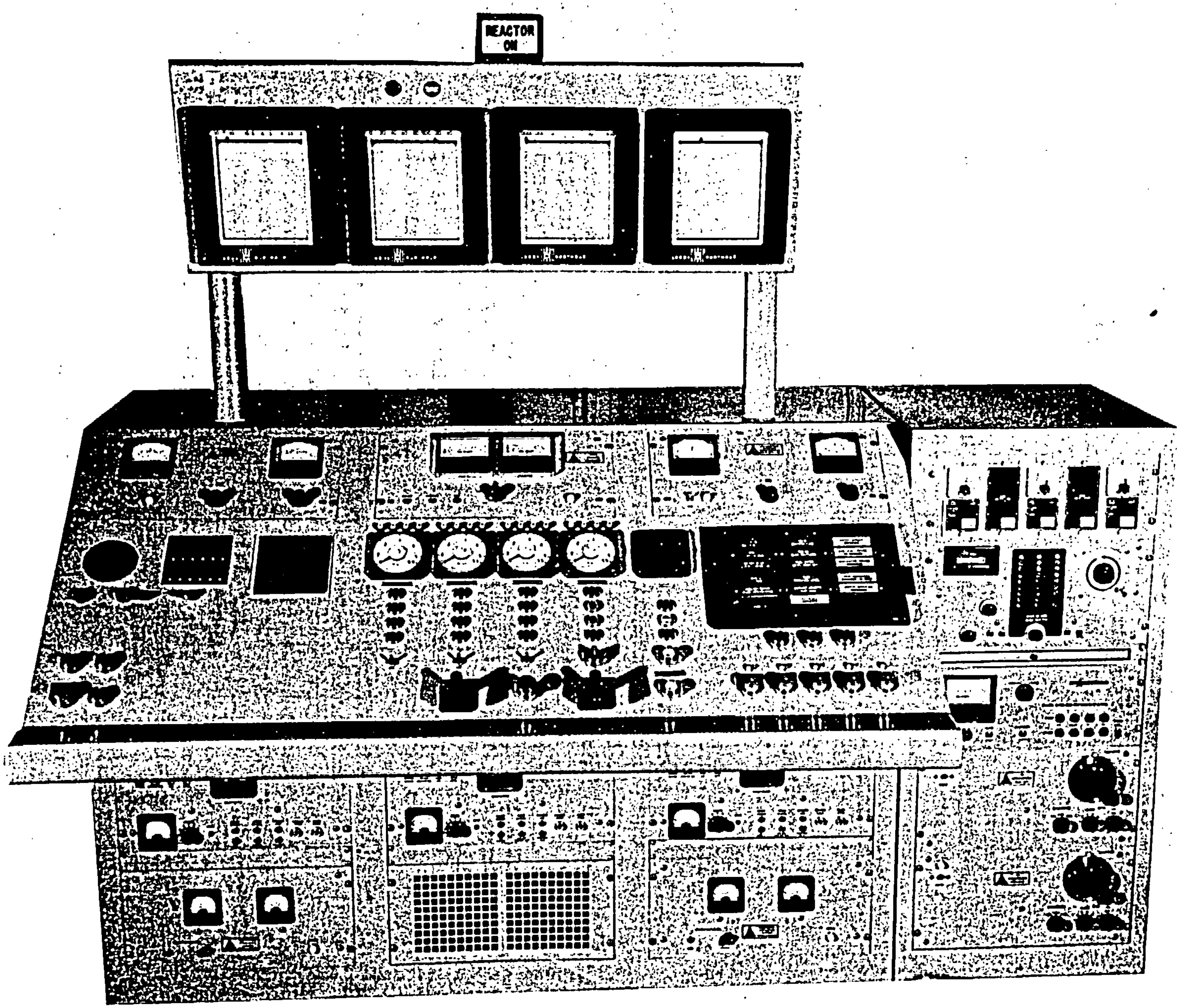
4. Control Console

The control console is located on the cantilevered portion of the reactor bridge facing the reactor structure so the operator can observe all work being performed. The console consists of two units which may be seen in Figure 7. The main unit is approximately 67 in. high with four miniature type recorders (Leeds and Northrup Type H) mounted 12 in. above it



SAFETY ROD AND HOUSING

Figure 6



REACTOR CONTROL CONSOLE

Figure 7

in such a manner that the operator, when seated at the console, may maintain both visual and aural communication with personnel working on the bridge. The prime consideration in the design of the main unit was centralization of the safety-shim and regulating switches, rod position indicators, scram button, and flux level and period indicators. With this arrangement, the entire control of the reactor is at the finger tips of the operator. Located in accessible but less centralized positions in the main unit is such equipment as composite safety amplifiers, annunciator panels, inter-com system, telephone, and power supplies for operating the gamma compensated ion chambers and associated equipment. An integrated power indicator is included in the console instrumentation to facilitate the recording of fuel exposure.

The auxiliary unit is approximately 22 in. wide and 47 in. high and is mounted at the right of the main unit. It contains the control panel for the radiation monitoring system, decade scaler and linear amplifier, and a test panel. The latter provides functions to facilitate checking and calibrating the Log N Period and Count Rate circuits.

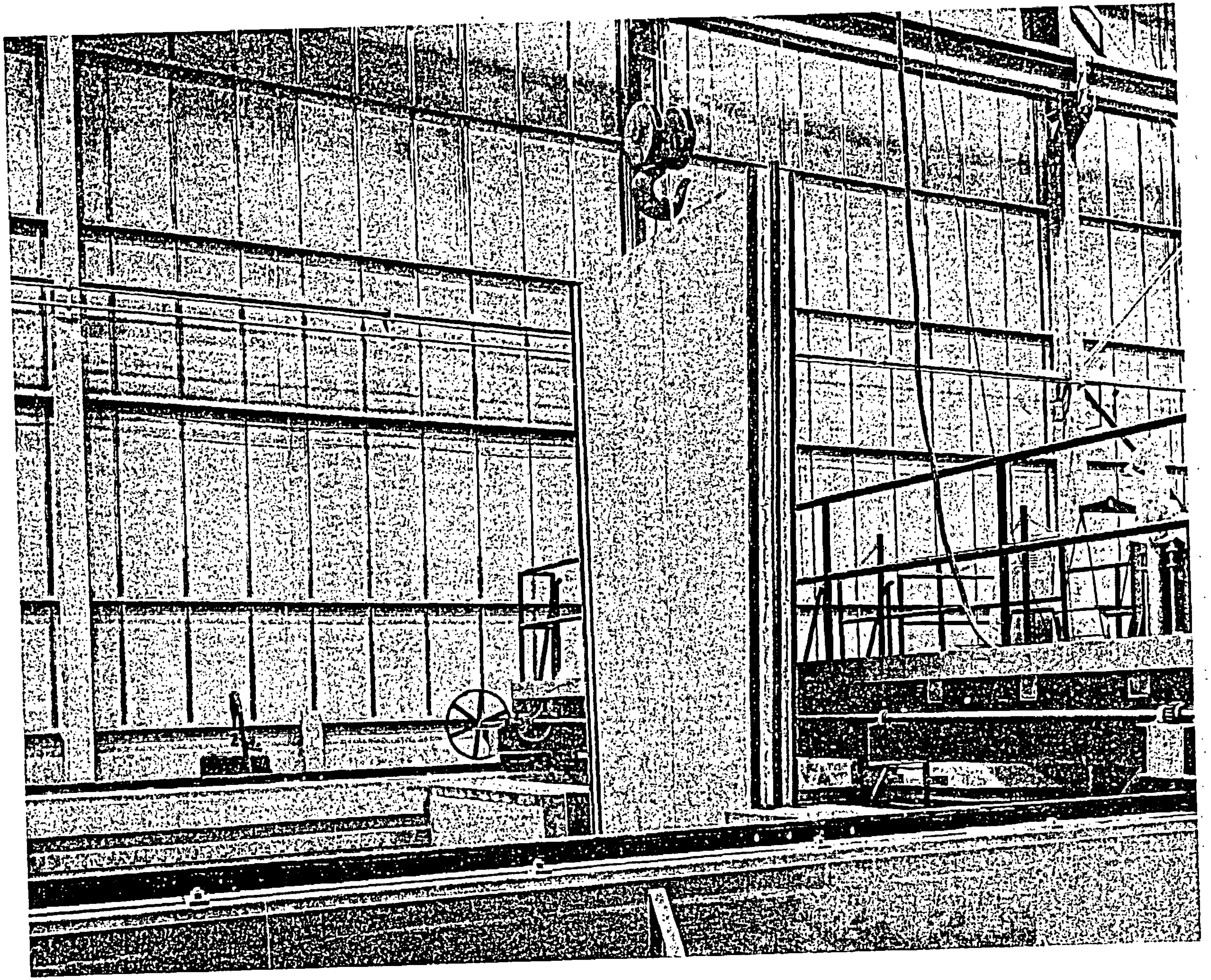
5. Reactor Pool

The pool measures 20 ft in width by 41 ft in over-all length including the three-sided extension at one end for the beam tubes. The water depth of 26 ft insures a minimum of 19 1/2 ft of water covering the core. The pool is separated into two sections by a concrete bulkhead. A 20 ft x 24 ft section will be used for bulk shielding studies, while the smaller portion will house the reactor during the experiments utilizing the beam holes, or pneumatic rabbit. The reactor may be moved between the two sections through a vertically sliding, watertight, aluminum gate, 5 ft x 22 ft long, located at the center of the concrete bulkhead. Figure 8 shows the gate partially removed. The larger pool's volume is about 93,600 gal, while the smaller contains 53,800 gal.

Pool walls are of ordinary reinforced concrete, 12 in. thick, except at the beam hole end where the thickness is increased to 5 ft, 6 in. In order to further attenuate the radiation from the reactor core before it reaches the beam room, the outer 4 ft of concrete contains ferrophosphorus aggregate. This high density insert reaches 9 ft 6 in. above the floor level in the beam room. The concrete sides and floor of the pool have received several coats of protective vinyl paint in order to prevent excessive leaching of minerals from the concrete into the demineralized water.

6. Cooling System

When operating at power levels in excess of 100 kw, convective cooling will not be relied upon. Instead, water will be pumped through the core



POOL GATE PARTIALLY REMOVED

Figure 8

from top to bottom. This will not only cool the fuel plates, but will prevent the radionitrogen formed by the O^{16} (n,p) N^{16} reaction from rising to the pool surface and raising the gamma level excessively.

A schematic flow diagram of the cooling system is shown in Figure 9. Water will be pumped through the core at 700 gpm and into an aluminum plenum chamber attached to the underside of the grid plate. Grid plate holes not containing fuel elements, reflector elements, or irradiation capsules will be plugged to prevent "short circuiting" the fuel channels. From the plenum, water will flow through a 6 in. aluminum tube capable of swiveling 360° about a vertical axis, and then through a flexible hose to the cooling water outlet at the side of the pool floor. The aluminum tube will be long enough to keep the flexible hose out of a radiation field sufficiently intense to damage it in a short time. The flexible coupling allows the core to be positioned as desired along its single axis of motion. The hose connects to a 6 in. pipe which rises to 18 ft above the pool floor before penetrating the wall. The coupling between the aluminum tube and the flexible section is shown in Figure 10.

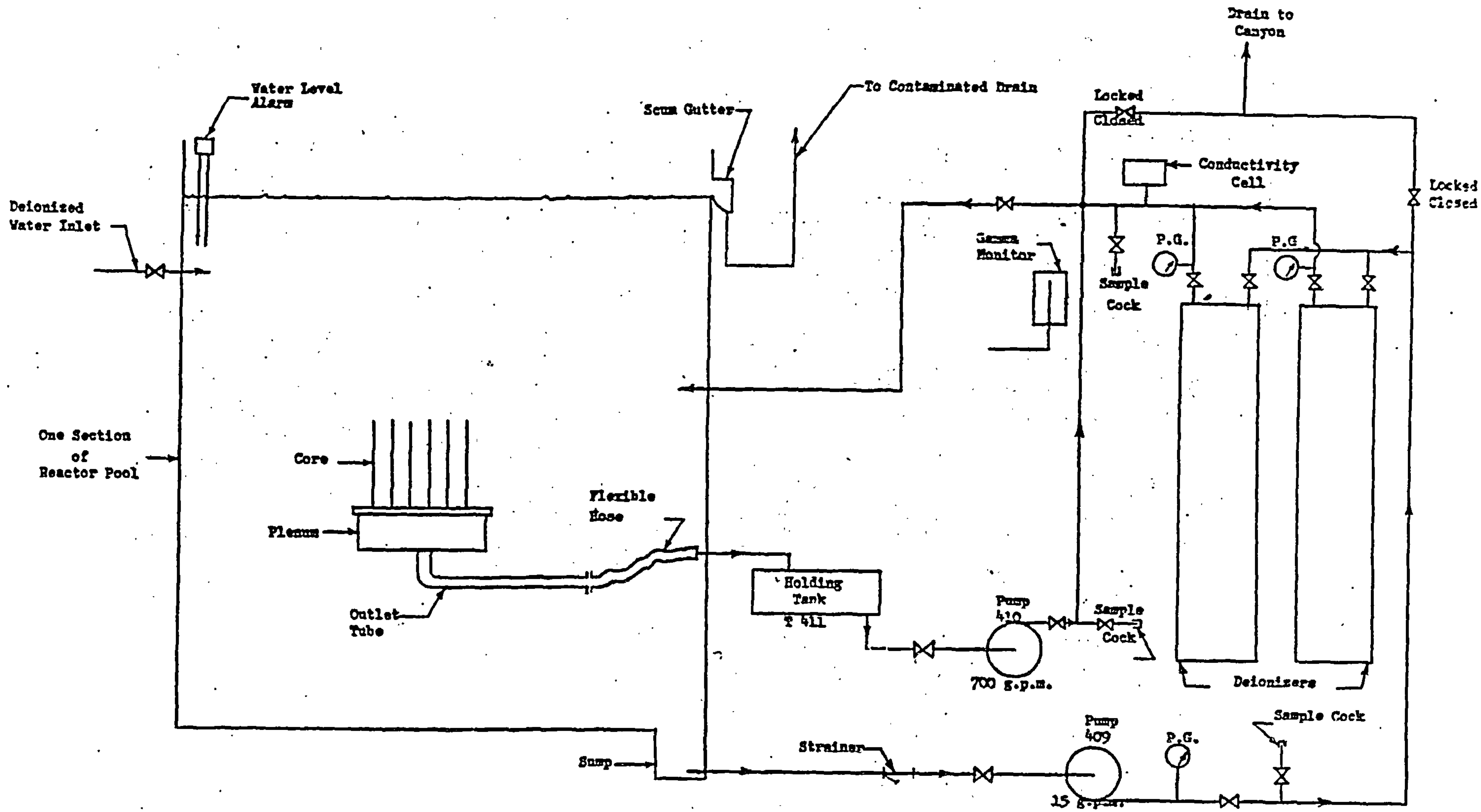
After leaving the pool proper, the 6 in. line leads underground to a 2000 gal hold-up tank. Transient time through this tank is sufficient to allow essentially all of the 7 sec N^{16} activity to decay. A line from the buried hold-up tank leads to the pump room which houses the circulating pump, and will eventually contain the heat exchanger. At the present time all of the equipment except the heat exchanger is installed. This will allow intermittent operation at 1000 kw by alleviating the N^{16} problem. However, continuous operation at this power will not be possible until the heat exchanger and associated cooling tower are installed. After this installation, the reactor will operate continuously with the bulk of the pool water at 90°F and about 10°F temperature rise through the core.

A large flapper valve is installed in the bottom of the plenum chamber. This will be manually operated from the reactor bridge. When the reactor is operated below 100 kw of power, this valve will be opened and cooling will be convective. When operating at power levels above 100 kw, forced cooling will be used and the flapper valve closed. Low flow through the cooling lines will be annunciated at the control panel.

B. Supporting Facilities

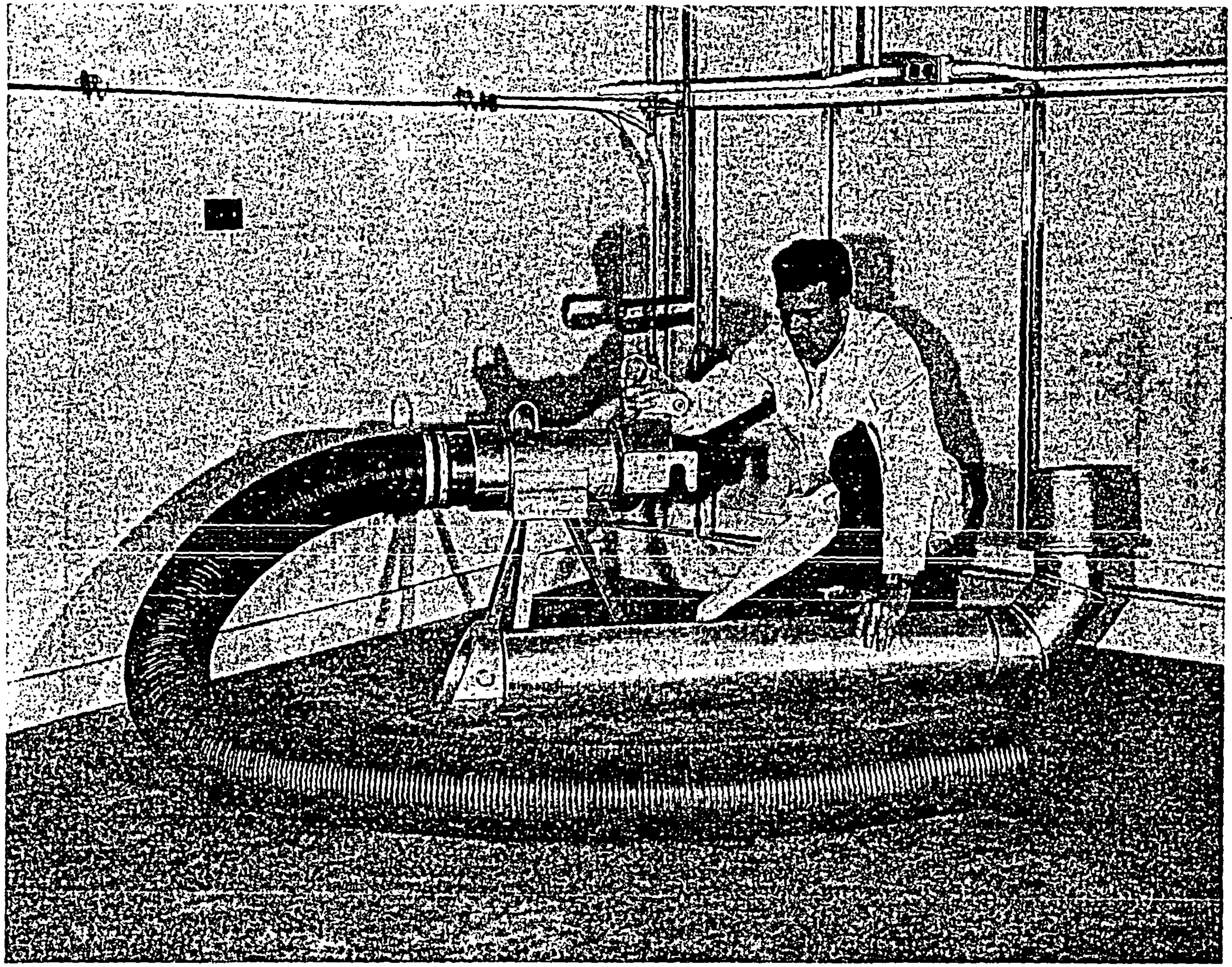
1. Reactor Building

The reactor pool is housed in a large bay 48 ft wide, 120 ft long and and extending 40 ft above the general floor level. At the beam hole end, the floor is dropped 20 ft to provide access to the tubes as they emerge from the pool wall.



REACTOR POOL COOLING & PURIFICATION SYSTEM

Figure 9



FLEXIBLE TUBE AND COUPLING FOR FORCED COOLING

Figure 10

An overhead bridge crane of 10 ton capacity runs the length of the bay and services both the reactor area and the beam room. A large overhead door at one end of the bay allows trailer-trucks to drive under the crane for removal of shipping casks and other heavy objects.

The exterior walls of the reactor bay, and of the rest of the building, are of typical curtain wall construction. They consist of aluminum panels fastened to the structural steel framework and insulated by a 1 in. layer of Fiberglas. The roof consists of metal deck, 3/4 in. Fiberglas insulation and four-ply roofing. There are no windows in the bay and only two doors opening directly to the outside. The estimated leakage rate with doors closed and ventilator off is one air change in 32 hours.

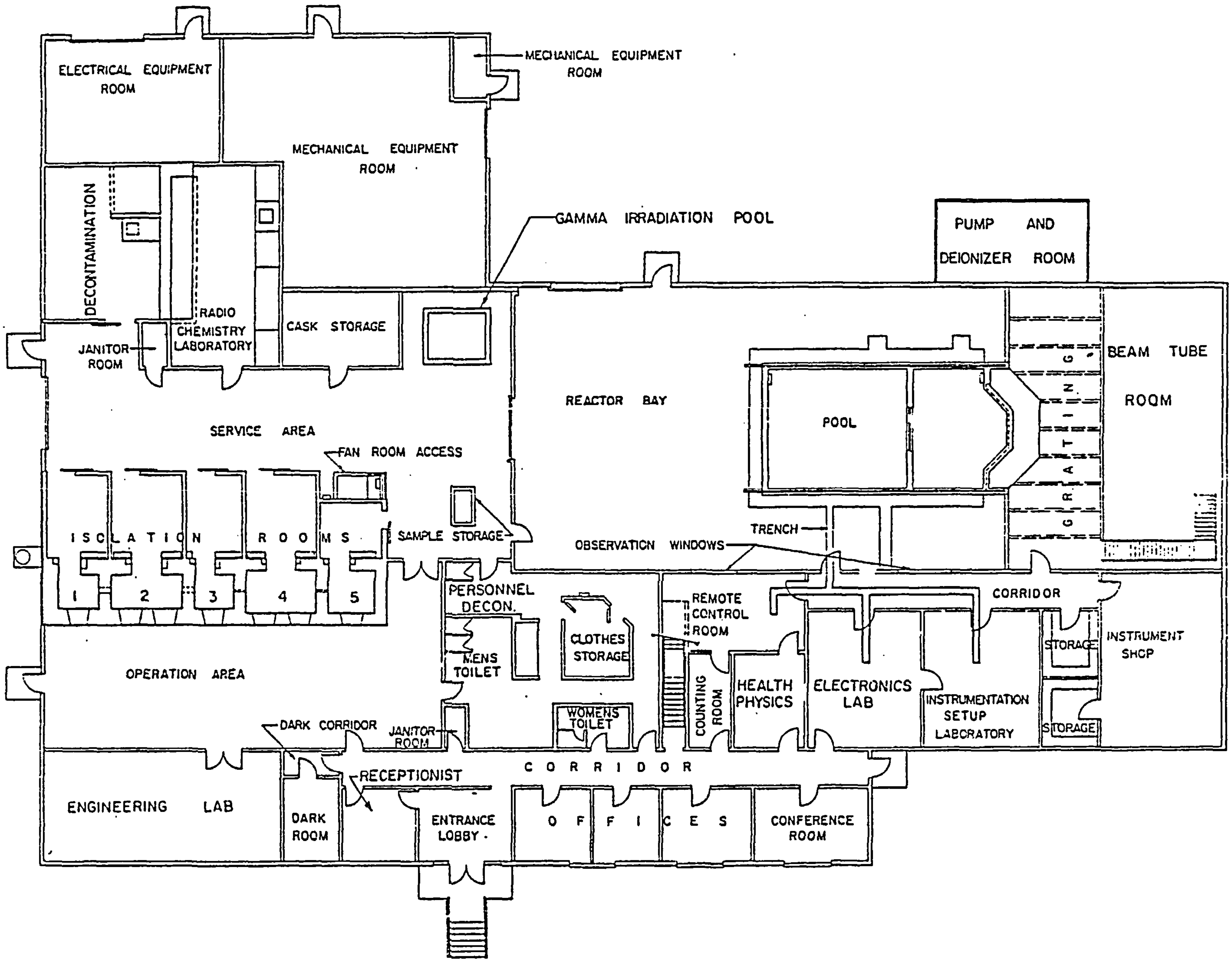
The reactor bay is part of a 24,700 sq ft building which also houses a hot laboratory for the study of radioactive materials. The main floor plan is shown in Figure 11. Associated with the reactor portion of the building are an instrument shop, electronics laboratory, remote control room and instrumentation set-up lab. Cable trenches lead from the reactor pool to the latter three rooms to permit location of equipment away from the reactor. It is hoped that this will prevent the area around the pool from becoming excessively cluttered. If it is found desirable in the future, the reactor control console may be moved from its present location on the bridge to the remote control room.

Additional building facilities include offices, counting room, health physics laboratory, engineering laboratory and dark room. To the rear of the hot cell block is a service area, radio-chemistry lab, decontamination area, and gamma irradiation pool. Normal access to the rear of the cell block is via a change room which isolates these potentially contaminated areas from the rest of the building.

All other areas in the building are expected to remain free of radioactive contamination. Accordingly, personnel working in the reactor area normally will by-pass the change room completely. However, if an accident should result in activity being released in the reactor bay, the bay can be sealed off from the other areas and access restricted to personnel who have passed through the change room.

2. Pool Water Supply System

In order to reduce fuel element corrosion and prevent build-up of impurities in the reactor pool water with consequent neutron induced activity which would result, the pool supply water is softened and deionized prior to being supplied to the pool. This is accomplished by passing the water, obtained from wells in the area, first through a water softener tank to remove the calcium and magnesium ions, and then through a mixed resin bed ion exchanger (deionizer) to remove



MAIN FLOOR PLAN CURTISS-WRIGHT REACTOR AND RADIOACTIVE MATERIALS LABORATORY

Figure 11

the remaining anion and cation impurities. The effluent water contains fewer minerals than distilled water.

The equipment consists of two softener units and two deionizer units with the associated equipment required for operation of the system and regeneration of the resins once they have become exhausted. Normally, one softener and one deionizer will be in use while the second pair is acting as a standby or is in the process of regeneration.

The equipment shown in Figure 12 is installed in the mechanical equipment room, and a 2 in. line runs to each section of the reactor pool for filling. Filling time will be approximately 24 hours. Figure 13 is a block diagram of the system.

An analysis of the dissolved solids in the well water gave results leading to the hypothetical combinations shown in Table IV.

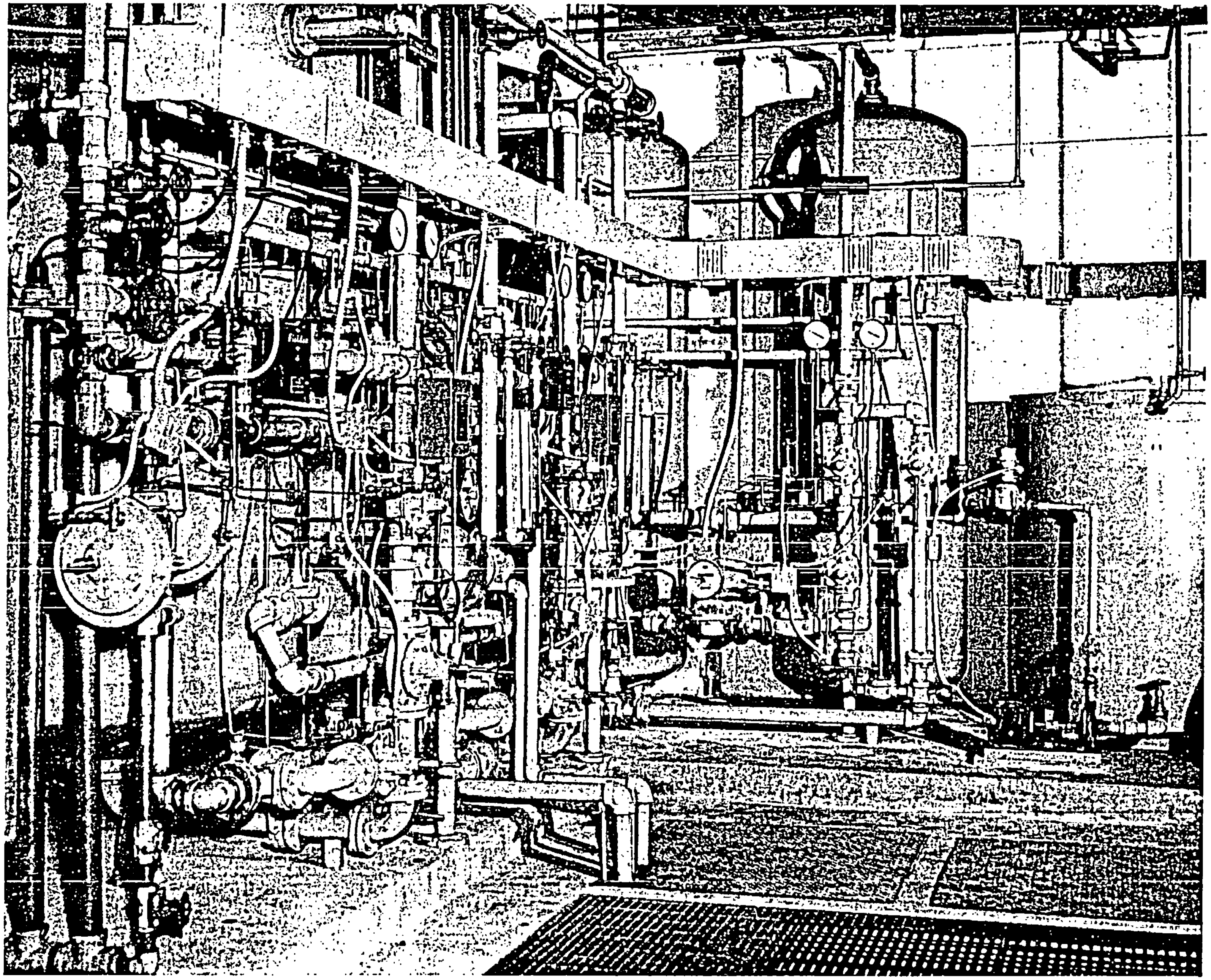
TABLE IV

Impurities in Well Water

		Parts per Million
Silica	SiO_2	9.6
Aluminum Oxide	Al_2O_3	Trace
Iron Oxide	Fe_2O_3	Trace
Calcium Bicarbonate	$Ca(HCO_3)2$	71.4
Magnesium Bicarbonate	$Mg(HCO_3)2$	23.5
Magnesium Sulphate	$MgSO_4$	2.4
Sodium Chloride	NaCl	16.5
Sodium Sulphate	Na_2SO_4	12.1
	Total Solids	135.5
	Total hardness as calcium carbonate	62.0

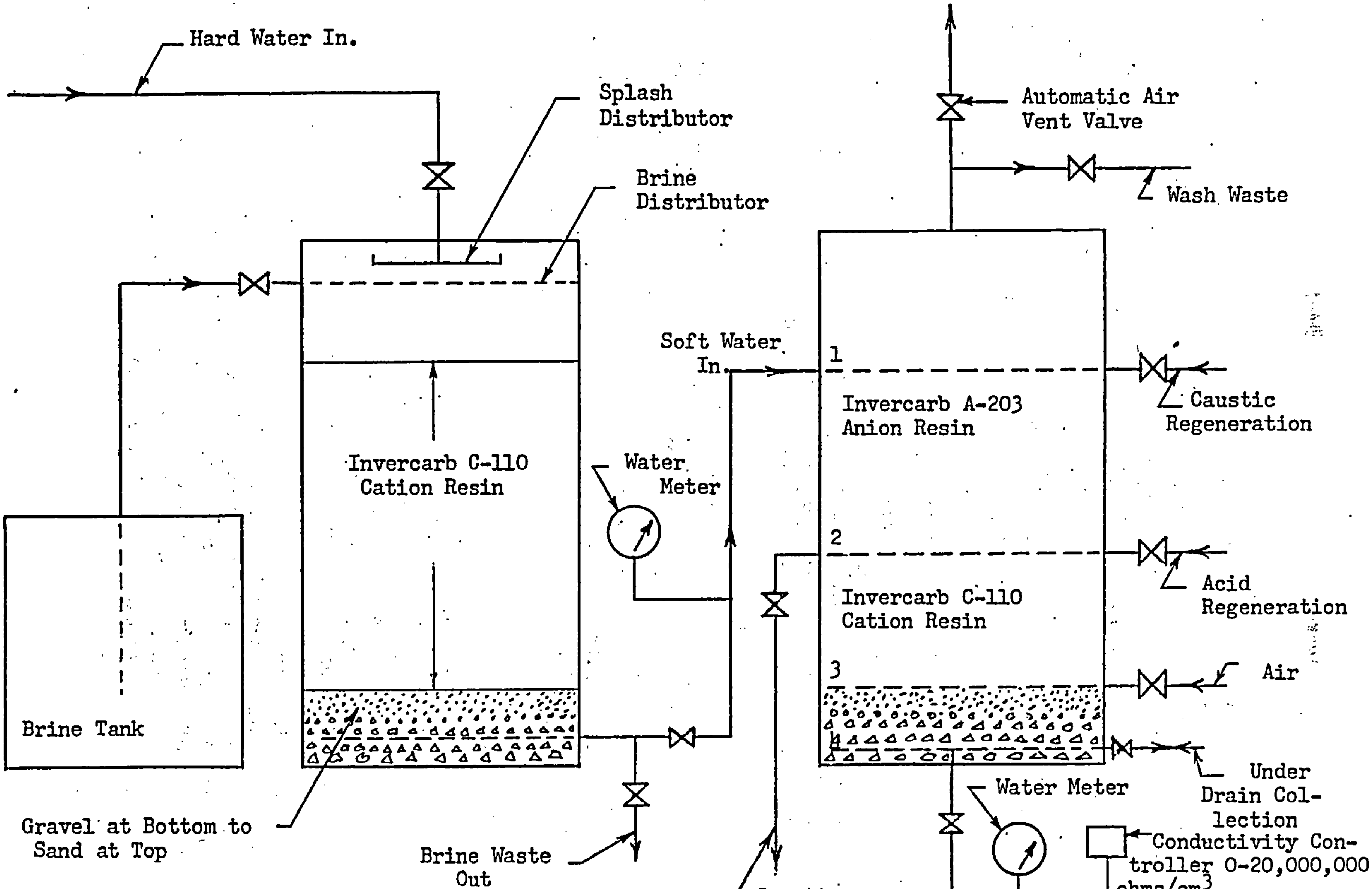
3. Pool Water Purification System

While each filling and addition of water to the pool is from a demineralized supply, there is still the possibility of impurity build-up in the water due to corrosion and leaching, introduction of handling implements into the water and the fact that the pool is open at the top. In order to reduce fuel element corrosion and prevent impurity build-up and hence the build-up of neutron induced activity, a continuous purification system has been provided.



POOL WATER PURIFICATION SYSTEM

Figure 12



BLOCK DIAGRAM
POOL WATER SUPPLY SYSTEM

Figure 13

As Figure 9 shows, this system shares some of its piping with the cooling system. Purification is accomplished by circulating the water through one of two mixed bed ion exchanger columns and thence routing it back to the pool. The second ion exchange column acts as a standby. The units are capable of deionizing the water at a normal flow rate of 15 gpm with a pressure loss of not more than 4 psi. An emergency flow rate of 100 gmp can be achieved with a pressure loss of not more than 22 psi by using a portable auxiliary pump. This is based on the effluent water having a 0.5 ppm total solids (50% in suspended form), 1 megohm-cm resistivity, and pH of between 7.2 and 10.2. The effluent from the units contains not more than 0.004 ppm total solids. A pre-column strainer, located in the line prior to the pump, removes most particulate matter so that it does not gather on the input surfaces of the ion exchange cartridges thus decreasing the flow rate through them.

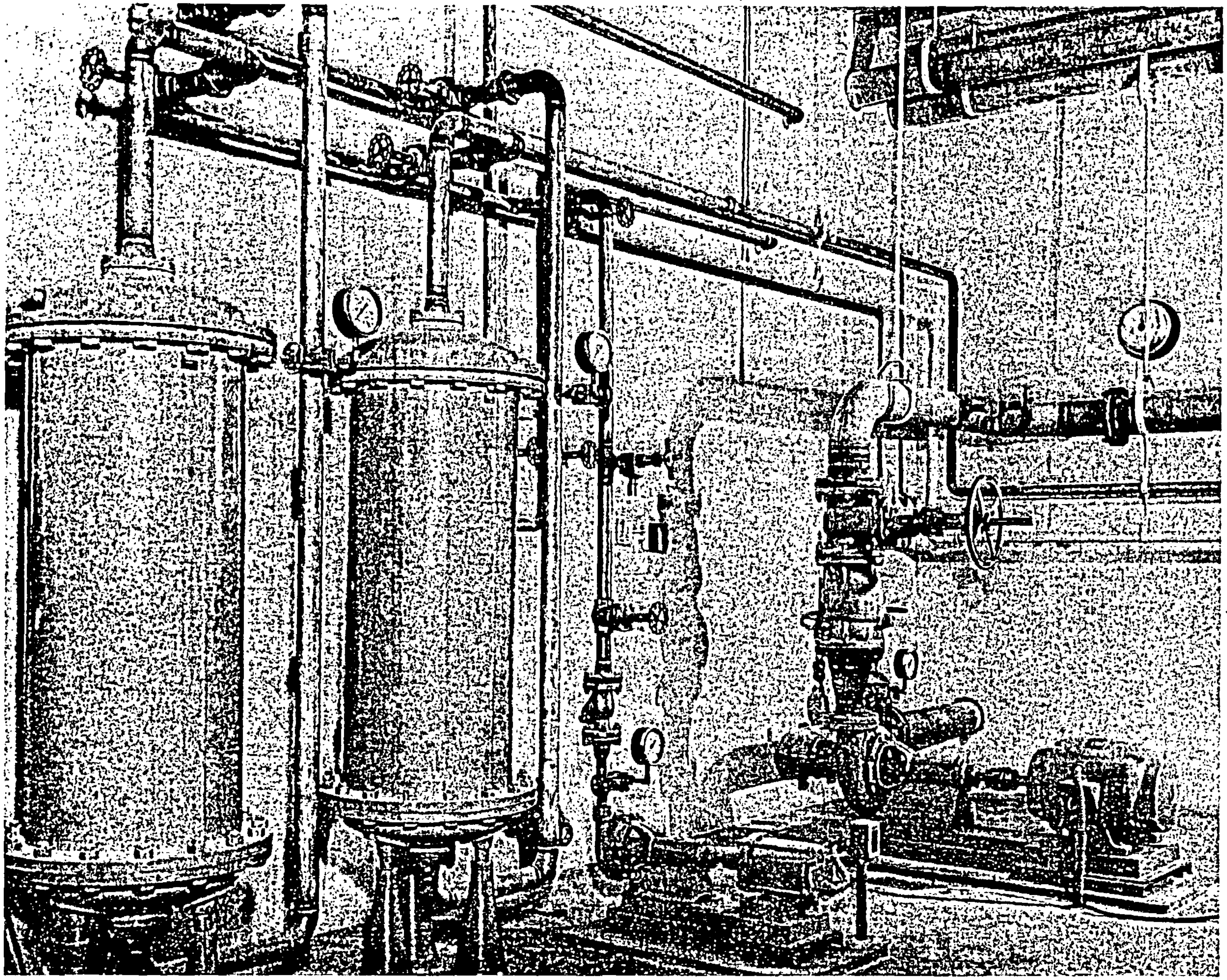
The efficiency of the ion exchange column in use is determined by means of a conductivity cell and sample-cock located on the common effluent line from the two columns. The cartridges which contain the resins are removable and, when exhausted, will be replaced. No attempt will be made to regenerate the resins. Used cartridges will be handled as radioactive waste.

The ion exchange assemblies shown in Figure 14 are mounted in the pump room on the south side of the building and if necessary may be shielded by 8 in. concrete blocks. It is not expected that the resins will become sufficiently radioactive under normal operation to make handling difficult due to gamma radiation. If the resins should become highly contaminated, perhaps as a result of a ruptured fuel plate sheath, it would be necessary to remove the ion exchange cartridges into a lead shielded container specially constructed for that purpose.

4. Beam Tubes

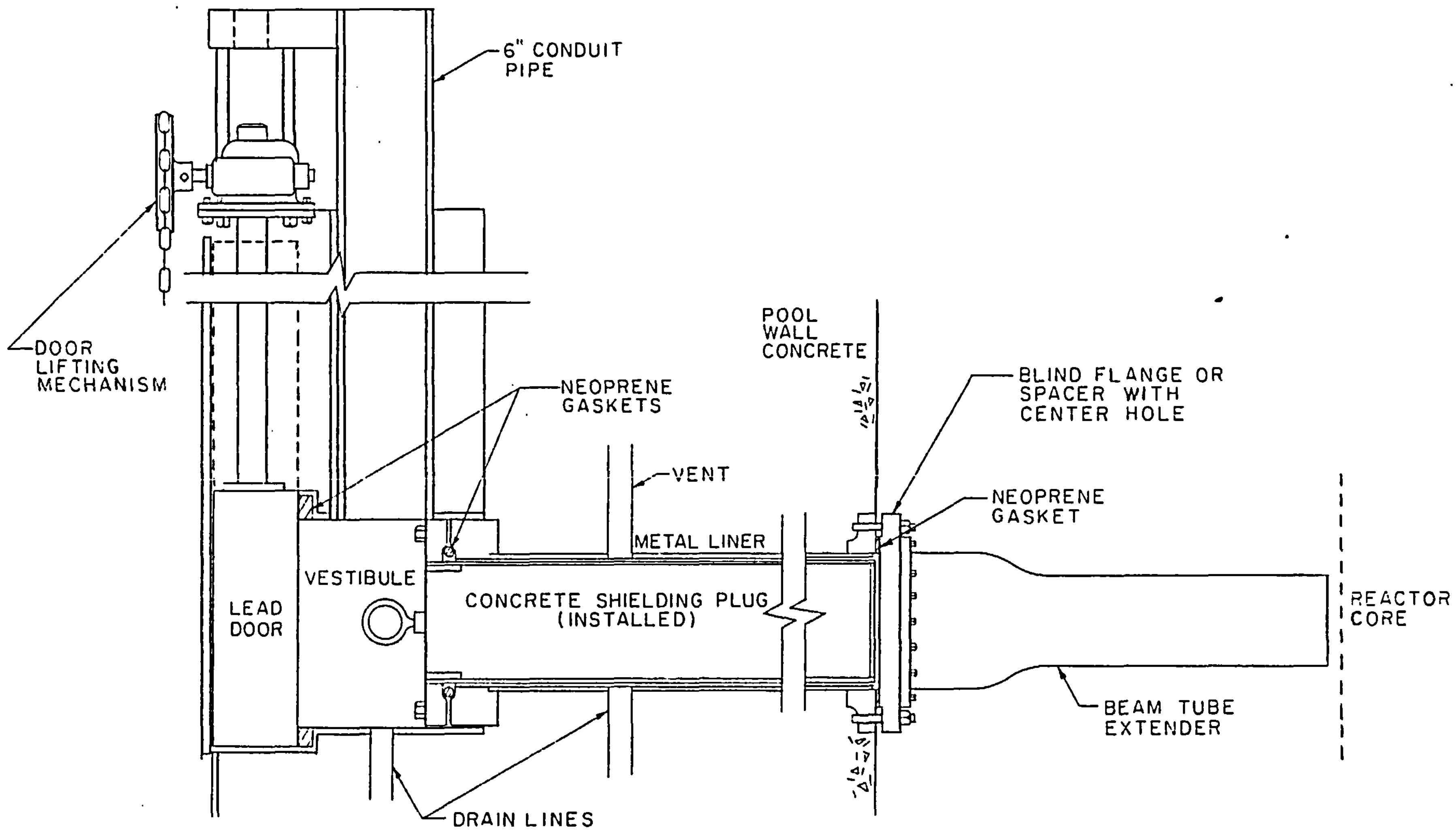
The beam tubes are provided primarily to obtain a collimated beam of neutrons which can be used for experimental purposes, and they are also used to provide dry irradiation chambers. The open ends of the beam tubes terminate at the beam room side of the west wall of the pool, and the operations required to install or remove equipment from the beam holes will be performed from the beam room.

Figure 15 is a cross-sectional view of a typical beam tube. The first section, constructed of boral (aluminum-boron carbide-aluminum sandwich), forms a liner for the hole through the concrete wall and extends from the pool wall outer face to its inner face. The second section, constructed of aluminum, extends from the inner face to within a fraction of an inch of the reactor core and is called a beam tube extender. The extender tubes are closed at their reactor ends, and are so arranged that these ends fall one above the other along the vertical center line of the reactor face. Figure 16 is a photograph taken during construc-



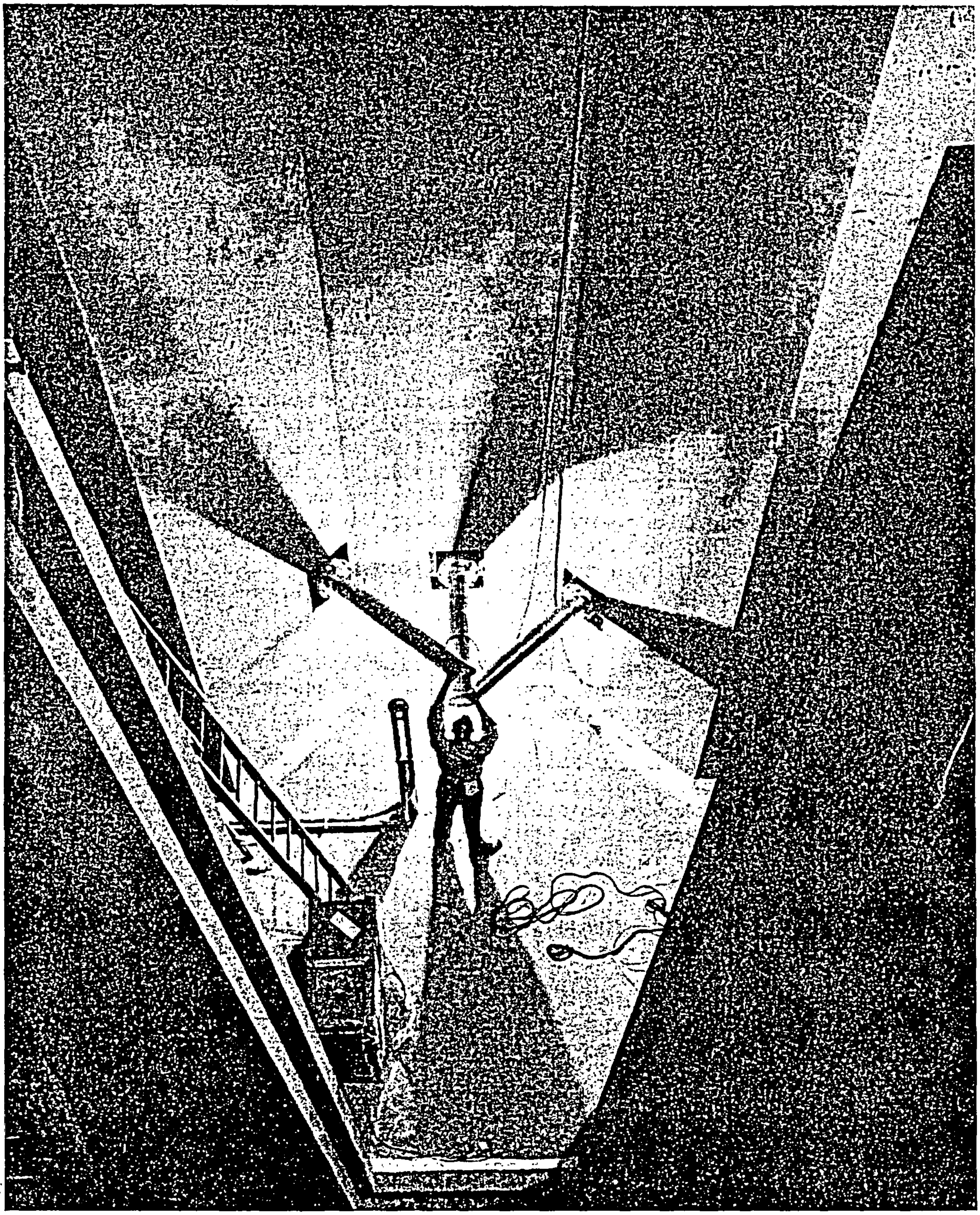
PUMP ROOM AND ION EXCHANGER ASSEMBLIES

Figure 14



TYPICAL BEAM TUBE

Figure 15



BEAM TUBE END OF POOL

Figure 16

tion, showing the beam tubes in place.

There are three different arrangements of the beam tubes which may be made. These are as follows:

The first section of the beam tube may be sealed by a water tight blind flange, in which case the pool must be drained in order to install a beam tube extender. Without the extender in place, the neutron flux at the inner face of the beam tube is small and the hole will be useful for irradiations at relatively low flux values.

A spacer having an approximate 8 in. diameter hole in its center may be installed in place of the blind flange, and a beam tube extender then flanged to this. With this arrangement, there is a direct path from the outer face of the beam hole to the inner end of the beam tube extender, and it is possible to insert equipment or materials to within an inch or two of the reactor core. The neutron flux at the inner end of the beam tube extender will be approximately 4×10^{12} n/cm^2/sec while a flux of about 10^9 n/cm^2/sec is expected at the exit aperture of the beam tube at 1000 kw operation.

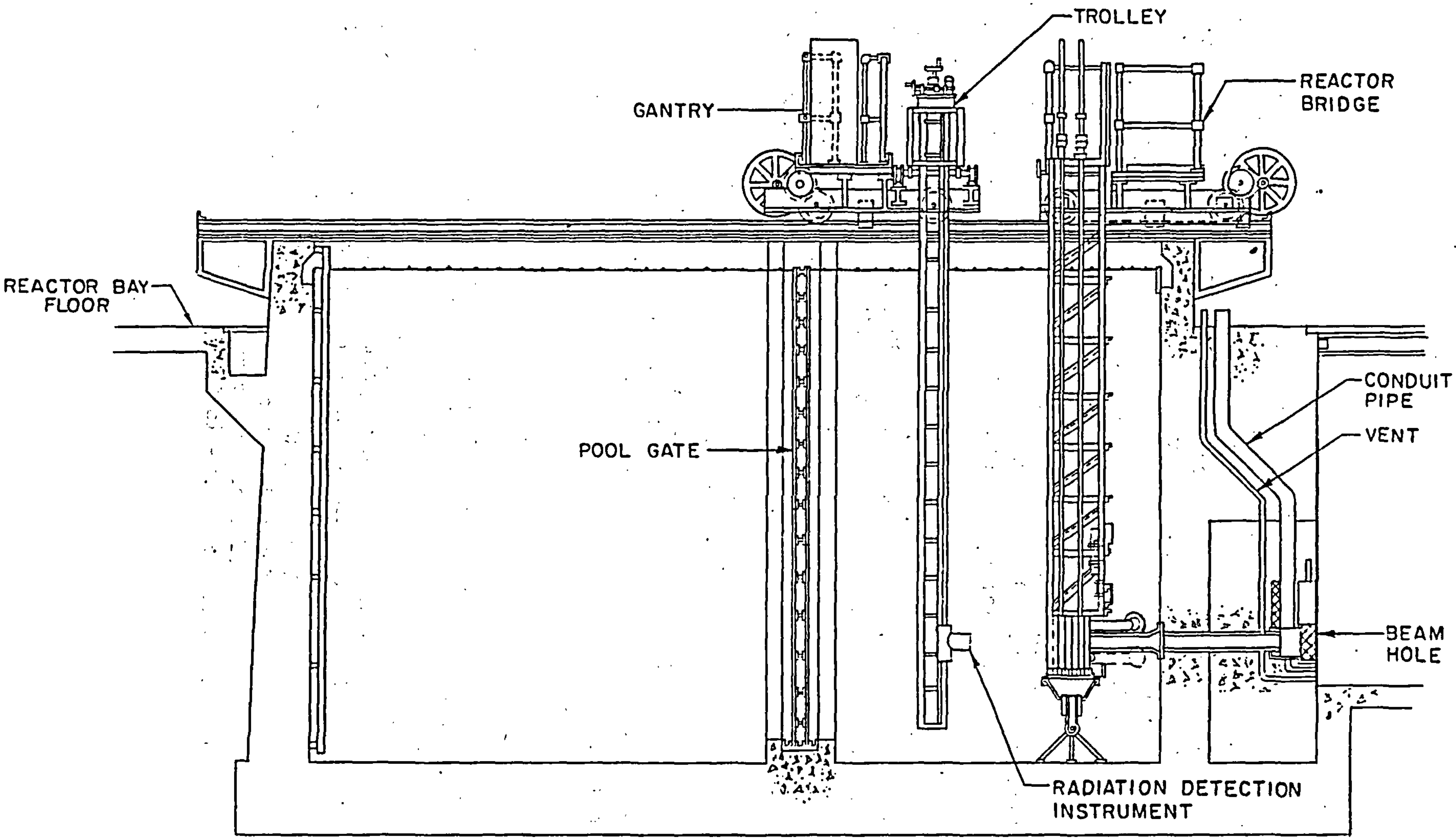
A beam tube extender may be flanged directly to the blind flange in which case equipment or materials may be inserted into the beam hole only as far as the blind flange.

The $\frac{1}{4}$ in. boral lining prevents loose concrete dust which has become radioactive from accumulating on installations within the tube and being removed with them. The boron filling keeps the slow neutron flux to a minimum, thus preventing the production of high energy gamma rays by neutron capture in the concrete shield and thereby reducing the shielding requirements.

Concrete shielding plugs are available for each hole when not in use. They are contained in a $\frac{1}{4}$ in. thick aluminum outer casing.

There are one inch drains leading from the beam hole liner and the vestibule to the contaminated waste system. These beam hole drains will carry away any water which might leak into the beam tubes.

As an additional safeguard, there is an outer door covering the opening of the beam hole. This is made watertight by means of a door-sealing gasket. Necessary connecting lines to equipment in the beam tube are fed in via a vertical 6 in. tube which originates above the pool water level. This tube is shown in Figure 17. The outer lead shielding door may be opened by loosening the door-sealing mechanism and lifting the door vertically by means of a worm gear screw jack.



REACTOR POOL FACILITIES — VERTICAL SECTION

Figure 17

5. Pneumatic Rabbit

The pneumatic rabbit is used mainly for short term irradiations and is extremely useful for the radio-assay of elements having a very short half-life. The equipment includes an irradiation carrier into which the material to be irradiated is placed. It is then possible to position the carrier pneumatically very close to the reactor core, the period of travel being about two seconds. Following the desired irradiation, it is possible to remove it in the same period of time. Figure 18 is a schematic diagram of the system.

The irradiation chamber, at the reactor end of the loading and unloading tube, is positioned approximately one inch from the center of the south side of the reactor core. It may be seen in Figure 16.

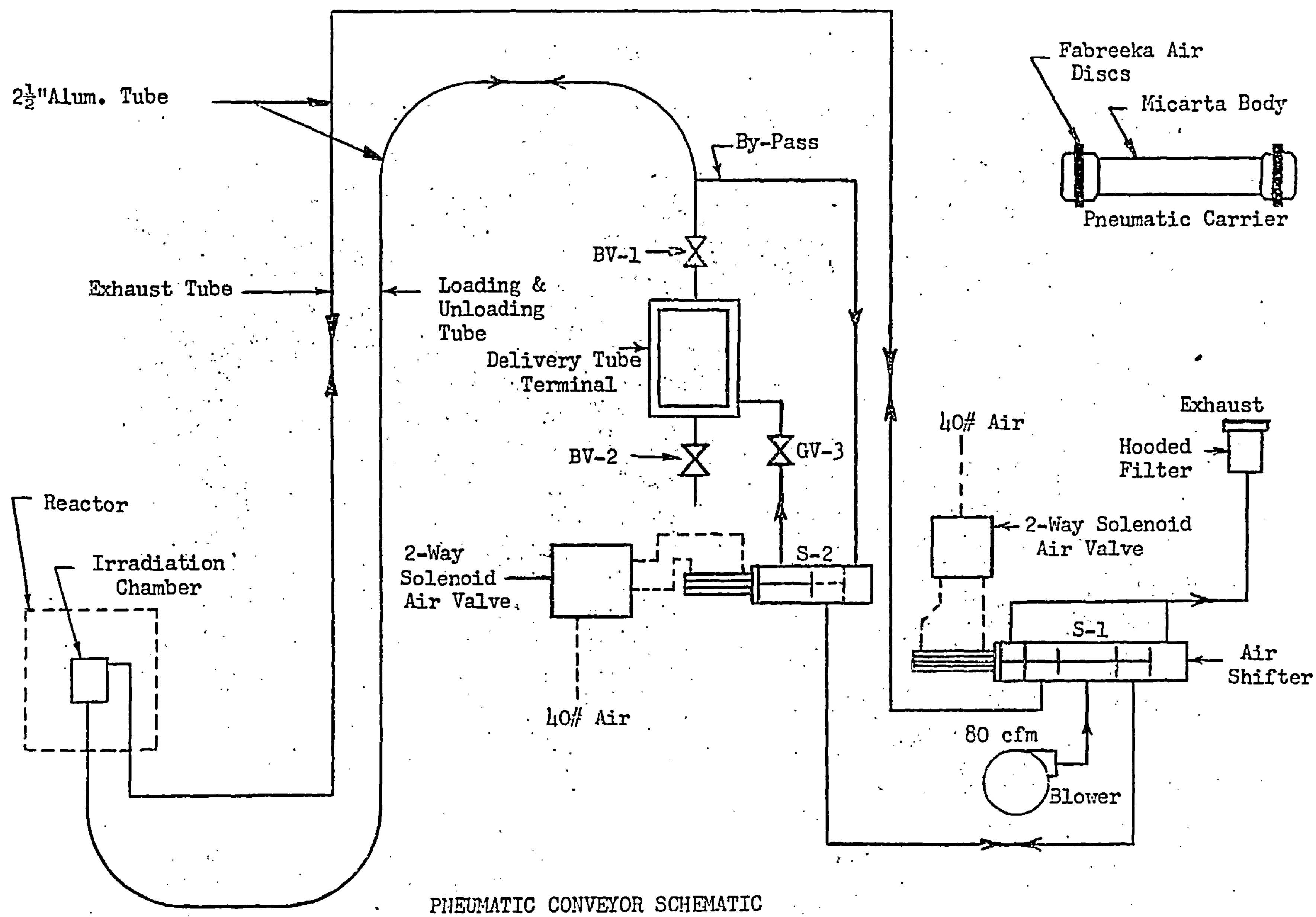
The blower has a capacity of 80 cfm of air at a pressure of 2 psi. This air is used for movement of the pneumatic carrier to and from the irradiation chamber at a speed of about 30 ft/sec. It is also used for cooling the carrier once it is in the irradiation position.

The delivery tube terminal is located on the east wall of the beam room together with the control panel. This is shown in Figure 19.

The pneumatic carriers are similar to design to the department store type. They are expendable and suitable for use in temperatures up to 137°F. The main body and heads are constructed of Micarta, and the two air discs of Fabreeka. The heads are threaded to the body portion of the carrier and it is necessary to unscrew one head to insert or remove material. When irradiated material is removed, the operation is carried out remotely.

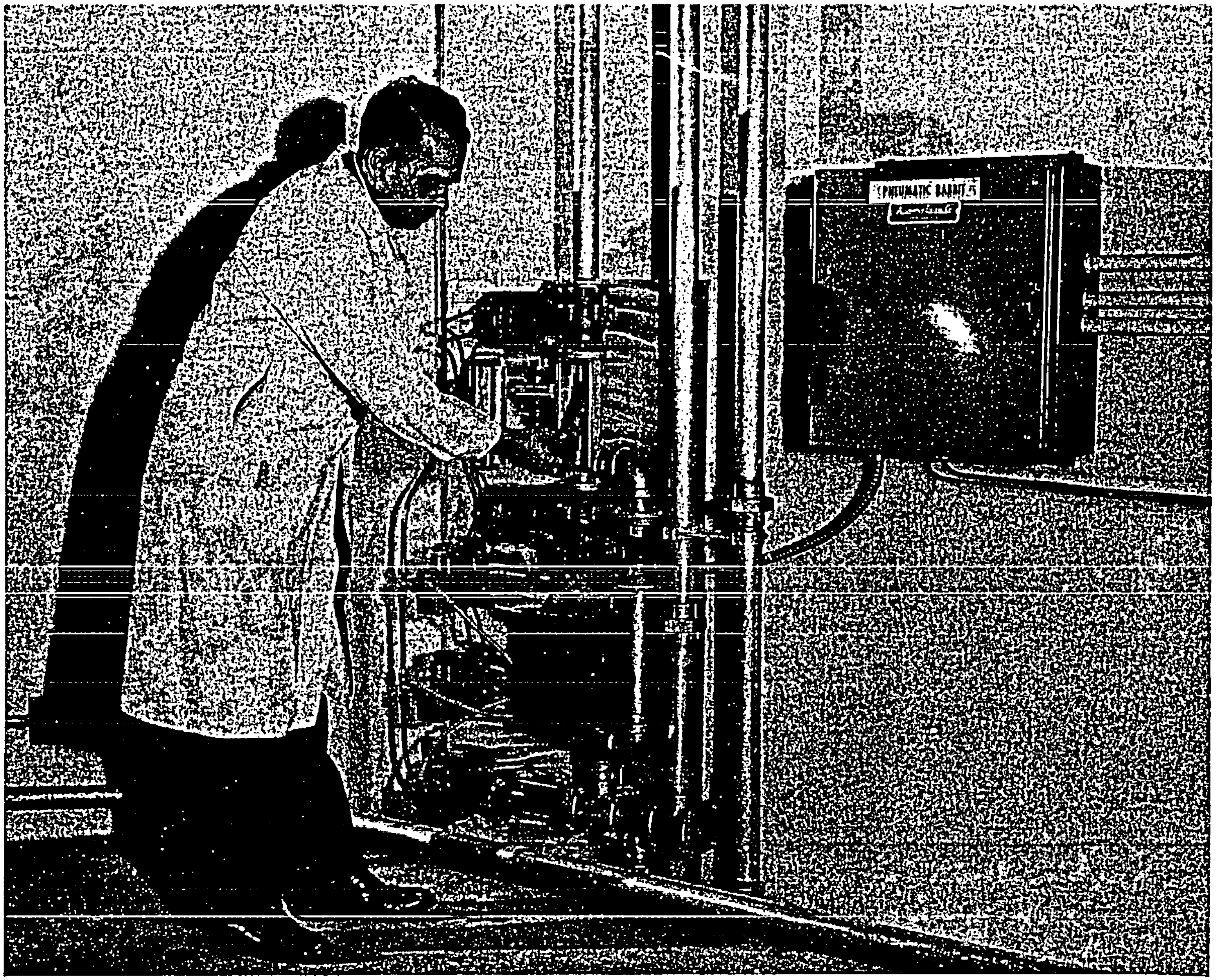
6. Instrument Bridge

The instrument bridge was designed to support a static load of 3,500 lbs at its center, and its basic construction is similar to that of the reactor bridge. The total width of the instrument bridge is 8 ft. Two rails, 24 in. apart have been mounted to permit an instrument cart to traverse the length of the bridge. A supporting tower, the lower part of which will be fabricated of 2S aluminum, will be attached to the bottom of the cart and will extend down to the reactor core level. An elevator will traverse the length of the tower, thus permitting objects to be positioned at any desired vertical distance from the reactor (see Figure 17). Four degrees of freedom described in Table V are available for positioning objects relative to the reactor core.



PNEUMATIC CONVEYOR SCHEMATIC

Figure 18



PNEUMATIC RABBIT TERMINAL AND CONTROL PANEL

Figure 19

TABLE V

Degrees of Freedom Available for Positioning
Objects Relative to the Reactor Core

Mode of Motion	Method of Attainment	Accuracy of Positioning
Traverse length of pool	Movement of instrument bridge	± 2 mm
Traverse width of pool	Movement of cart on bridge	± 2 mm
Rotation in horizontal plane	Rotation of tower relative to cart	± 0.5 degree
Vertical	Movement of elevator on tower	± 2 mm

7. Radioactive Waste Handling Facilities

a. General. The tentative plans which have been laid for the disposal of radioactive wastes from the reactor-hot lab complex are based on the following estimated quantities:

Waste		Volume Generated Annually
Liquid	High Level	5,000 gal
	Low Level	75,000 gal
Solid	High Level	1,000 cu ft
	Low Level	4,000 cu ft

It is planned to bury all low level solid waste in an established burial ground on site. This site has been set aside specifically for burial of toxic wastes and is appropriately posted and fenced. Solid waste containing more activity than can be buried under existing Federal and State Regulations will be stored until shipped off-site.

It is anticipated that 90% of the low level liquids can be released, under controlled conditions, directly to the environment. The remaining 10% will be concentrated by evaporation and shipped off-site together with the other high level solid wastes. Part of the high level liquid wastes can be concentrated, and all of it will be solidified and contained in concrete.

As a result of these operations, approximately 90 tons of waste plus concrete or metal shielding will have to be disposed of off-site each year. This will probably take the form of about three hundred 55-gallon steel drums.

b. Liquid Waste Treatment Plant. A plant for the treatment of liquid contaminated wastes is housed in a separate building about 50 ft from the main building. The location is shown on the plot plan, Figure 20. The capacity of the plant was based on an estimated flow of 300 gal per day of which about 10% might actually require treatment.

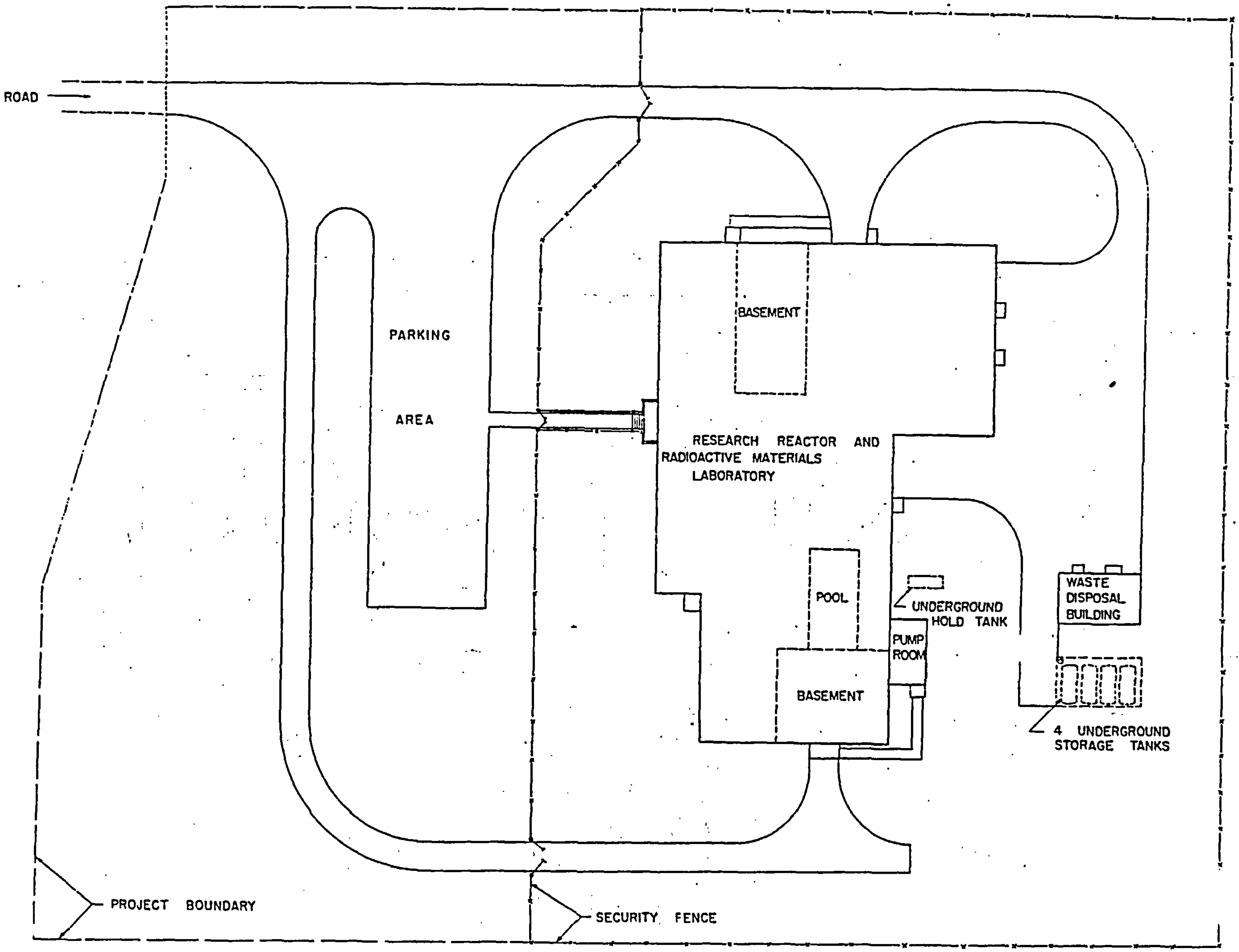
Figure 21 shows that the flow of wastes to the treatment plant is via one of two collection systems. A "low level" waste system originates in areas of potential radioactive liquid wastes and contamination such as floor drains in the hot cells, radio-chemistry laboratory drains, and decontamination effluents from the fume hoods in the decontamination room. Drains from areas of unlikely but possible radioactive contamination lead to a "suspect" waste system and originate from such places as the service area, change room showers, reactor area, and the personnel decontamination sink in the change room. Each system may be terminated in either of two 3000 gal underground tanks. When one tank in one system is full, the other tank in the same system may receive drainage while the contents of the full one are being drained or circulated for treatment.

There are two pumps for each system. One pump in each system will operate when required, the second acting as a stand-by. When a tank is full, the contents will be mixed by circulation through the pump and back to the tank again. A sample will then be taken from the sampling cock on the pressure side of the pump and an analysis will be made for radioactive content. If the sample is below the maximum permissible level for release, the contents of the tank will be pumped to disposal, as described below, via the gravity head tank. If the sample is above the permissible level, the contents of the tank will be pumped to the evaporator.

Following evaporation, the sludge will be drained to drums which will be shipped off-site for ultimate disposal. The water vapor from the evaporator passes through a heat exchanger type condenser, and the condensate flows into the vacuum receiver tank where it is sampled and then discharged via the gravity head tank to the stream system described below or recycled through the evaporation process.

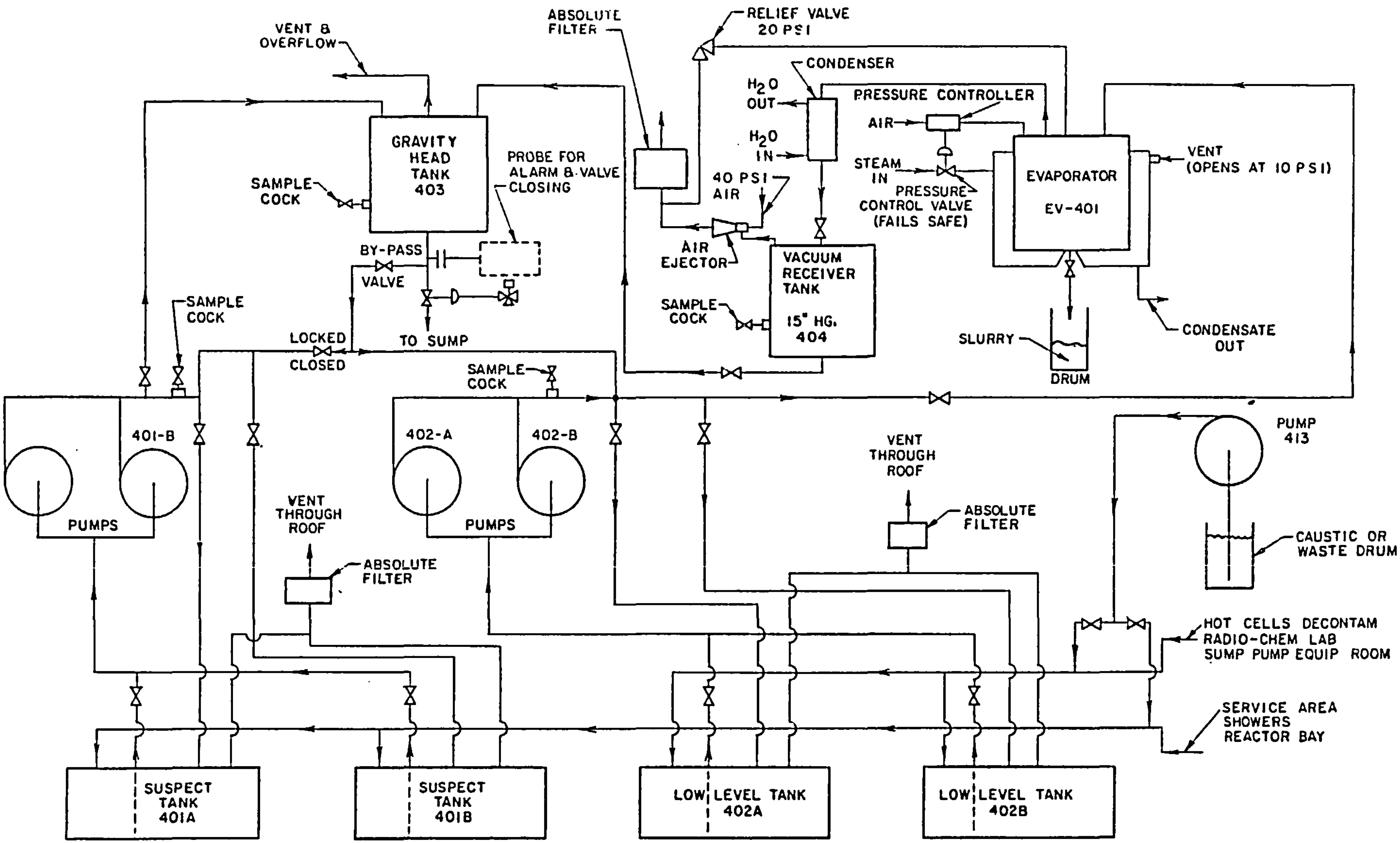
The vacuum receiver tank is kept at a vacuum of about 15 in. of Hg by means of a jet ejector operated by compressed air. This causes the liquid in the evaporator to boil at a relatively low temperature (approximately 80°C). The jet ejector exhausts its air through an absolute filter to the atmosphere, above roof level.

A radiation monitor with an alarm is situated on the drain line from the gravity head tank. If, through unforeseen circumstances, a slug of active liquid is being drained to the streams, the monitor will cause a valve in the drain line to close. The liquid in the gravity



SITE LAYOUT

Figure 20



WASTE TREATMENT PLANT

Figure 21

head tank then can be routed back to the storage tank for reprocessing.

A station has been provided for the addition of caustic to any storage tank for acid neutralization. This station also may be used for the transfer of radioactive waste solutions from other laboratories to one of the storage tanks.

Each group of two storage tanks is vented above roof level through absolute filters.

The system has been designed to be flexible. The following operations are possible:

1) The contents of any storage tank can be pumped to the evaporator.

2) The contents of any storage tank can be pumped to the gravity head tank.

3) The contents of any storage tank can be pumped to any other storage tank.

4) The contents of the gravity head tank can be routed to any storage tank.

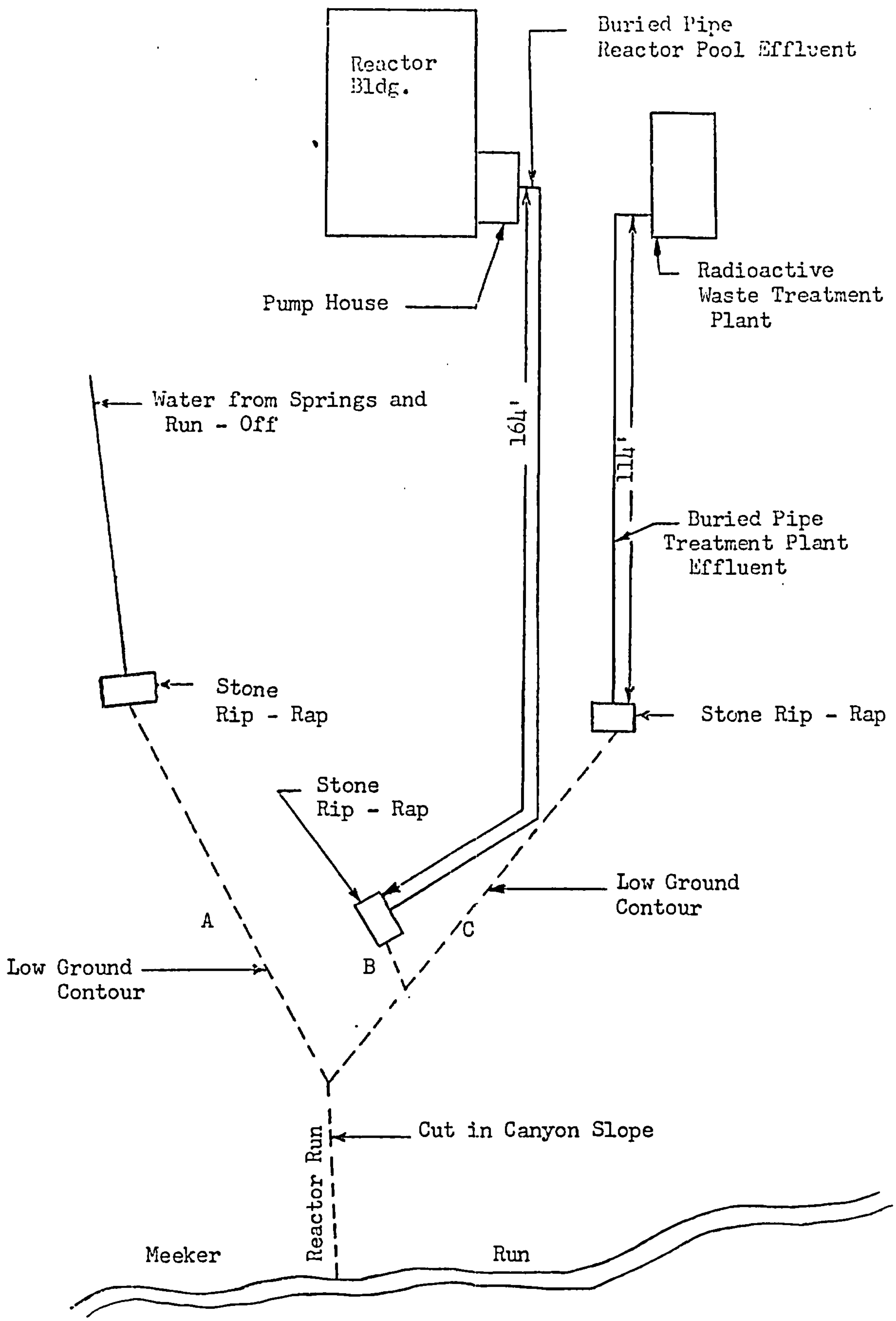
c. Criterion for Release of Liquid Wastes to the Environment. Radioactive liquid wastes, when sampled and found safe for release, are released to the area stream system. Figure 22 shows the path of released wastes in the vicinity of the laboratory installation. Meeker Run flows at the bottom of a canyon which is located approximately 200 yds from the reactor site.

Figure 23 shows the approximate Quehanna site boundary with the location of the reactor and hot lab installation designated by the letter "R." The stream being joined by other streams, flows out of the Curtiss-Wright boundary in a general southerly direction.

Wastes released from the treatment plant or the reactor pool are carried by underground pipe to points where the contours of the terrain give positive flow to Reactor Run and thence to Meeker Run. The ends of the pipes terminate over crushed stone "rip-rap" for erosion control.

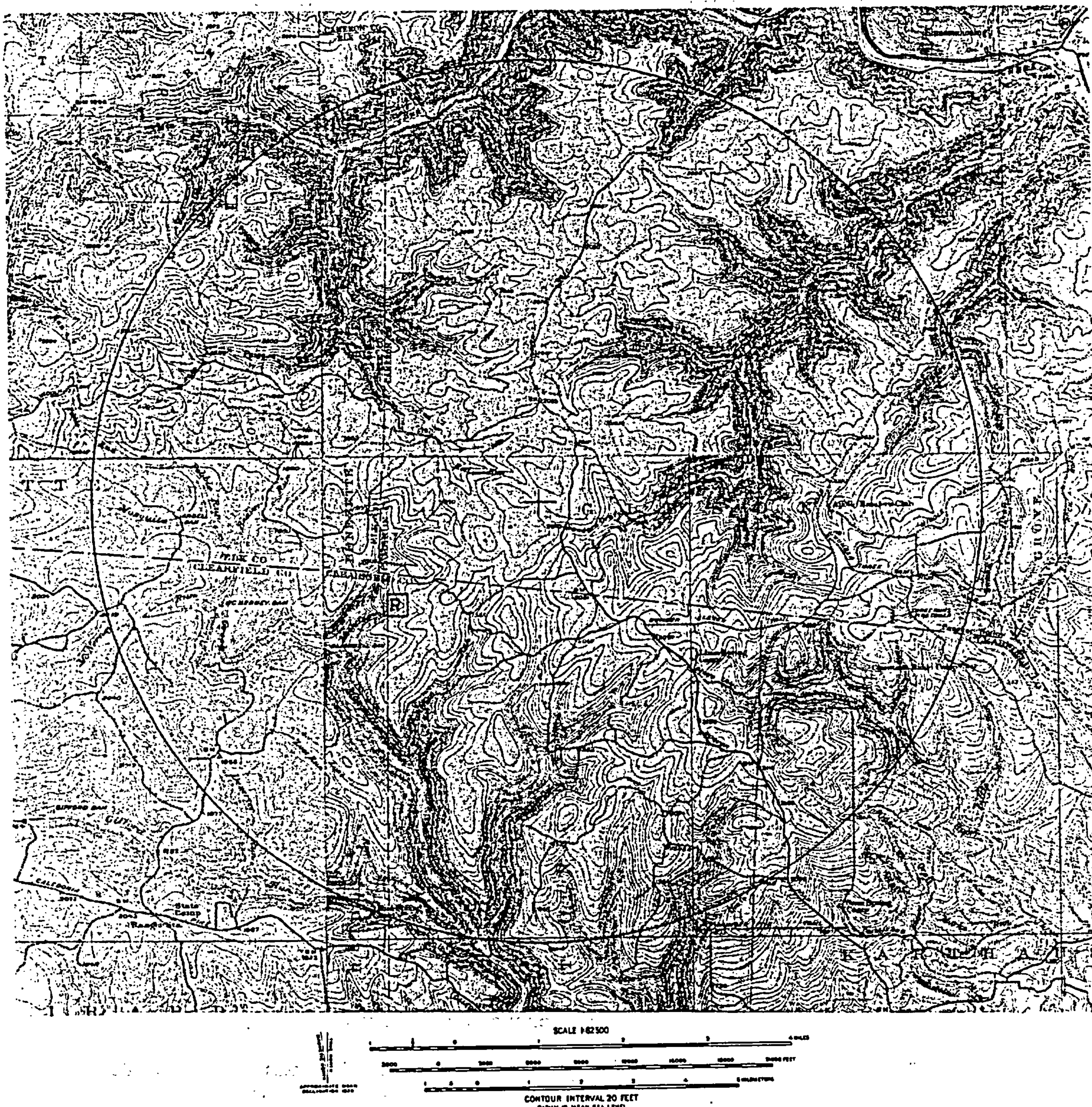
The flow of spring water following Route "A," shown in Figure 22 is a little over 30 gpm so wastes following Route "C" will be diluted by at least that flow rate plus whatever volume of water may be flowing due to run-off during periods of wet weather.

Investigations indicate that the average flow in Meeker Run, where Reactor Run joins it, is about 1000 gpm. A weir, having a "V" shaped



SCHEMATIC DIAGRAM - PATH OF LIQUID WASTE RELEASE FROM REACTOR AND HOT LAB

Figure 22



THE QUEHANNA SITE

R is Reactor

Figure 23

overflow, is installed in Meeker Run to measure minimum flow rates. Wastes are released at such a rate and with such a radioactive concentration that the average radioactive content of the water in Meeker Run, just downstream from the point of release, is not more than 1/10 of maximum permissible concentration. At no time will wastes be released at a rate which would cause the concentration in Meeker Run to be above MPL.

Figure 23 shows that much greater dilution will occur before the wastes leave the site, and drainage area calculations indicate that average concentrations in Mosquito Creek, where it leaves the site, will be well below 1/100 MPC.

A thorough program of sampling is carried out to ensure that the average limit, quoted above, is not exceeded. This includes continuous sampling of any effluent being released to the stream system, sampling of stream water at various points, and a sampling of stream silt and algae.

d. Reactor Pool Water Disposal. Section I-B-3 describes the pool water purification system.

If it becomes necessary to empty one or both sections of the pool, the water is mixed, prior to sampling, by circulation through the circulating pump and back to the pool. If its radioactive content is found to be below a predetermined concentration (dependent on the flow in Meeker Run), it can be pumped directly to the stream system at 700 gpm. Figure 22 shows the path which it will take.

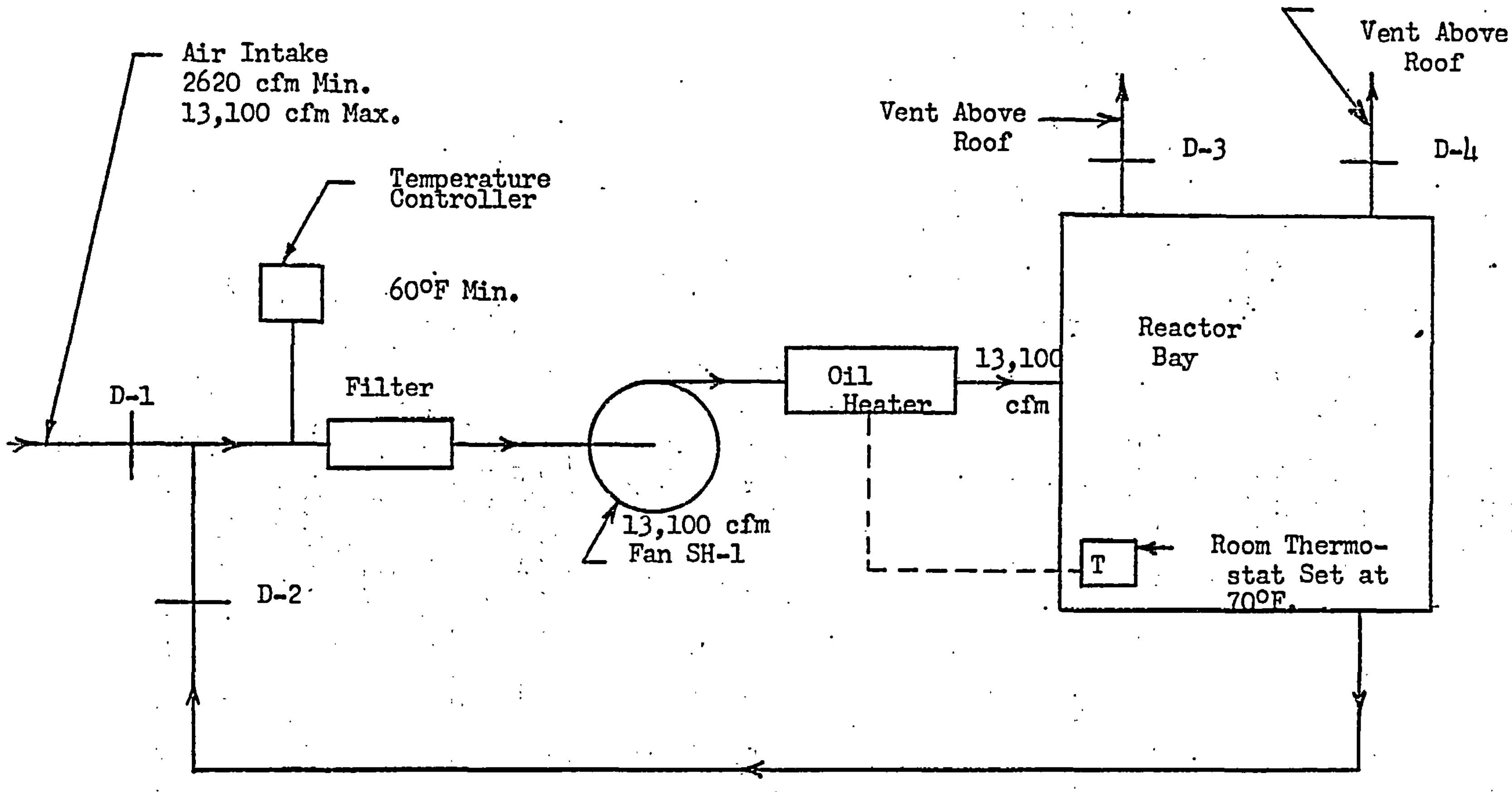
If the radioactive content of the water is too high to allow direct release, it is recirculated through the deionizers until its radioactive concentration is sufficiently low to allow direct release.

8. Ventilation

Heat and ventilation for the reactor bay are supplied by a single overhead space heater. This oil fired unit provides a minimum of 20% outside air and at least six air changes per hour.

The exterior walls of the bay are fabricated of aluminum panels fastened to the structural framework and insulated by 1 in. layer of Fiberglas. The roof consists of metal deck, 3/4 in. Fiberglas insulation and four-ply roofing. There are no windows and only two doors open directly from the bay to the outside. Estimated leakage rate with doors closed and ventilation off is one air change in 32 hrs.

Figure 24 is a schematic diagram of the system. The supply fan is suspended from the ceiling of the reactor bay and draws its air from



REACTOR BAY SUPPLY & EXHAUST

Figure 24

above roof level. Associated with the supply fan is an oil fired space heater which is actuated by a thermostat in the reactor bay. There are two exhaust vents located at the opposite end of the roof from the supply unit.

A system of recirculation normally is used whereby a portion of the exhaust air is recirculated to the intake of the supply fan, the amount depending on the temperature of the combined recirculated air and intake air. A temperature controller modulates dampers to maintain the temperature of the air into the fan at a minimum of 60°F.

In the event the air in the reactor bay becomes contaminated, it is possible to cut off the recirculation system and provide a 100% fresh air supply with total exhaust to the roof vents.

An air sampler located in the reactor bay will provide a continuous record of the amount of radioactive air contamination. Should this indicate radioactive air concentrations which could prove hazardous by being released to the outside atmosphere, the fan can be stopped and all dampers closed, thus retaining the contaminated air in the bay with the exception of the leakage at the rate mentioned above.

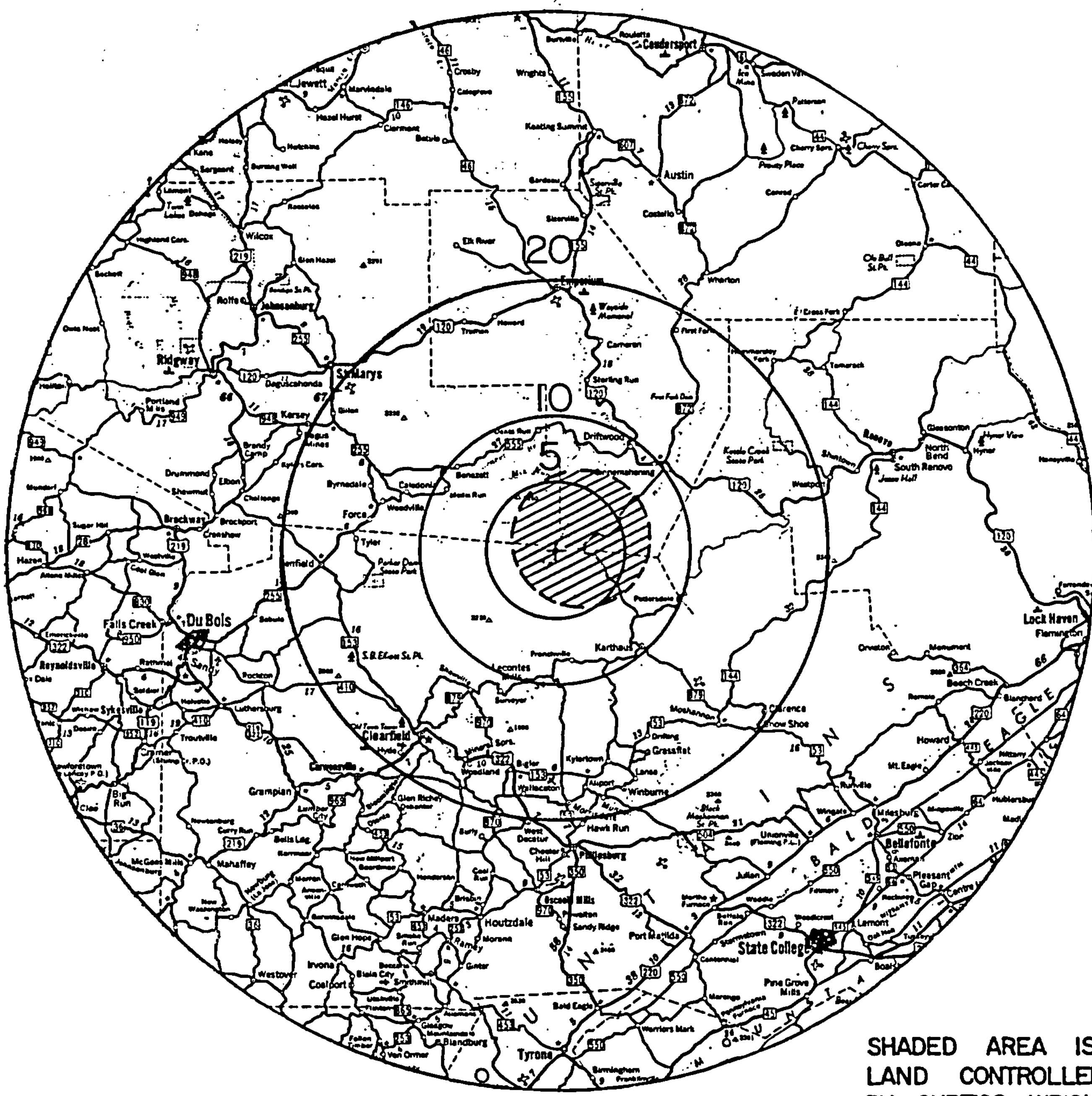
C. Site

The Radioactive Materials Laboratory and Research Reactor is located on an 80 square mile tract of land which Curtiss-Wright has bought or leased for 99 years in order to build a research and development center for the Corporation. This tract, in the form of a 16-sided figure approximating a circle of 10 mi. in diameter, lies in North Central Pennsylvania (see Figure 25). It encompasses portions of Elk, Cameron, and Clearfield Counties.

As Figure 23 indicates, the reactor itself is located in the southwestern quadrant of the circle, a minimum of 3 miles from the boundary of the property. Most of the land immediately outside the boundary is state forest. The site and surrounding area is representative of the general topography of the Appalachian Plateau. The area is relatively flat and lies at an elevation of about 2,000 ft above mean sea level, with the highest point exceeding 2,200 ft. The plateau is cut by deep gorges which dip in some places to 1,100 ft above sea level. The northern half of the site is drained by Red Run and Wykoff Run which empty into the Sinnemahoning Creek. Mosquito Creek and its tributaries drain the southern half of the area into the West Branch of the Susquehanna River at Karthaus.

The reactor is located on the side of a deep gorge through which flows Meeker Run. This one of the larger streams which merges with Mosquito Creek before it leaves the Curtiss-Wright site.

40 MILES FROM REACTOR



SHADED AREA IS LAND CONTROLLED BY CURTISS-WRIGHT

LOCATION OF REACTOR RELATIVE TO ENVIRONS

Figure 25

The Quehanna area shows the physiographic characteristics of the Appalachian Plateau. An initial geologic reconnaissance of the site and the available literature indicate the underlying strata are but slightly disturbed. A generalized geologic section taken from a report by George DeBuchannane, of the U. S. Geological Survey, is included as Appendix I. It appears that in the area of the reactor, there is a layer of coarse-grained sandstone up to 200 ft or so in thickness. The underlying shale formation is probably impervious to fluid flow and is not fractured by folding. The penetration of surface water, therefore, is quite limited and lateral flow would soon return ground water to the streams in the area. The drainage of effluent from the reactor site into local ground waters is considered further in Section IV-B.

The meterological characteristics of the area have been summarized by D. H. Pack of the U. S. Weather Bureau, and are presented in Appendix II.

The countryside surrounding the site is largely uninhabited. The closest towns of any appreciable size are 10 mi from the reactor. Table VI lists the towns within 25 mi radius of the reactor with a population in excess of 200. This area has a population density of approximately 28 people per sq mi.

TABLE VI

Towns Nearest Reactor Site*

	Population		Population
Bald Hill	250	Lanse	350
Benezette	200	Lecontes Mills	600
Bloomington	200	Morrisdale	655
Blue Ball	900	Moshannon	500
Brockport	415	Munson	450
Byrnedale	500	Oak Grove	300
Caledonia	300	Oshanter	300
Chester Hill	954	Orviston	400
Clarence	530	Penfield	600
Clearfield	9,357	Philipsburg	3,988
Crenshaw	550	Quail	600
Curwensville	3,332	Renovo	3,751
Driftwood	289	Saint Marys	7,846
Drockton	250	Sinnemahoning	450
Drury Run	300	Snow Shoe	670
Emporium	3,646	South Renovo	862
Force	506	Sterling Run	225
Frenchville	558	Tyler	300
Glen Richey	425	Wallaceton	440
Hawk Run	651	Weedville	600
Hyde	750	Westport	201
Karthaus	575	Woodland	650
Keewaydin	200	Winburne	700
Kylertown	500		

*From Rand McNally Commercial Atlas and Marketing Guide, Eighty-sixth Edition, 1955. Includes towns within 25 mile radius of reactor.

II. INSURING SAFE OPERATION OF THE REACTOR

A. Reactor Control and Safety Systems

Three safety-shim rods and one regulating rod are used in controlling the reactor. Each of the three safety rods, made of boron carbide and cadmium provides a $\Delta k/k$ of 2.7% for a light water reflector and a Δ k/k of 4.0% for a graphite reflector. The safety rods are magnetically coupled to their respective drive mechanisms. During a scram, the rods fall freely with an acceleration approaching that of gravity. The maximum rate of withdrawal of the safety rods is $6\frac{1}{4}$ in. per minute. At their most effective position, about 50% withdrawn, this speed corresponds to a rate of change of reactivity for any one rod of about 0.021% per second for a water reflector and 0.031% per second for a graphite reflector. The safety rods may be withdrawn individually, or any combination of the three rods may be withdrawn simultaneously as desired.

Since the migration length for light water is so small (6.4 cm), the reactivity effect introduced by a given safety rod in a typical lattice configuration is nearly independent of the position of the other safety rods. Actually, some degree of "rod shadowing" has been observed in inconclusive experiments perfomed in other reactors of similar construction. Because of the samllness of this effect, for a first approximation, it may be assumed that the maximum rate of change of reactivity for all three rods being moved simultaneously is equal to three times the value of one rod, viz., 0.063% per second for a water reflector and 0.093% per second for a graphite reflector.

One stainless steel regulating rod is used in the control system. This rod provides a Δk/k of 0.7% for a light water reflector and a Δ k/k of 1.2% for a graphite reflector. The regulating rod is mechanically coupled to its drive mechanism which has a maximum withdrawal rate of 25 in. per minute. In its most effective position, the maximum rate of change of reactivity of the control rod is 0.018% per second for a water reflector and 0.029% per second for a graphite reflector.

The position of all rods is continuously indicated to within $\pm$ 0.05 in. at the reactor control console by an electrical transmitting system. Micro-switches, mounted on the drive mechanisms, are actuated when the rods are in certain positions. The information from these switches is indicated on a bank of lights on the console. The positions indicated by these lights include upper and lower limits for all rods and shim range for safety rods. In addition, other lights indicate proper coupling of the safety rods to their respective magnets, and "impulse" lights indicate control and insertion or withdrawal.

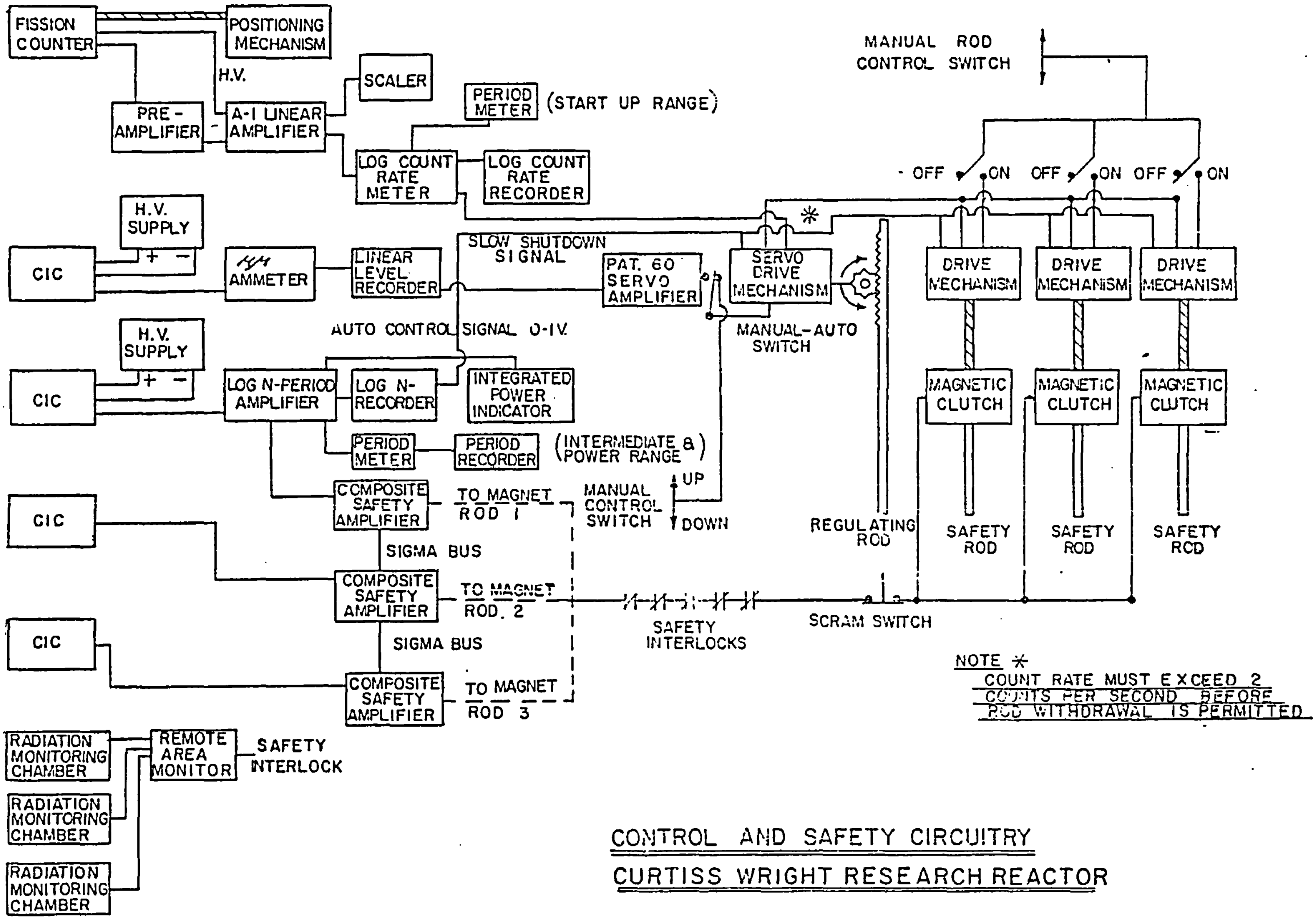
A Leeds and Northrup type PAT 60 Servo Amplifier System is used to automatically control the power level at the desired operating value. This system is interlocked so that it cannot be energized unless the actual power level is greater than 90% of the desired operating value.

Four gamma compensated ionization chambers and a U-235 lined fission counter serve as the flux level detecting devices. A block diagram of these units and their associated circuits is shown in Figure 26.

Two types of automatic shutdown, "slow shutdown" and "scram", provide protection against nuclear incidents. The slow shutdown action drives all control rods to their lower limit at full speed. This slow shutdown feature is designed to be the first line of defense against an incipiently dangerous condition such as a relatively slow rate of increase flux level above the desired operating value. A microswitch in the Log N recorder initiates this action whenever the flux level exceeds 120% of the desired operating value. The reactor "scram" action interrupts the current to the safety rod magnetic coupling devices, causing the safety rods to fall into the reactor under the effect of gravity.

Two methods are provided for initiating the scram - the "fast scram" and the "slow scram." The fast scram action is initiated by electronic interruption of the current to the safety rod coupling magnets and requires a minimum of time to effect this interruption (response time less than 50 milli-seconds). The fast scram is reserved for only the most serious situations, viz., flux level exceeding 140% of desired operating value and reactor period less than 5 seconds. A hazardous condition warranting a fast scram is sensed by any one of three ionization chambers - two sensing linear level and connected directly to their respective safety amplifiers, and the remaining one connected to the Log N system which furnishes a period signal to the safety amplifiers. The slow scram requires an appreciably longer time than the fast scram to disconnect the safety rods from their magnets since the action is initiated by disconnecting the power to the safety amplifiers. The relatively slow decay of the resultant transient thus limits the time required for the current to decrease to a value where the magnets can no longer support the safety rods. As indicated in Table V, the slow scram is initiated by movement of the reactor bridge during operation and by manual operation of the scram pushbotton.

Gamma radiation detectors are located in the beam room and on the reactor bridge. The bridge monitor is interlocked with the reactor safety system (see Table VII) to cause a slow shutdown in the event that radiation levels become excessive. A coolent activity monitor is also interlocked to cause a slow shutdown if a preset activity level is exceeded. This monitor is a delayed neutron detector which will signal the release of fresh fission products. During low power experiments and at power levels up to 100 kw, it will be mounted on the reactor bridge. In order to control the radiation hazard, at power levels above 100 kw, it is necessary to pump water through the core and into a hold up tank to allow the N^{16} to decay. (See Section I-A-6) The delayed neutron monitor, there-



CONTROL AND SAFETY CIRCUITRY
CURTISS WRIGHT RESEARCH REACTOR

Figure 26

TABLE VII

Reactor Safety System

Situation	Detector	Unit Initiating Action	Resulting Action	Annunciation
140% Neutron Flux	Chamber	Safety Amplifier	Fast Scram	Yes
Period-Fast Scram	Chamber	Safety Amplifier	Fast Scram	Yes
Reactor Bridge Moves	Motion Sensing Switch	Switch	Slow Scram	Yes
Manual Scram	Operator	Scram Pushbutton	Slow Scram	Yes
120% Neutron Flux	Chamber	Log N Recorder	Slow Shutdown	Yes
Period-Slow Shutdown	Chamber	Period Recorder	Slow Shutdown*	Yes
Coolant Flow Interrupted	Flow Meter	Pressure Switch	Slow Shutdown	Yes
High Radiation Level	Ionization Chamber	Radiation Monitor	Slow Shutdown and Evacuation Alarm*	Yes
Coolant Activity Excessive	BF_3 Counter	Delayed Neutron Monitor	Slow Shutdown*	Yes
Coolant High Temperature	Resistance Thermometer	---------------	None	Yes
Safety Amplifier Indication	Safety Amplifier	---------------	None	Yes
Low Level Period-30 sec.	Fission Counter	---------------	None	Yes
Deionizer Depleted	Conductivity Cell	---------------	None	Yes
Safety Circuit Bypassed	Relay	---------------	None	Yes
Safety Rods Withdrawn Less than 50% full travel	Switches	Regulating Rod Control System	Prevents Withdrawal of Regulating Rod	No
Startup Count Rate less than 2 counts/sec.	Fission Counter	Log Counter Rate Recorder	Prevents Withdrawal of Safety Rods*	no

*Indicates that safety circuit may be defeated (bypassed) fore, will be removed from the bridge and installed in the pump room just downstream from the hold up tank. At this time, it will be supplemented by a monitor capable of detecting aged fission products.

When the water circulating system is in use, the water temperature will be monitored by a sensing element at the pool outlet. Flow rate will also be monitored and an interruption in the flow will cause a slow shutdown. When the heat exchanger and secondary loop to the cooling tower are installed, additional temperature and flow monitors will be included.

The performance of the deionizer will be monitored by periodically checking the specific resistance of the pool water. This will provide information so that the resins can be replaced before becoming seriously depleted. After the heat exchanger has been installed and operation at 1000 kw for extended periods is possible, an area monitor will be installed in the pumproom to warn personnel of excessively high radiation levels before they enter the room.

B. Administrative Controls

The likelihood of an accident involving a nuclear reactor can be greatly reduced by the clear definition of responsibility for the various phases of operation. The prime responsibility for the safe operation of the reactor, therefore, will be assigned to one man, the head of the Reactor Operations Section of the Research Reactor Division.

In order to prevent any compromise of safety, the responsibility for experimental work will be completely divorced from operational considerations. Experimental work utilizing the reactor will not be conducted by personnel of the Reactor Operations Section. The experimental program will be coordinated by the Chief of the Research Reactor Division who will decide what experiments will be conducted and with what priority. As a check on the experimental program, all proposed, potentially hazardous projects will be referred to the Curtiss-Wright Reactor Safeguards Committee for review.

The duties of the individuals and committee mentioned above, as well as the duties of reactor shift supervisors and operators, are as follows:

1. Chief, Research Reactor Division

a. Review all requests for reactor time from the standpoint of technical feasibility and desirability, and determine the relative priority of the proposed programs which he actually approves. Approval must be obtained before any experiment or irradiation affecting reactor operation in any way may be performed.

b. Advise the originator of a request for reactor time concerning the reason for disapproval, if this action was taken.

c. Refer approved requests for reactor time to the head of the Reactor Operations Section for his approval.

2. Head, Reactor Operations Section

a. Assume primary responsibility for safe operation of the reactor, and insure that experimental requirements do not unduly compromise safety.

c. Supervise all reactor operating personnel.

d. Supervise training of new operating personnel and licensing of new operators.

e. Keep all records and logs of reactor operation up to date and in proper form.

f. Issue regular monthly reports on reactor operations.

g. Take appropriate corrective action to prevent reoccurrence of any significant malfunction, violation, or accident in connection with reactor operations. Issue a special report to the Curtiss-Wright Reactor Safeguards Committee (CWRSC) covering any such incident and the corrective action taken.

h. Co-operate with Research Reactor Division Chief in scheduling and co-ordinating experimental programs and service irradiations.

i. Review requests for reactor time which have been approved by the Division Chief. Approval or disapproval shall be based strictly on considerations of operational safety rather than the merits of the experiment. After action is taken, the request shall be referred to the CWRSC.

3. Curtiss-Wright Reactor Safeguards Committee (CWRSC)

The Curtiss-Wright Reactor Safeguards Committee shall consist of members having extensive training and experience in at least one phase of reactor operations or experimental work, utilizing reactor radiation. The chairman of this committee shall be the Assistant Manager of the Nuclear Power Department. The responsibilities of this committee shall be as follows:

a. Review all requests for reactor time which are forwarded to it. This review shall encompass only matters concerning health and safety, and shall not touch upon the technical feasibility or advisability.

b. Approve, provisionally approve with recommendations for change in the program, or disapprove all properly submitted requests, and advise the interested parties of the outcome of the review.

c. Review special reports issued by the Reactor Operations Section Head following any significant malfunctions, violations, or accident. In addition to this review, the Committee shall either approve the corrective action already taken, or recommend further action.

d. Keep informed concerning reactor operations by studying the monthly reports issued by the Reactor Operations Section Head.

4. Reactor Engineer

a. Accept responsibility for the safe operation of the reactor at all times during his shift except when relieved by the Reactor Operations Section Head.

b. Make all minor decisions regarding operation of the reactor during his shift, and all decisions required immediately.

c. Manage the reactor fuel inventory, specify element rotation and reprocessing schedules, keep records and issue reports concerning the fuel inventory.

d. Remain in the reactor building at all times during his shift; supervise routine startup, shutdown, alteration in power level, movement of overhead bridge crane, and any movement of any object in that portion of the pool in which the reactor is operating at the time.

e. Carry out appropriate checks of the safety circuits and supervise routine maintenance.

f. Supervise the keeping of records and logs and insure that all records for each of his shifts are complete and accurate.

g. Instruct operators and operator trainees in the theoretical and practical phases of reactor operation and maintenance.

h. Relieve the operator at the console from time to time, and give all necessary assistance to the operator and any experimental personnel working with the reactor.

5. Reactor Operator

A reactor operator shall be a person holding a valid AEC operator's license. His duties shall include:

a. Manipulation of the reactor controls under the direct supervision of the director of reactor operations or a supervisor during startup, shutdown, alteration of power level.

b. Manipulation of controls and surveillance of instrumentation during steady state operation.

c. Remaining on the reactor bridge at all times during operation unless properly relieved by the shift supervisor or another operator.

d. Keeping all such records and logs as shall be required.

e. Advising the supervisor of any unusual behavior on the part of the reactor and its controls, and taking any necessary action to prevent damage to the reactor and protect health.

C. Operational Controls

1. General

The complete understanding of and compliance with a well conceived set of operation instructions by all operating personnel will greatly reduce the probability of a reactor accident. While not insuring against human error, these rules reduce the chance of an error causing trouble. A number of general rules governing operation of the reactor are set forth in the following paragraphs.

No one except a licensed operator may manipulate the reactor controls. The only exception will be an operator-trainee who may operate the reactor when a reactor engineer is present in the reactor bay.

Loading or unloading of the active lattice, or movement of the reactor bridge may be done only under the direction of a reactor engineer. This will be enforced by keeping the bridge and fuel element handling tools locked in place with the only keys in the possession of the reactor engineer.

In loading any configuration for the first time or following any significant change in nearby experimental equipment or specimens, the reactor will be brought to criticality by means of a critical experiment under the direction of a reactor engineer.

Following the loading of a configuration previously logged, the approach to criticality will be made under the direction of a reactor engineer, but need not be done by means of a critical experiment.

Announcement of the intention to start up the reactor will be made over the public address system, as well as announcement of the final power level when this is attained.

The reactor will always be operated with the minimum possible excess reactivity loaded into the core.

2. Critical Experiment Procedure

When a new configuration of fuel and/or reflector elements is to be used in the reactor, source multiplication in the core will be measured after each element is added. The data obtained will be plotted (as it is obtained) to allow prediction of the point at which the reactor will go critical. In the case where a large sample or experiment is to be positioned in or near the core, the reactor will be unloaded, the sample or experiment positioned, and the same procedure used to approach criticality. The steps in the procedure are as follows.

a. An estimate of the critical mass of the projected loading will be made.

b. The control rod fuel elements and rod drives will be installed in the desired positions.

c. The reactor checkout procedure will be carried out, as for a reactor startup. Note that during the initial stages of the experiment, it will be necessary to bypass the 2 cps interlock between "Count Rate" and "Magnet Current." This will be done under direct supervision of the person in charge of the critical experiment.

d. The regulating and safety rods will be raised to the 50% point.

e. A source will be installed and approximately 50% of the critical mass calculated in step (a) will be loaded, with a constant watch on the count ratemeter. Whenever fuel elements are loaded or unloaded, fuel element numbers and positions will be carefully recorded both in the log book and on the loading chart. At this point, the count rate in the fission chamber channel will be determined using the scaler, to give a measure of the source multiplication.

f. The rods then will be fully withdrawn and another count made. Then the rods will be driven back to the 50% point.

g. One additional fuel element will be loaded, and the measurements of steps (e) and (f) repeated. This data will be plotted to give the "Subcritical Multiplication Curve" as soon as it is obtained, before any further loading is done. The curve obtained from plotting the data taken with the rods fully withdrawn gives an indication of when it will be possible to make the reactor critical by withdrawing rods. The data taken with the rods at "50%" gives a curve which indicates the possibility of going critical during the actual loading operation.

h. Step (g) will be repeated until the reactor goes critical at which point rod positions will be recorded. If the reactor goes critical without sufficient excess reactivity for operational use, the loading will be continued, using the "50%" Subcritical Multiplication Curve to insure the criticality is not reached during loading of an element. This completes the critical experiment and at this point, the reactor will either be shut down or operated, as specified by the Reactor Supervisor. At the completion of the experiment, fuel handling tools will be locked and the plots of the data obtained and the loading chart will be attached to a page in the log book.

The person loading fuel will maintain a position which will allow instant reversal of motion of the fuel element if the operator at the console orders it. The loader will maintain positive control over the fuel element until the operator specifically gives permission to release it.

3. Start-Up Preparation

The reactor will not be operated if any instrument or device associated with the control and safety circuitry is not functioning properly. Immediately before the reactor is started up, the operator will go through a checklist which will give definite assurance that all systems are operating correctly. After each step on the checklist is completed, the operator will record the readings made, or in cases where no reading is required, will simply initial the appropriate blank on the Reactor Checkout Procedure Form. A copy of this checklist is included as Appendix IV. After completing the checklist, the form will be signed by the operator and the shift supervisor and then filed.

4. Start-Up (Cold, Clean)

Under conditions of cold, clean start-up, the following procedure will be followed:

a. Make sure that the Reactor Checkout Procedure form has been properly completed and signed. Enter the following information in the Reactor Log Book: date, time, purpose of startup, core configuration, name of operator and supervisor, and list of samples in or near the core.

b. Raise the safety-shim rods 6 in. and inspect the core to make sure that rods and rod drives are operating properly.

c. Announce intention to start up to entire building over the intercommunication system.

d. Raise safety-shim rods until shim range is reached, closely watching the low level period and count rate recorders. If it appears that the reactor will go critical before the safety-shim rods reach the shim range, lower all rods and notify the supervisor.

e. Raise the regulating rod to the 50% point, watching meters as before. This places the regulating rod in the optimum position for control.

f. Raise all safety rods in small increments until the reactor is critical, as indicated by a slow but steady increase in count rate. The reading on the low level period meter should not be shorter than 20 seconds. Note: As the count rate meter approaches its upper limit the rod motion should be stopped and the fission counter repositioned to keep the meter on scale.

g. When the reactor power level has increased to the point where indication is obtained on both the LOG N and LINEAR LEVEL recorders, the safety-shim rods may be withdrawn slightly to obtain a shorter period (never shorter than 20 seconds).

h. As power level increases, adjust fission chamber height to keep the "Count Rate" recorder on scale.

i. As power level increases, adjust MICRO MICROAMMETER to keep the LINEAR LEVEL recorder on scale.

j. When desired power level is reached (as indicated by the LINEAR LEVEL recorder), stabilize the reactor manually using the SAFETY-SHIM RODS and then place the reactor on "AUTO" control.

5. <u>Start-Up (high Residual Power Level)</u>

Under conditions of high residual power level, start-up should only be attempted with the utmost caution. Between increments of rod withdrawal, adequate time <u>must</u> be allowed so that equilibrium conditions can be observed.

6. <u>Procedures During Operation</u>

At hourly intervals, the operator will record in the log book:

a. Recorder indication
b. Rod positions
c. Indication on Radiation Monitoring System
d. Coolant flow and temperature

7. <u>Shutdown</u>

For routine shutdown, intention to shutdown will be announced over the PA system and the reactor taken off of "AUTO."

a. Routine shutdown will be accomplished by driving in the safety and control rods (not by dropping rods).

b. As the rods are being driven in, the operator must stand by to stop the motion of the safety rods if there is an indication of jamming by the "JAM" annunciator. As soon as the rods are fully inserted, the rod drive switches must be restored to the neutral position.

c. As the rods are being inserted and the power level is dropping, the operator will follow it by changing the micro-microammeter switch, the scaler switch, and the fission chamber height.

d. When the rods are completely inserted, the operator will

1) Turn off recorders.
2) Turn scaler "COUNT" switch to "OFF" position.
3) Move fission chamber to fully inserted position.
4) Set micro-microammeter switch and scaler switch to most sensitive position.
5) Perform necessary duties in connection with experiments and and sample irradiation as directed by the supervisor.

6) Make sure log book and all records are complete.

e. The Magnet Power Key Switch will be turned off, and the key removed and given to a reactor engineer only. The fuel handling tool and bridge handwheel are to be padlocked and the underwater lights turned off.

D. Emergency Power Plant

In any reactor system, it is necessary that electrical power be supplied to certain sections at all times. This necessitates an auxiliary generator, engine driven, which will automatically supply the required power whenever there is a failure of the normal supply.

In the Curtiss-Wright installation, the main emergency power system consists of a generator rated 43.75 kva, 35 kw, .8 PF, 227/480-volt, 3-phase 4-wire, 60 cycle. The generator is driven by a propane gas fueled engine delivering a rated output of 81.5 hp at a speed of 1,800 rpm. The propane fuel is obtained from the building gas system.

An automatic transfer panel transfers the load from the primary source to the emergency source when the line voltage falls below 85%, and returns the load to the primary source when the line voltage has been restored to 95% or more of normal. Time delay relays are incorporated in the transfer switch which permit the emergency unit to reach rated voltage and speed before the transfer is effected. This takes approximately 10 seconds.

When primary power is again available, this power must be applied to the supply bus for 15 minutes before the critical load is automatically transferred from emergency to primary power. The automatic transfer switch is electrically and mechanically interlocked so that there is no feed-back from the supply bus to the emergency generator, or vice versa. Full relay protection guards against phase failure. A built-in test switch to simulate power failure is provided for maintenance checks and testing. An on-off switch on the generator set permits operating the engine without interrupting the normal source of supply. A four position control switch on the control panel permits selection of four operating positions marked "stop," "handcrank," "test," or "automatic."

During a normal power failure, it is necessary to provide emergency lighting, instrumentation, and heating and ventilation for the hot cells. Emergency lighting is supplied in most rooms in the building and mainly consists of one electric bulb per room. This does not apply to the hot cells where 50% emergency lighting is provided so that an experiment, which is at a critical stage when normal power fails, may continue.

E. Shielding from External Radiation Hazards

The points of closest approach to the core, normally accessible to operating personnel, are the reactor bridge and the beam hole room. A minimum of $19\frac{1}{2}$ ft of water will normally cover the center of the reactor. According to data from the BSR (Ref. 1), the dose rate due to the penetration of gamma rays from the core at 1000 kw will be about 7 mr/hr at the water surface. The neutron flux will be negligible.

Continual recirculation of the pool water through a mixed-bed ion exchanger at 15 gpm will maintain the concentration of dissolved substances far below that which could present a problem from the standpoint of external radiation hazard. When the reactor is operated at more than a few watts, the water in the coolant recirculation loop will be continually monitored and activity buildup will be detected long before it becomes an external hazard. The continual measurements of water conductivity will serve as an additional check on purity, and hence, induced activity. A small amount of activity will accumulate on the ion exchangers. However, it is not anticipated that this will normally become excessive. Replacement of the cartridges will be necessary occassionally because of exhaustion of the ion exchanger capability of the resin. This operation will be done under strict health physics supervision. During high power operation, the background in the equipment room will be continually recorded by a remote monitoring system, and also checked daily by a health physicist.

The 7-sec N^{16} activity induced by fast neutron irradiation of the water presents a problem at high power levels. For example, at 1000 kw, N^{16} will be produced at a rate of about 2 curies/sec. However, experience with BSR has shown that the reactor can operate up to 100 kw with free convective cooling without running into appreciable dose rates from the N^{16}. At higher power levels, the water rises rapidly by convection from the core, significant quantities of N^{16} reach the surface, and the dose rate around the pool becomes excessive. To prevent this, at power levels greater than 100 kw the reactor will be force-cooled by pulling water through the core from top to bottom and pumping it directly into a hold-up tank at 700 gpm. The details of the cooling system have been described previously. If the flow of cooling water drops significantly, the reactor would be automatically shut down. The flapper valve at the bottom of the plenum would then be opened manually to allow free connective cooling of the core, thus allowing afterheat to be removed without boiling.

The 2000 gal hold-up tank will provide sufficient time delay for essentially all of the N^{16} to decay before the water enters the pump room. Since the tank is buried it presents no external radiation hazard.

Shielding for the beam hole room is considerably more than adequate from the biological point of view because of the need to keep instrument backgrounds as low as possible. The core is restricted so that it cannot approach the pool wall closer than 4 ft. In addition to this thickness of water, the shield consists of 18 in. of normal density concrete, and 4 ft of ferrophosphorous concrete with a density of at least 4.5. This will reduce the gamma dose rate, due to radiation from the core at 1000 kw, to less than 0.05 mr/hr at the shield surface. There will be essentially no penetration of the shield by neutrons. Secondary gamma radiation due to interaction between neutrons and the shielding material will be negligible compared to that which arises in the core itself. The result, then, is that there will be essentially no neutron flux in the beam hole room, and a gamma flux of the same order of magnitude as the natural background.

The above considerations apply, of course, to the situation when all beam holes are plugged. During experiments in which it is necessary to have gamma and/or neutron beams emerging into the room, special precautions will be taken. Suitable "beam-catchers" will be used to limit the length of travel as much as possible. Careful surveys will be made of the resulting radiation fields and the exclusion areas (those in which the radiation field exceeds MPL) will be suitably roped off and posted. In addition, access to the beam room is normally available by means of a stairway leading from the upper operating level. Entrance to the room therefore, is, easily controlled. Radiation levels will be continually monitored by several detectors strategically located in the beam room.

F. Security and Fire Protection

Access to the reactor building is subject to limitation at several points, so that it may be considered a highly controlled area from the security standpoint. Only Curtiss-Wright employees, all of whom have received some degree of security clearance, or properly authorized visitors may enter the corporation's property through one of the several gates. After gaining access to the property, a person must travel more than 7 mi to reach the reactor building.

The Radioactive Materials Laboratory, which houses the reactor, and the Waste Disposal Building are enclosed by a chain link fence topped by three strands of barbed wire reaching to the height of 9 ft. The layout of the buildings, fences and approaches are shown in Figure 20. During a normal operating shift, Gate 1 will bs open so that personnel may enter the lobby directly from the outside. A guard will be stationed in the lobby and will permit only employees or properly authorixed visitors to enter the building unescorted. All other individuals must be under constant surveillance by an authorized escort. Entrance to the building by another door normally will be prevented by the fence and by keeping all other outside doors locked. The number of persons authorized to open other doors will be restricted.

During off-shift hours, Gate 1, as well as the main entrance, will be locked. The building will be checked periodically by a roving watchman punching a number of watch clocks. Surveillance of the building after dark will be aided by extensive flood lighting of the building exterior and the surrounding area.

Fire fighting equipment is installed in and about the building in accordance with the requirements of the National Board of Fire Underwriters. Most sections of the building, in which radioactive work is not carried on, are protected by an automatic sprinkler system. When any part of this sprinkler system is actuated, an alarm will sound throughout the building. It is also arranged that an alarm will sound if the water pressure in the sprinkler system drops below a preset level.

It is not practical to use sprinkler systems in most areas where chemical and radioactive work take place because the reagent which should be used to put out the fire depends largely on the material which is in the laboratory and that which is burning. In such areas, therefore, automatic fire detectors have been installed. If the temperature in one of these areas rises above a preset limit, it will result in the continuous ringing of all fire alarm bells throughout the building. Automatic detectors are located in the reactor bay, remote control room, reactor pump room, mezzanine fan rooms, operation area, above the isolation rooms, the service area, decontamination room, and the radiochemistry laboratory.

The entire fire alarm system will operate from the normal power bus. In case of a power failure, the system will automatically switch over to a 24 volt D.C. supply obtained from storage batteries kept charged by means of a trickle charger. Provision is made for the system to be connected to a future central fire station.

In the areas in which only fire detectors are installed it will be necessary to combat fires with locally available fire extinguishing apparatus. This may include water, foam, carbon dioxide, or powdered sodium chloride. The reagent available, as mentioned above, will depend upon the type of fire anticipated.

There is no provision for fixed sprinkler or automatic fire detection equipment in the hot cells. Each experimental installation is evaluated individually for an associated fire hazard, and appropriate alarm and fire extinguishing apparatus is installed with the experimental equipment when it is advisable.

Around the outside of the building, there are three fire hydrants. The first is located approximately 60 ft from the northeast corner of the building, the second about 100 ft from the northwest corner, and the third, a pumper hydrant and hose-reel house which contains 200 ft of $2\frac{1}{2}$ in. hose and 300 ft of $1\frac{1}{2}$ in. hose. This is sufficiently long to reach any section of the building.

An electrically driven pressure pump on the fire protection pumping system starts at 95 psig and stops at 100 psig and will supply 500 gpm at 100 psi. A booster pump cuts in if the pressure in the fire lines drops to 85 psig and cuts out again when the pressure reaches 110 psig. In case of an electrical power failure, a propane engine driven pump will cut in automatically when the water pressure drops to 75 psig but must be stopped manually.

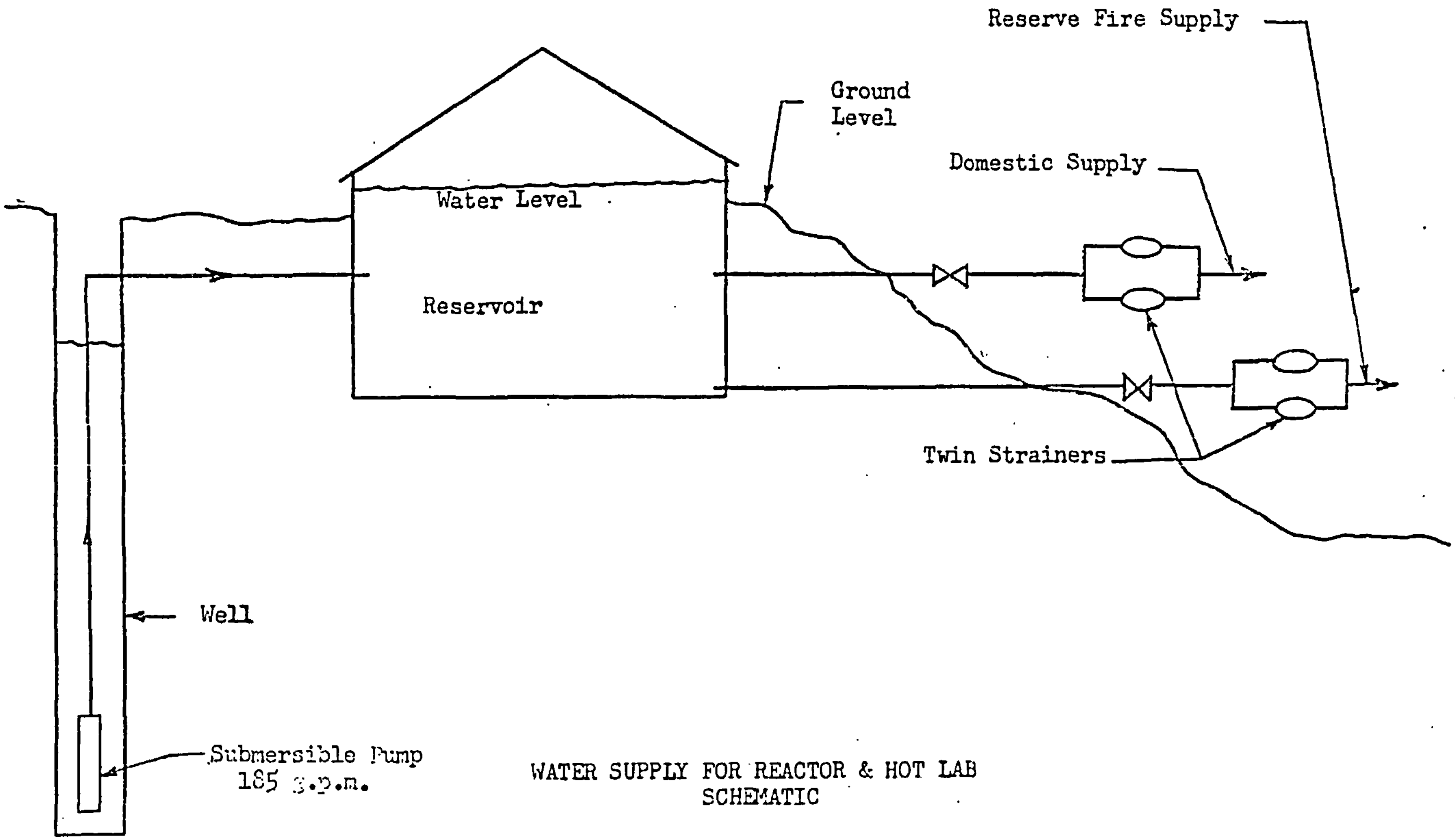
Reference to Figure 27 shows how water is obtained from a surface storage reservoir which is covered by an aluminum structure and which holds not less than 135,000 gal of usable water volume when the surface is not frozen. The domestic water suction line removes water from a higher elevation in the reservoir than the fire protection suction line so that in case of a water draw-down, there will always be 50,000 gal of water available for fire protection.

A water supply for the reservoir is obtained from a spring-fed well. A submersible pump set at a depth of 460 ft discharges its effluent to the reservoir at a point located below the frost line. The pump is rated at 185 gpm with a 500 ft head and is operated automatically by means of a high and low level control unit. Manual controls are located in the main pump room.

Provision has been made for combating forest fires. A security ranger with extensive experience in forest management is a permanent member of the staff, and performs, as a major duty, the prevention and management of forest fires. Mobile equipment and a well trained emergency crew are available at all times. In addition, the state will supply man power and equipment as needed. An on-site meteorological program enables the rangers to determine the likelihood of forest fires and take appropriate measures. The reactor building is surrounded by a cleared area several hundred feet wide which serves as a fire break. Because of the availability of water to hose down the building and the fire resistance of the exterior construction materials, it is felt that a severe fire in the surrounding woods would do essentially no damage to the facility.

G. Fuel Management

Upon receipt of the fuel elements from the fabricator (the first set of elements is being supplied by the Sylvania-Corning Nuclear Corporation), they will be stored in a safe configuration in racks mounted along the pool walls. Tampering with the elements or possible theft will be prevented by the security measures restricting entrance to the building and by keeping the handling tools locked up when not actually in use. The best deterent against theft after a few kilowatt-hours are logged will be the residual activity of the fuel elements themselves.



WATER SUPPLY FOR REACTOR & HOT LAB
SCHEMATIC

Figure 27

According to the operational time-table as it is now envisioned, the reactor should go critical about the end of January, 1958, reach a power level of 1000 kw about mid 1958, and begin routine operation for extended periods at 1000 kw early in 1959. It is anticipated that fuel element burnup of at least 10% can be achieved. The actual limit which is set will be determined by experimental requirements and the rate of corrosion of the fuel plates. If the above timetable proves correct, the first replacement of fuel elements will occur sometime around mid-1959.

A careful log will be kept up to date so that the actual percent burnup of any given fuel element will always be known. Elements will be rotated from high to low flux areas and to the gamma irradiation facility so that uniform burnup is achieved. In this way approximately 20 fuel elements will become ready for replacement at one time. Following the initial replacement, fuel elements will burn out at the rate of about 20 every 6 to 12 months.

Burned out elements will be transferred to semi-permanent storage in the pool racks or to the gamma irradiation facility in the 15 ft deep pool provided in the hot lab service area. These elements will be stored for a cooling period of up to 60 days. Then they will be loaded under water into suitable shipping containers and sent to a reprocessing plant.

H. Emergency Procedures

Emergency procedures will be published to anticipate as many credible accidents as possible. These procedures will have as their object the rapid mobilization of manpower to cope with the situation at the site and to take whatever precautions are necessary off-site.

A communications center is to be set up at the Quehanna site some 5 mi from the reactor. This center will act as a command post for directing emergency operations. It will be in communication with the reactor building by telephone (three lines) and, if these lines should be destroyed, by radio. Any incident in the laboratory will be announced over the building loudspeakers. Following such an announcement, the communications center will be notified of the incident by telephone or radio. The communications center will then notify health physics, plant protection, and other organizations which will act quickly according to predetermined plans. The reactor laboratory is also equipped with a siren which serves as an evacuation alarm. It is connected to the radiation monitor on the reactor bridge and sounds automatically whenever a preset radiation level is exceeded. Every attempt will be made to ascertain quickly the extent of any release of activity of the environment. Vehicles equipped with detection equipment will be available at the reactor site and at the communications center. Under stable atmospheric conditions in which some off-site exposure might occur, there will be ample time to warn people in the cloud's trajectory, which almost always will be along the Mosquito Creek. There are roads which lead to the Mosquito Creek Valley at about 5, 8 and 13 mi downstream. Karthaus lies about 15 mi downstream. During periods of high wind velocity there is little danger of off-site personnel receiving substantial doses. The prediction of the cloud's track will be possible from meteorological data available at the site.

In the event of release of activity into ground water, the path of flow is well defined and appropriate steps could be taken quickly through state health authorities. Since all streams in the area which serve as municipal water supplies are already chemically contaminated, water for drinking purposes would necessarily have to be treated. Therefore, intake of water from the radioactively contaminated stream could easily be prevented for the short time necessary.

I. Health Physics

All activities involving radioactive materials or radiation sources will be reviewed continually by a permanent health physics organization. The health physics program to be carried out in connection with reactor operations includes area and personnel monitoring.

Personnel monitoring will include the issuance of film badges for a permanent exposure record, as well as pocket dosimeters when this seems advisable. Persons who may be in a neutron field in the beam room will also wear neutron monitoring devices (pocket chambers and nuclear emulsions). A minimum of one health physicist will be on duty at all times during reactor operations. He will be available for advice on radiation problems, and will monitor all non-routine activities in which there is a possibility of exceeding the maximum permissible levels of exposure.

III. ACCIDENTS INVOLVING THE REACTOR

A. Reactor Power Excursions

1. Reactivity Requirements for Reactor Operation

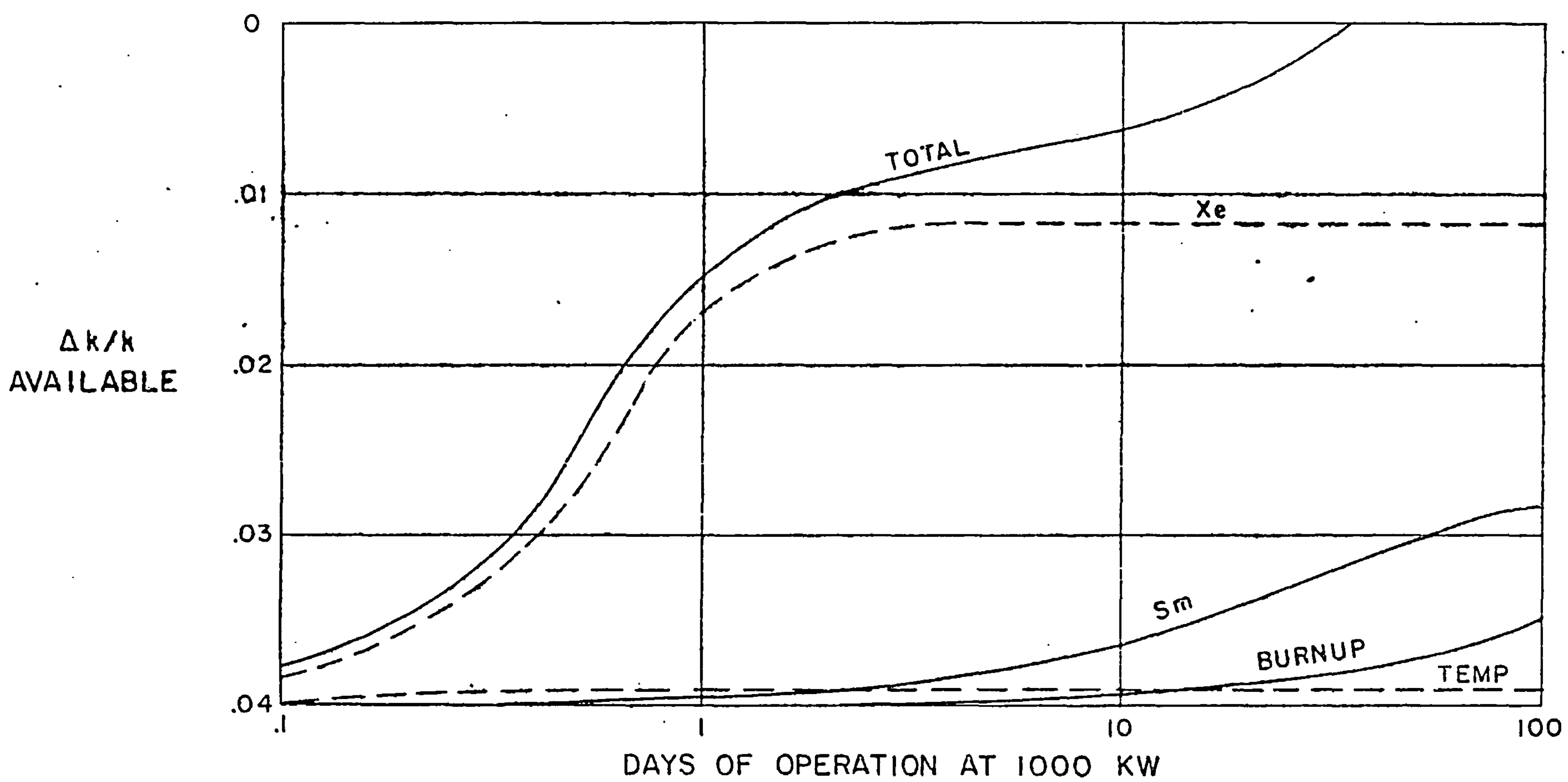
One source of danger inherent in the operation of a reactor of the swimming pool type is the sudden introduction of a large amount of reactivity. It is pertinent, therefore, to state the maximum amount of reactivity which would ever be available for rapid addition. Table VIII lists the reactivity requirements for operation at 1000 kw.

TABLE VIII

Estimated Reactivity Requirements

Source	Reactivity Required
Negative temperature coefficient	.001
Poisons (Xe, Sm, etc.)	.032
Experimental requirements	.002
Adequate rate of change of power level	.002
Addition of smallest increment of reactivity available	.003
Total Requirement	.040

As indicated in Table VIII, the reactivity available for experiments has been limited to 0.2%. Experiments requiring more than this will be set up with the core unloaded. The reactor will then be brought to the desired power by means of a critical experiment. There is, therefore, no need for a large amount of "built-in" reactivity to compensate for the introduction of experiments which are important to the neutron economy of the core. The change in reactivity available in the core as a function of operating time at 1000 kw is shown in Figure 28. It is apparent that starting with 4.0% $\Delta k/k$, the reactor can operate continuously for more than 10 days and still have sufficient reactivity available for control purposes and experimental requirements. Since the reactor will be shut down at least as often as one day per week for routine maintenance, adjustments in the fuel loading can be made frequently to compensate for burnup, samarium poisoning, etc. Because of the relatively low poison build up following a shutdown, no difficulty is anticipated in overriding these poisons (primarily xenon) with the reactivity available within the 4.0% limit.



CHANGE IN AVAILABLE REACTIVITY AS A FUNCTION OF OPERATING TIME

Figure 28

It is of interest to note that the reactivity available decreases very rapidly during the first few hours of operation at 1000 kw due primarily to xenon buildup. As a result, $\Delta k/k$ is reduced from 4.0% to less than 1.5% in the first day of operation. For most of the operating lifetime of the reactor, 1.5%, rather than 4.0%, can be considered the maximum $\Delta k/k$ available to produce a power excursion.

2. Reactivity Worth of Beam Tubes

Another possible method of causing a stepwise introduction of a significant amount of reactivity would be a sudden substitution of reflector material for a void next to the core, e.g., the flooding of a beam tube. A multigroup diffusion theory analysis has shown that the total worth of the reflector covering one entire side or face of the core is only 1% $\Delta k/k$.

Values for the worth of a 6-inch beam tube are available in the literature. Experiments at ORNL (Ref. 2) give a value of 0.36% $\Delta k/k$ for the void plus aluminum with a water reflector, and 0.25% with a BeO reflector. Work at Pennsylvania State University (Ref. 3) has provided values for the void alone of 0.38% for a centered 6-inch tube and 0.10% for an offset 6-inch tube. The total worth of the aluminum in three tubes was found to be 0.25%. The estimated total worth of the beam tubes in the Curtiss-Wright Research Reactor with a water reflector are as follows:

	Centered 6 in. Tube	Offset 6 in. Tube
Al + void	0.54	0.15
Al	0.16	0.05
Void	0.38	0.10

The total worth of the three tubes is approximately 0.84% $\Delta k/k$. This value is consistent with the results of the multi-group calculation for the worth of an entire core face.

3. Results to be Expected from Excursions of Various Magnitudes

In order to evaluate the potential hazard represented by a 1000 kw swimming pool reactor, it is necessary to predict the results of two types of accidents, viz., a sudden stepwise introduction of a given amount of reactivity into the core, and the introduction of reactivity at a steady rate until some given total amount has been added. The latter will henceforth be called a ramp addition. The 1953-1954 Borax-1 experiments (Ref. 4) and the more recent Spert-1 tests (Ref. 7) have yielded considerable data on the results of both types of accidents. Unfortunately, however, available data are incomplete and not directly applicable to swimming pool reactors. Much of the data was obtained with the water moderator at saturation temperature rather than sub-cooled as is the case for pool-type reactors of the Curtiss-Wright Research Reactor type.

Moreover, the data were obtained with a 2-4 foot head of water rather than the 18-22 foot head generally required by pool-type reactors. Despite these disparities, the Borax and Spert experiments were conducted with reactors which were sufficiently similar to the Curtiss-Wright Research Reactor (CWRR) to allow some useful conclusions to be drawn.

With the water moderator at saturation temperature and a 2 foot head of water, the Borax-1 reactor withstood a step increase of 2% $\Delta k/k$ without damage to the fuel plates. When the water was sub-cooled to ambient temperature (approximately 80°F) the initial excursions became more severe. For a given Δk, the energy released in a subcooled excursion was of the order of five times as great as for the saturated case. Nevertheless, the reactor still could withstand a step $\Delta k/k$ addition of 1.5% without mechanically damaging the fuel elements or approaching the melting point of aluminum cladding. According to Ref. 10, the Borax experiments "prove that the reactors investigated possess a high degree of inherent safety." Perhaps the best demonstration of their inherent safety was the final destructive Borax-1 experiment. This test was initiated by a 4% step reactivity addition with the water at ambient temperature. The resulting excursion destroyed parts of the reactor, and melted most of the fuel plates. However, most of the active debris was accounted for within a 100 yd radius about the site. Thus, despite having no containment whatsoever, no large fraction of the reactor material left the immediate vicinity of the reactor in the form of air-borne material.

The step experiments with subcooled moderator were continued with the Spert-1 reactor. This reactor is quite similar to Borax-1. It was found that Spert was capable of shutting itself down without damage following step additions of up to 1.5% $\Delta k/k$. Although the Spert-1 reactor tolerated step function increases in k as well as the Borax reactor, the detailed behavior of the two reactors showed some striking dissimilarities following the initial power surge.

The Spert-1 reactor was also used to study response to ramp additions of reactivity, i.e., the control rod was withdrawn at a constant rate until a pre-determined $\Delta k/k$ had been added. The first of these tests were carried out with the water at room temperature, a ramp rate of 0.35% per sec, and total reactivity additions up to 2.5%. The reactor was undamaged although divergent oscillations appeared toward the end of the runs when more than 2.25% had been added. As indicated by Ref. 6, this was not expected since the reactivity was being increased continuously. Additional tests on long term stability at room temperature were conducted with ramp rates of 0.09% per second. Total additions up to 2.25% $\Delta k/k$ at this rate resulted in stable behavior with only mild oscillations after the ramp addition was completed. A change in hydraulic head from 2 to 4 ft did not appreciably effect this behavior. When the ramp tests were attempted with boiling moderator water, instability became apparent at relatively low reactivity additions.

The Borax-1 and Spert-1 tests, the early calculations of Claiborne and Poppendiek (Ref. 8), and Edlund and Noderer's analysis (Ref. 9) indicate that with the moderator subcooled the CWRR could withstand a step function increase in reactivity of as much as 1.5%, and a ramp addition of almost 2.5% at a rate of 0.09% per second without mechanically damaging the fuel elements or approaching the melting point of the elements. Increasing the water temperature to the boiling point would increase the maximum step input of reactivity the core could withstand, but would decrease the ability to accept large ramp increases without goint into diverging power oscillations. It is planned to operate the CWRR at a bulk water temperature of 90°F.

4. Conclusions

As indicated above, a step function addition of 1.5% Δ k/k can probably be tolerated by the CWRR. It is very difficult to imagine how an accident of this magnitude could occur. The total worth of the reflector covering an entire reactor face would not be disastrous. The flooding of a single beam tube would introduce considerably less than one dollar of reactivity. Small experiments of the type which could conceivably be removed rapidly will be limited to a total worth of 0.2%. Larger experiments will be installed with the reactor shut down, and start-up will be by means of a critical experiment. In such cases, special precautions will be taken to insure that the experiment and the core cannot be separated suddenly. No experiment with a total worth of more than 1.5% Δ k/k will be installed under any circumstances.

Experiments at the Pennsylvania State University Reactor have shown that the worth of an outside fuel element is less than 1.5%. If, through a compounding of human errors, an additional full fuel element were placed next to a barely sub-critical core the resulting excursion would not result in damage to the fuel elements. Since the core will always be loaded from the "inside out" the chance of an element being placed in a central position in a nearly critical core is extremely remote. The more likely accident of this type, viz., a jammed control rod withdrawing a fuel element from the core and allowing it to fall back suddenly, is prevented by the control rod guide tubes which rest on top of the control rod fuel elements and are securely fastened in this position to prevent vertical motion.

An accident caused by a ramp addition of reactivity might be imagined through some combination of human and instrument failure. Difficulty with the servo control system might result in a sustained withdrawal of the regulating rod. However, the maximum rate of addition of reactivity that this could cause would be 0.029% per second and the total Δ k/k added would be 1.2%. These values apply to the worst case, viz., the graphite reflected core. They are well within the 0.09% per second rate and 2.5% total which, as stated above, could be tolerated without damage to the core.

A less likely, but more serious, accident would be the withdrawal of the three safety-shim rods simultaneously and continuously. In the worst case, i.e., a graphite reflected core and rods at their most effective position initially, the maximum rate of change is 0.09% per second. At this rate, it would take over 25 seconds of continual withdrawal of all three rods to approach the danger point. This would be ample time for the operator to take appropriate action.

It appears from the above considerations that it would require a truly extraordinary combination of human and instrument failure to produce an excursion of sufficient magnitude to cause melting of fuel plates and release of activity to the environment.

B. Chemical Reactions

Because thermodynamic calculations show that aluminum and aluminum alloys should react with water over a wide range of temperatures, it is necessary to examine the conditions under which reactions might take place. A survey of the available literature has been made to compare the conditions of reaction described in the literature with the situation existing in the swimming pool reactor under run-away conditions, i.e., where the temperature of the aluminum-uranium fuel elements would reach the melting point.

Inasmuch as almost all the reports on this subject contain restricted security information they will not be discussed here. However, it may be said that in the Borax experiments (Ref. 4), where conditions were similar to a swimming pool reactor, "The very brief study which has been made to date has not revealed any aspect of the results of the excursion which can definitely be shown incompatible with the hypothesis that the explosion was purely a 'steam' explosion." It is known that most of the fuel elements reached the the melting temperature (mp 660°C for Al) and it was surmised that a large fraction reached temperatures in the range 2000 to 3000°F. At any rate "...even if a significant fraction of the aluminum reacted, the reaction stopped before a still largerfraction was involved."

A metal-water reaction, if it took place to a large degree, would cause the dispersion of fission products, and containment within the reactor building would be very difficult. On the other hand, the heat of the reaction, causing the cloud to rise, would help reduce the radiation dose at ground level.

IV. POSSIBLE CONSEQUENCES OF A RELEASE OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

A. Introduction

This section considers the possible consequences of a nuclear incident, including the radiation hazard to the surrounding population. Any analysis, at best can be only an approximation of what might in fact occur, but at least a reasonable upper limit of hazard may be established. Inasmuch as the radioactive material escaping the reactor core, but confined to the reactor building, presents no hazard to off-site personnel, and only a slight hazard beyond the reactor area, only a release to the general environment will be considered.

The Quehanna site is isolated from large population centers. In the hazard analysis below the following specific locations will be considered:

Location 1: The shortest distance from the reactor to the site boundary. This is about 4.8 kilometers SW.

Location 2: The shortest distance to a population center on-site. This would be a distance of 8 kilometers ESE to the Main Area where the Research, Plastics and Administration Buildings are located.

Location 3: The shortest distance to an off-site population center. This is a distance of about 16 kilometers to each of four towns: Karthaus, SE; Driftwood, NNE; Sinnemahoning, NE; and Benezette, NW. Karthaus with 575 people is the largest.

Location 4: Along Mosquito Creek with Karthaus about 24 kilometers down the valley.

B. Radiation Hazard Due to Release of Radioactive Material to the Atmosphere

The magnitude of the hazard will depend on the quantity of fission products released from the reactor, the heat content of the cloud formed, the size and shape of the particulates, the micrometerorological conditions, the terrain and the location of the population whose exposure is being considered.

1. Activity Available

a. Mixed Fission Products. The rate of release of β and γ energy by fission products per megawatt of reactor power may be expressed by

$$E_\gamma = 2(10^{17})\left[t^{-0.2}-(t+t_o)^{-0.2}\right]$$
$$E_\beta = 2.2(10^{17})\left[t^{0.2}-(t+t_o)^{-0.2}\right]$$

where t is the decay time in seconds and t_o is the operating time of the reactor in seconds. If an average β energy of 0.4 Mev and an average γ energy of 1.0 Mev is assumed, and if the reactor is assumed to have operated for a long time, then the activity may be expressed as

$$Q_\gamma = 7.8(10^6)t^{-0.2} \quad \text{curies/Mw}$$
$$Q_\beta = 14.8(10^6)t^{-0.2} \quad \text{curies/Mw}$$

b. Specific Isotopes. The rate of formation of specific isotopes is given by

$$\text{Rate} = 8.42(10^5)\ C\lambda \ \text{curies/sec/Mw}$$

where C = fission yield of the isotope in atoms/fission
λ = decay constant of the isotope in sec^{-1}

The quantity present after a reactor operating time t_o and a decay time t is

$$Q = 8.42(10^5)C\left[1-e^{-\lambda t_o}\right]e^{-\lambda t} \quad \text{curies/Mw}$$

If continuous operation at 1 Mw is considered, the time required to reach the assumed maximum burnup of 10% is $3(10^7)$ seconds. The table below lists the most significant isotopes present at the end of this time.

TABLE IX

QUANTITY OF SPECIFIC ISOTOPES AVAILABLE

Isotopes	C (atoms/fission)	half life	Q (curies)
Sr-89	.048	53 d	4.0 (10^4)
Sr-90 + Y-90	.059	25 y	1.4 (10^3)
Y-91	.059	61 d	5.0 (10^4)
Ba-140 + La-140	.063	12.8d	5.3 (10^4)
Ce-144 + Pr-144	.061	282 d	3.1 (10^4)
I-131	.029	8.14d	2.5 (10^4)

c. Effect of Intermittent Operation. The activities listed above represent an upper limit because continuous operation for this length of time at full power would never be achieved. In actual practice, there would be many shutdowns and some operation at less than full power. A simple situation will be considered where the reactor is operated at full power for 8 hrs and then shut down for 16 hrs up to the point of 10% burnup. The activity of any isotope can be shown to be

$$Q = 8.42(10^5)C(e^{\lambda/3}-1)\left(\frac{1-e^{-\lambda(m+1)}}{1-e^{-\lambda}}-1\right) \quad \text{curies/Mw}$$

where λ is the decay constant in days^{-1} and m is the number of days of operation.

Under this sort of intermittent operation, the inventory of Sr-90 + Y-90 is reduced by 5%, Sr-89, Y-91, Ba-140 + La-140 by 66% and Ce-144 + Pr-144 by 50%, compared to continuous operation. The gross beta-gamma activity one hour after completing an 8 hr operating period would be about 40% of the activity one hour after shutdown following continuous operation.

2. Meteorological Parameters

The dispersion of the fission products released to the atmosphere may be estimated by means of the Sutton diffusion equations. The meteorological parameters required are wind speed u, the diffusion coefficients Cx, Cy, Cz and the stability parameter n. The validity of Sutton's treatment at large distances, for all values of n and for terrain that is other than level and uniform is uncertain. The uncertainty, however, probably is no greater than the uncertainty inherent in many of the estimates required for this analysis. Inasmuch as no previous data exists for the Quehanna site the meteorological parameters used by Smith (Ref. 12) in the recent study of reactor hazards will be used. The local topography will be crucial in determining the micrometeorology, and great variation may exist between different locations. Two conditons will be treated, viz., inversion conditions that may be typical of night-time, and average daytime lapse conditions.

	Inversion	Lapse
u (m/sec)	3	5
n	0.55	0.25
Cy, Cx	0.40	0.40
Cz	0.05	0.40

3. Inhalation Dose

An observer in the path of the cloud will take radioactive material into his body by inhalation. Generally, the material will concentrate,to a large extent, in a particular portion of the body, the critical organ. The inhalation dose from a particular isotope to the critical organ for a person standing at the axis of the cloud through its passage can be expressed as

$$D = \frac{1.16(10^4)Q}{C_y C_z u x^{2-n}} \frac{Ef}{M\lambda} (1-e^{-\lambda t})$$

Where D = dose in rep to the critical organ
M = mass gm of the critical organ
f = fraction of inhaled isotope reaching the critical organ
λ = the effective decay constant of the isotope in the critical organ, in day^{-1}
Q = source strength, in curies
E = average energy of the isotope per disintegration, in Mev
t = time after inhalation over which the dose in calculated, days
x = the distance in meters from the reactor to the point of interest
C_y and C_z and u have the same meaning as before.

The dose to that organ receiving the greatest damage is taken as the inhalation dose in later calculations. For the operating time considered, the bone dose is the significant inhalation dose. While the thyroid receives a larger effective dose it is very radioresistant and therefore is not considered the critical organ. Figures 29 and 30 give the maximum inhalation dose for the important contributors for the two weather conditions. The dose given is the accumulated dose for an infinite time after intake and, therefore, is not directly comparable to a dose of the same magnitude given in a short period of time. Fig. 31 illustrates this point for the case of lethal doses in experimental animals. It is seen that the lower the dose rate, the less damage for a given total dose; e.g. if the irradiation of an animal is spread over 16 days instead of one day, twice the dose is required to achieve the same effect. Figure 32 shows the accumulated bone dose as a function of time after the incident.

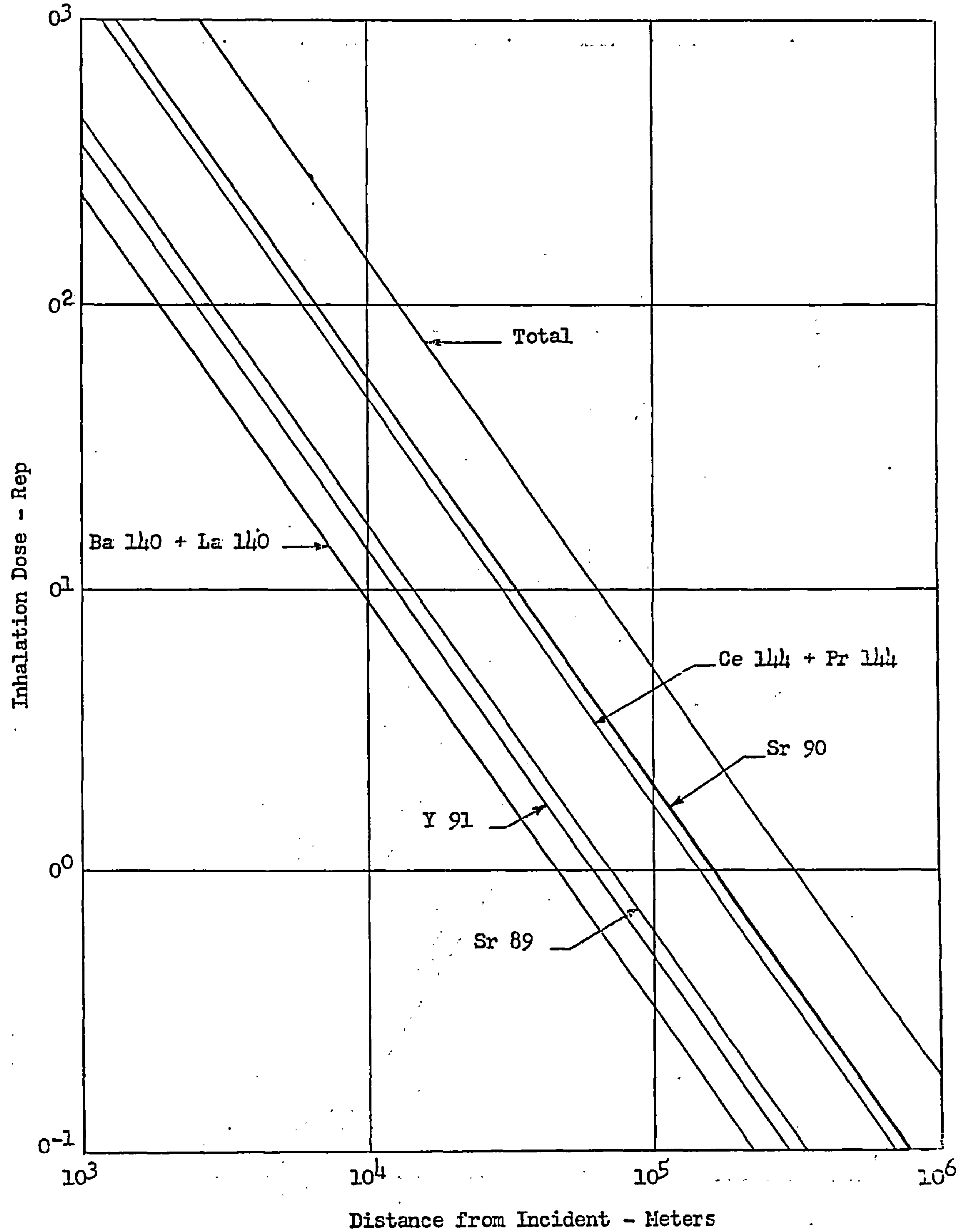
4. External Dose from the Passing Cloud

a. External β-Dose. The external β-dose to a person standing at the cloud axis throughout its passage is given by Ref. 13 as

$$D_\beta = \frac{Q_\beta \exp\left[-\frac{h^2}{C_z^2 x^{2-n}}\right]}{\pi u C_y C_z x^{2-n} \, 1.06(10^{11})} \quad \text{rep}$$

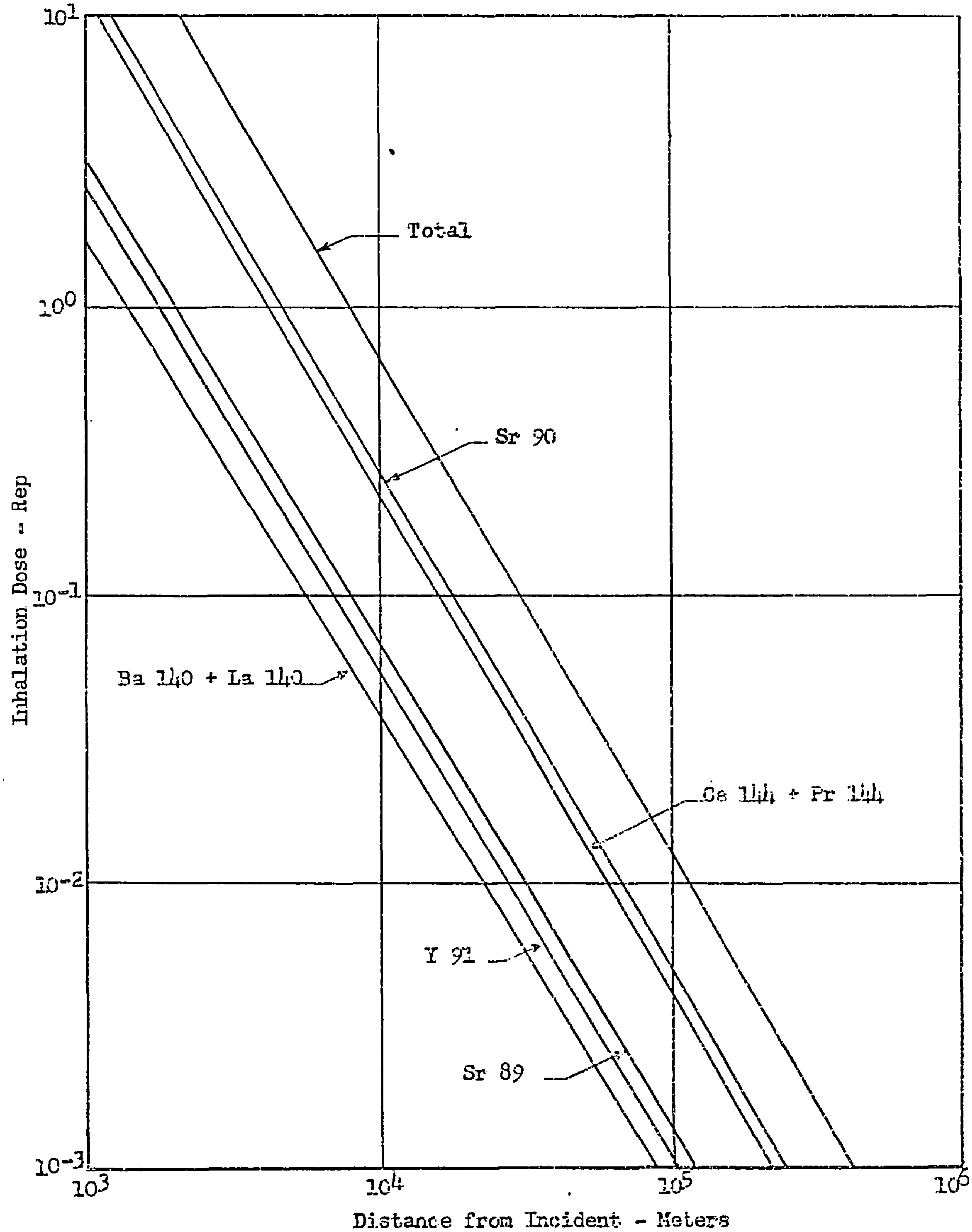
where Q_β is the total β-source strength in Mev/sec, and h is the height in meters of the cloud. Figure 33 shows the dose-distance relationship for an instantaneous ground source. This β-dose ignores all shielding afforded by clothing or shelters.

b. External γ-Dose. Figure 34 gives the external γ-dose as determined by Holland's method (Ref. 13).



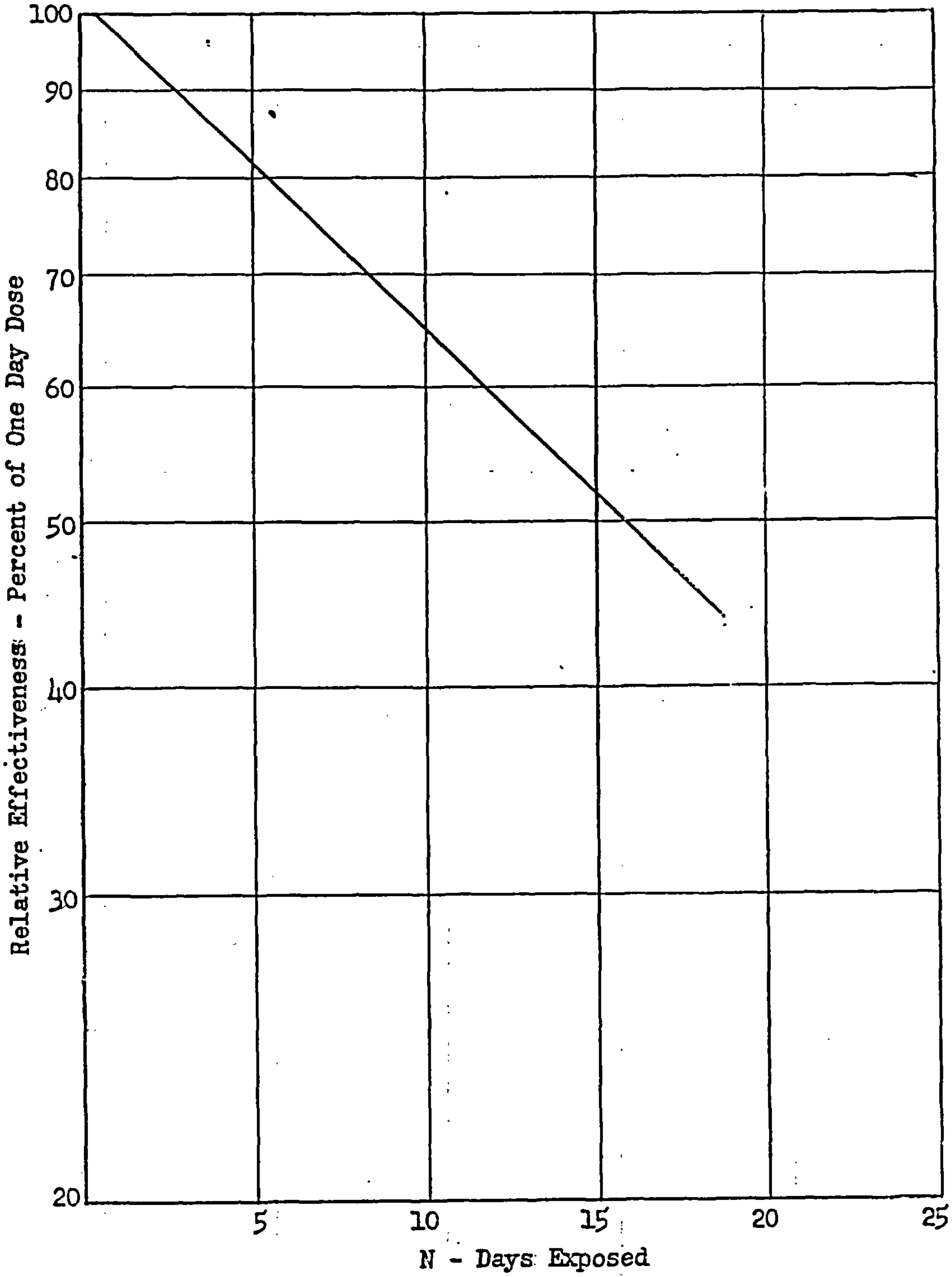
TOTAL INTEGRATED INHALATION DOSE TO BONE. INVERSION CONDITIONS
100% GROUND LEVEL RELEASE AFTER OPERATION
AT 1 MW FOR 1 YEAR

Figure 29



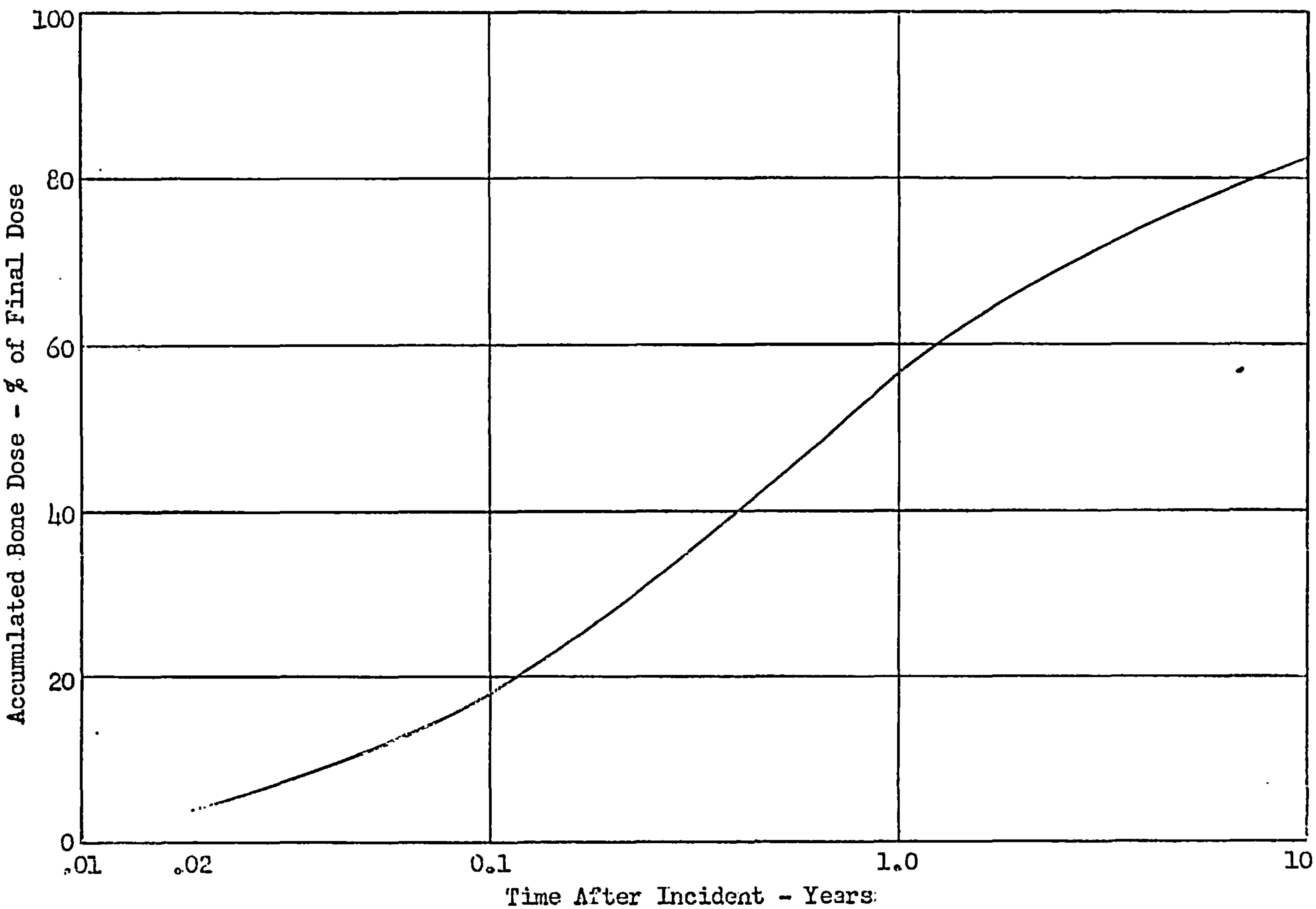
TOTAL INTEGRATED INHALATION DOSE LAPSE CONDITIONS,
100% GROUND LEVEL RELEASE AFTER REACTOR
OPERATION AT 1 MW FOR 1 YEAR

Figure [illegible]



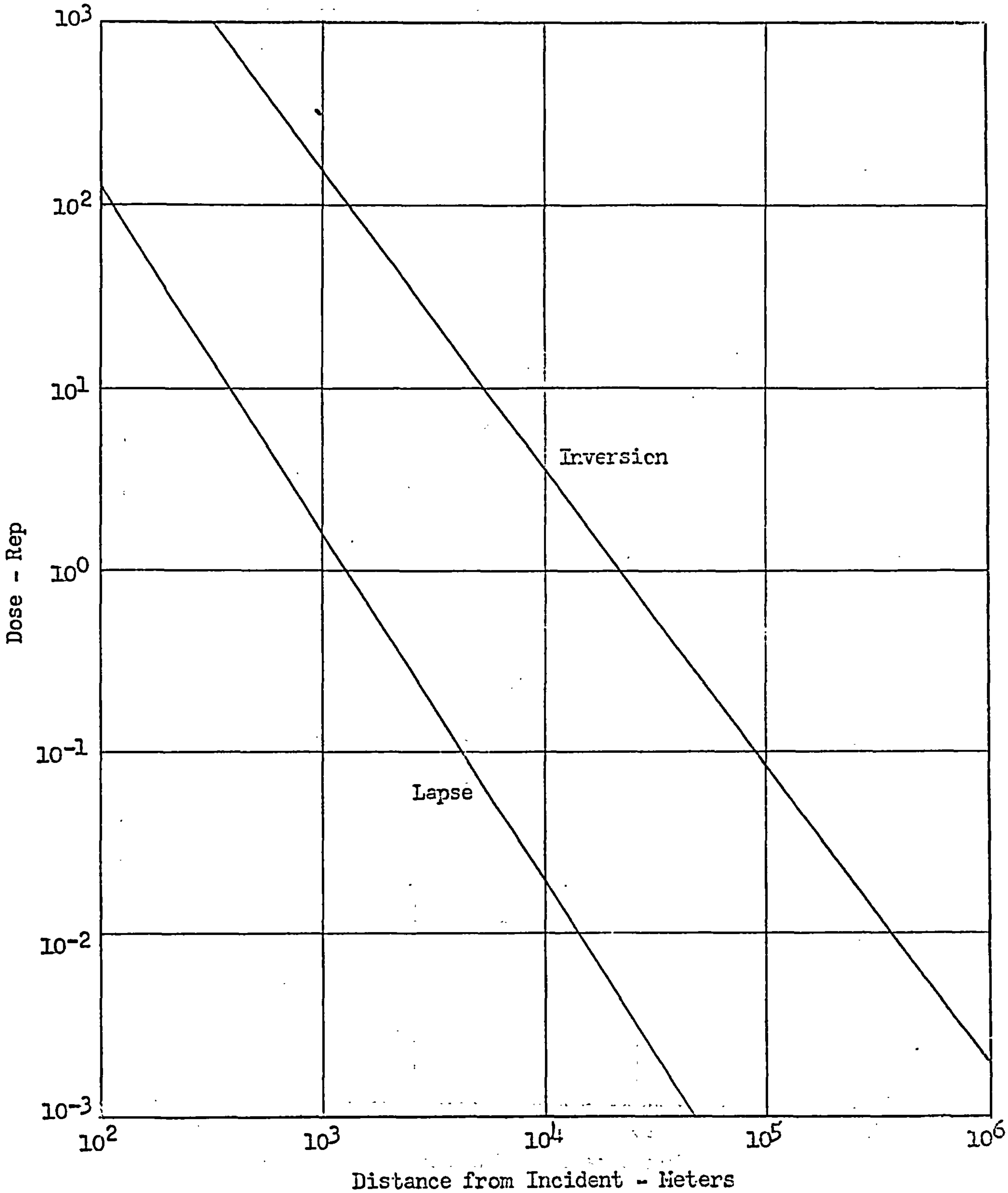
RELATIVE EFFECTIVENESS OF A DOSE FRACTIONATED OVER N DAYS
COMPARED TO THE SAME DOSE GIVEN IN ONE DAY

Figure 31



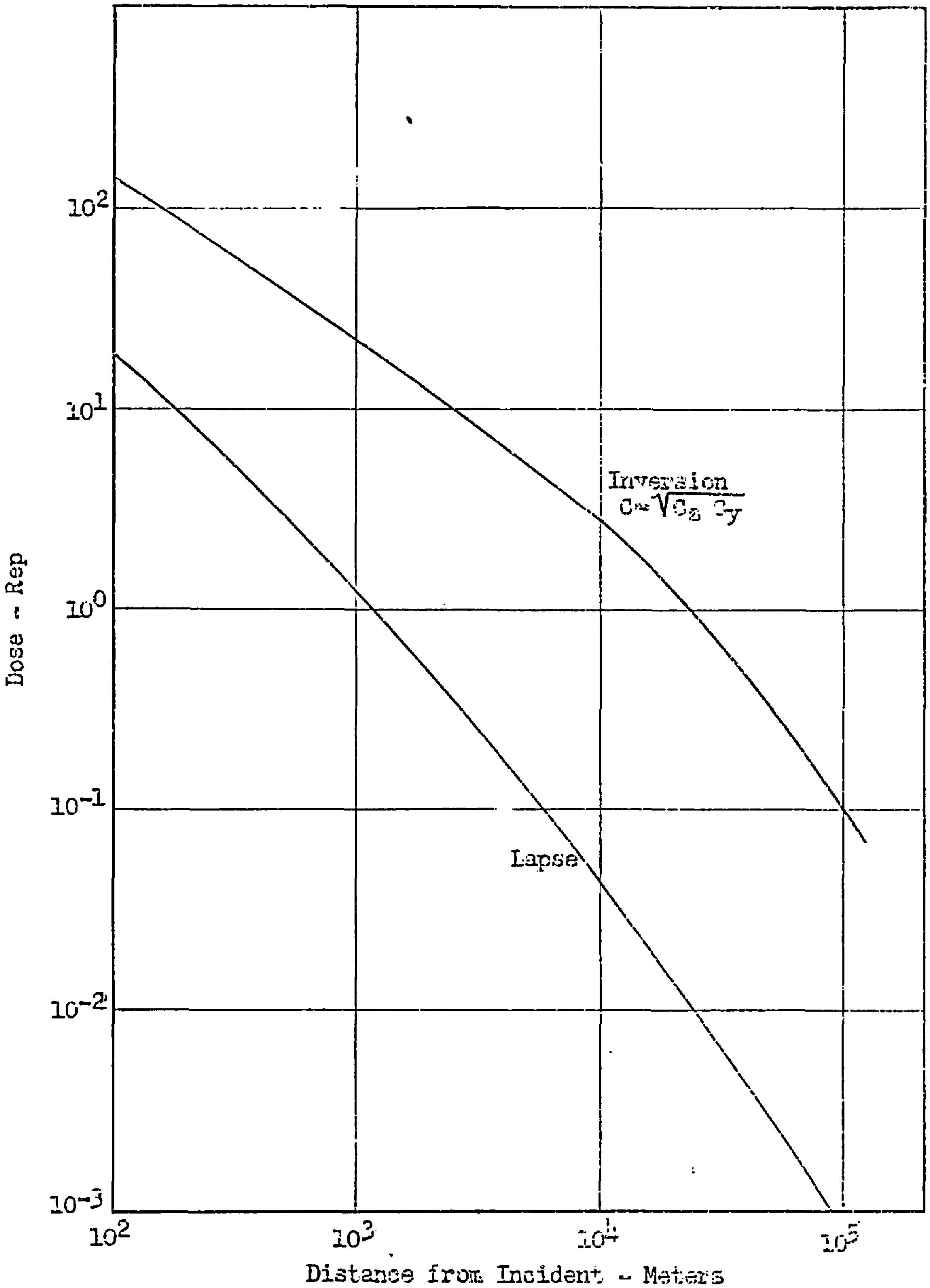
ACCUMULATED BONE DOSE AS A FUNCTION OF TIME

Figure 32



EXTERNAL BETA DOSE FROM CLOUD

Figure 33



EXTERNAL GAMMA DOSE FROM CLOUD

Figure 34

5. γ-Dose from Ground Deposition

The γ-dose from activity deposited on the ground (see Ref. 13) may be expressed as

$$D = 3.8\ (10^{-3}) W_o \left[t_2^{0.79} - (x/u)^{0.79} \right]$$

where W_o = surface deposition in c/m^2
x/u = the time to reach a point x from origin, u as before being wind velocity in m/sec
t_2 = the time over which the dose is integrated

a. Dry Fall-Out. A knowledge of the particle size distribution, density and shape is required to specify fall-out deposition. This is unknown, but a maximum deposition W_o can be calculated for any particular location. This deposition is given as

$$W_o = \frac{nQ}{2e\pi^{\frac{1}{2}} C_y x^{(2-n/2)}} \qquad c/m^2$$

b. Rain-Out. The rain-out deposition, and therefore dose, is 2/n times as great as for dry fall-out. This factor is equal to 3.65 for inversions and 8 for lapse.

6. Modifying Factors for a Credible Accident

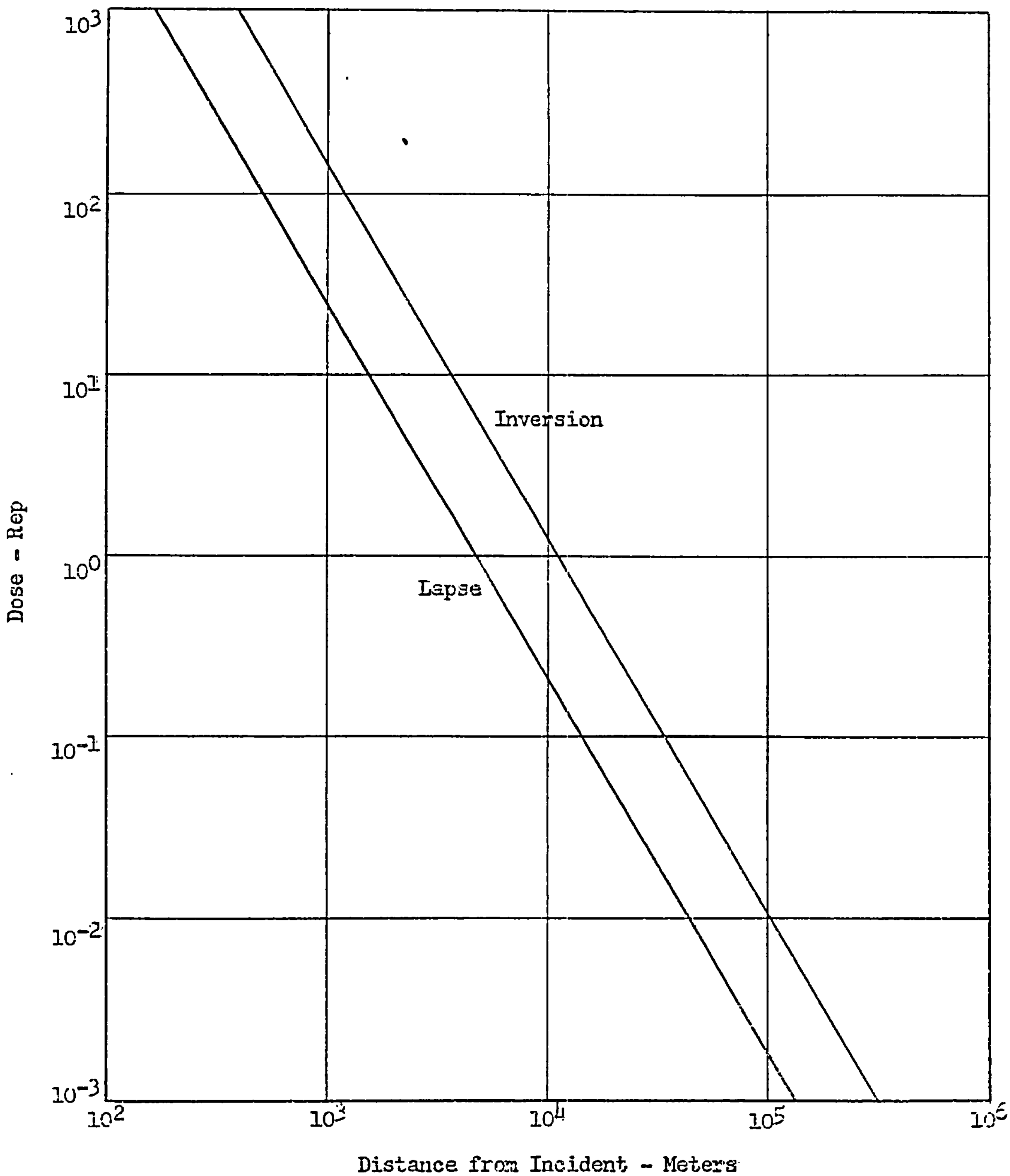
a. Activity Released. It is unrealistic to assume that 100% of the fission product inventory is released from the reactor and is of such size that it can remain airborne for an appreciable time. The Borax experiment (Ref. 4) gives "no indication that any large fraction of the fission products left the vicinity of the reactor." Workers at Oak Ridge (Ref. 11) report that only 1% or less of the fission rare gases and 0.01% to 0.1% of the iodine present in aluminum clad elements are liberated after heating the elements to well above the melting point for an hour. A reasonable upper limit to the quantity of fission products released to the atmosphere following an incident would be 10%.

b. Rise of the Cloud. If an appreciable amount of material is released it would very likely be at an elevated temperature and hence would rise. This rise would have a marked effect on the dose, particularly at short distances. The reduction factor is

$$\exp\left[- \frac{h^2}{C^2 x^{2-n}} \right]$$

where h is the cloud rise in meters.

According to Sutton (Ref. 13) the heat required to vaporize all of the aluminum of the fuel elements would give a total rise of about 400 meters. Actually, it would be expected that the cloud height would vary with meteorological conditions. However, if 400 meters is used, the following reduction factors are obtained.



EXTERNAL GAMMA DOSE FROM SURFACE DEPOSITION
IN THE FIRST 12 HRS FOLLOWING INCIDENT

Figure 35

x	Inversion	Lapse
4.8 km	0	0.57
8.0	0	0.80
16.0	0	0.93
24.0	0	0.97

It is of interest that following a major release, there is virtually no dose at the locations considered under inversion conditions.

c. Continuous Source. If the reactor contents are not released in a short period of time and can be considered as a continuous source, a reduction in dose may be expected for two reasons:

1) Reduction due to radioactive decay before release.

2) In Sutton's continuous source treatment, the wind direction is considered as unvarying. Excluding an inversion situation with the wind blowing down valley, this is most unrealistic if the release extends beyond a few hours. If the cloud sweeps uniformly through an angle θ during the release, the area swept will be uniformly irradiated (neglecting the edges). If the angle θ is equal to 30° for inversion conditions and 45° for lapse, the following reduction factors are obtained.

x	Inversion	Lapse
4.8 km	0.44	0.98
8.0	0.38	0.92
16.0	0.32	0.85

d. Reduction Due to Fall-Out. As the cloud moves out from the origin it will, in general, be depleted due to fall-out, rain-out or impaction. The deplection rate will depend on the settling velocity of the particles which is a function of particle size, shape and density. Since little is known about the particle size distribution to expect, depletion of the cloud will be neglected in this study. As far as the inhalation dose is concerned this is very pessimistic, particularly as the distance increases. The Borax experience indicates extensive fall-out in the immediate reactor vicinity. There may be some compensation for the decrease in the direct inhalation dose by an increased inhalation of particles stirred up after deposition.

7. Area Affected

The estimation of hazard requires some knowledge of the area covered by the cloud as well as the dose rate. The cloud width may be defined as the crosswind distance to points where the concentration is p percent of the axial concentration. From Sutton's equations

$$y = Cx^{(2-n)/2}\left[\ln \frac{100}{p}\right]^{\frac{1}{2}}$$

where y, the half-width, is the distance in meters from the axis to the point where the concentration is p percent. Figure 36 gives the half width where p is 10%.

8. Maximum Credible Accident

The maximum credible accident is considered to be one in which owing to instrument or human failures, a power excursion occurs of sufficient magnitude to cause the melting of the fuel elements. The ensuing disruption of the core than causes a shutdown. However, the temperature and geometrical configuration of the molten aluminum is considered to permit a metal-water reaction as a result of which a total of 10% of all fission products are liberated as particles sufficiently small so that they remain airborne and leave the reactor vicinity.

It should be emphasized that while the accident described above is credible, the probability of such an occurrence is extremely small. As indicated in Section III, an initial excursion of sufficient magnitude to raise the fuel plates above their melting point is, itself, a very difficult thing to produce. Such an excursion alone would not be expected to release more than 1% of the contained fission products. To release more than this, the reactor excursion must be coupled with a second very unlikely event, a metal-water reaction which would liberate considerably more energy than the flux excursion.

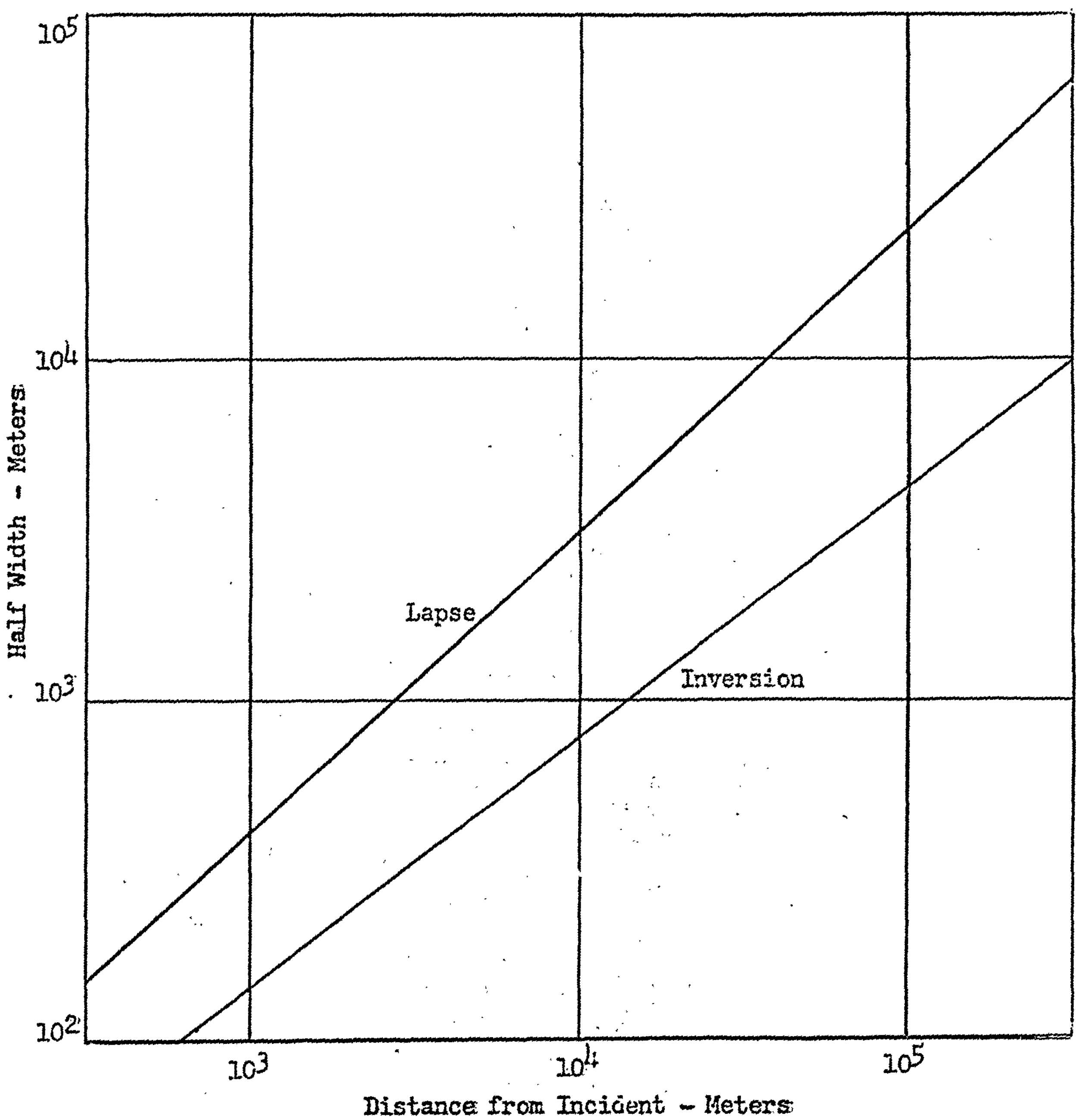
The effect of this maximum credible accident at the four critical locations mentioned previously are considered below:

Location 1. The shortest distance to an off-site point. This location is part of State Forest lands and contains scattered hunting cabins which are very seldom occupied. Because of the topography, it is unlikely that during an inversion the cloud could carry in this direction, but rather the cloud would follow the valley of Meeker Run and Mosquito Creek. Neglecting depletion of the cloud due to fall-out for the inhalation dose and external dose, and considering an instantaneous ground level release, the doses to a person remaining on the cloud axis are as follows:

Inhalation bone dose	0.23 rep
External beta dose	0.01 rep
External gamma dose	0.01 rep
12 hour fall-out dose	0.10 rep
12 hour rain-out dose	0.80 rep

This obviously represents no serious hazard.

Location 2. Main area on site. This area generally is populated only during daylight working hours. Both lapse and inversion conditions will be considered although here, too, it is not at all probable that the cloud would travel in this direction under inversion conditions. With the same assumptions as before, the doses on the cloud axis are:



CLOUD HALF-WIDTH

Figure 36

	Inversion (rep)	Lapse(rep)
Inhalation	19.0	.09
External beta	0.5	.003
External Gamma	0.3	.006
12 hour fall-out	0.2	.03

Even this maximum case, where no modifying factors were used, does not represent a serious hazard. If depletion of the cloud, cloud rise, a lower fission product inventory, shelter afforded by buildings or off-axis positions were considered, the doses would be far less. It should be emphasized that the inhalation dose, which is the main contributor, is spread out over a long time. K.Z. Morgan (Ref. 15) has suggested as a maximum permissable intake for a single exposure of radioisotopes that quantity which would deliver to the critical organ 0.3 rem in a seven day period. The calculated inhalation dose in rep is increased by a factor of 5 because of an assumed non-uniform distribution of the isotopes in the bone. The dose in the first week for the inversion case above is only 0.75 rep and declines steadily for each succeeding week. This is 12.5 times the suggested limit. An initial cloud rise of less than 60 meters would reduce the dose to the limit.

Even if the total dose were delivered in a short time there would be no injury expected at the 20 rep level, which is below the 25 rep sometimes suggested as a maximum acceptable emergency dose.

The doses at locations 3 and 4 likewise represent no significant hazard, being approximately 7 and 4 rep, respectively, for the inversion case with the inhalation dose again the major contributor. Doses at these locations under lapse conditons are entirely negligible.

It will be noted that in all the above examples, no reduction in dose was considered as a result of intermittent operation, cloud rise, a non-instaneous release or fall-out. Because of the distances involved from reactor to populated areas, there is ample time available for warning and evacuation . This is particularly true for the more serious inversion case which would generally involve low wind speeds. In addition, because of the time interval involved during inversions, the possibility exists of a meteorological change favoring greater dispersion.

C. Radiation Hazard Due to Release of Radioactive Materials to Streams

Radioactive materials may enter the surface water system and be carried to population centers in two ways. Activity may be released to the atmosphere and be washed out by rain and, if the ground is saturated, run off into the streams. A much less probable eventwhich is not considered credible would be the release of fission products to the pool water and simultaneous rupture of the pool wall.

The run-off eventually would find its way into Mosquito Creek and thence to the West Branch of the Susquehanna River. The Mosquito Creek is not used for drinking water and the Susquehanna is, in fact, polluted from other sources. Any accidental consumption of active water would not be wide-spread.

As the maximum credible accident involving release of activity to water, a situation will be considered in which, following the release of 10% of the fission products into the atmosphere and complete washout by a 0.1 in. rain, a person at Karthaus accidentally ingests 250 ml of contaminated water. The drainage area of Mosquito is 70 sq. mi. Ingestion will be assumed to take place about 24 hrs. after the incident.

The exposure of the gastrointestinal tract, if all of the activity remains there, will be of the order of 0.06 rep/hr. Elimination and distribution throughout the body would soon reduce the dose rate.

The dose due to specific isotopes going to critical organs is

$$D = 5.5(10^7)AV \frac{fE}{M\lambda} (1-e^{-\lambda t})$$

where A = activity in water in c/ml
V = volume of water consumed in ml
f = fraction of intake reaching the critical organs
D = integrated dose to time t
λ = effective decay constant in day^{-1}
t = time over which dose is integrated in days

The total dose is less than 0.5 rep and again is spread out over a long period. The immersion dose is negligible. It is apparent that the situation outlined would not present a serious health hazard.

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APPENDIX I — GEOLOGY AND HYDROLOGY OF THE QUEHANNA SITE

This material is abstracted from a report by George D. DeBuchananne entitled "Reconnaissance of the Geology and Hydrology of a Proposed Site for the Curtiss-Wright Corporation Nuclear Development and Propulsion Facilities near Karthaus, Pennsylvania". During the reconnaissance the geologic and hydrologic features of the area were not investigated in detail. Previous investigations of the geology and hydrology include reports by S. H. Cathcart, S. W. Lohman and J. W. Mangan (References 1 to 6). At the end of this section is a list of material including published and unpublished references, referring directly or indirectly to the area, which were used to gain background information.

A. Geology

The Quehanna site, which is in the Appalachian Plateau physiographic province, is underlain by strata that have been disturbed but slightly from their original attitude, and lie nearly horizontal in most places. An anticline and a syncline do, however, cross the northern half of the area and undoubtedly have a marked effect on the occurrence of ground water in their immediate vicinity.

The underlying rocks at the Quehanna site range from Devonian to Pennsylvanian in age. The generalized geologic section, Table A-1, represents a composite section based on the literature. Exposures of the geologic formations in the area are so limited that it was impractical to attempt to measure a geologic section.

Cathcart (References 1, 2, 3) reports that an anticline and a syncline occur in the northern half of the area. In a reconnaissance trip down Red Run, along Bennett Branch, and Sinnemahoning Creek and then up Wykoff Run, scattered out-crops gave field evidence of these structures. The so-called Sinnemahoning syncline lies between the Wellsboro and the Chestnut Ridge anticlines.

B. Hydrology

The hydrological cycle at the Quehanna site is similar to that in other humid areas. Moisture in the form of precipitation falls to the earth's surface; some is lost to the atmosphere by evaporation; some runs across the land surface to surface drainage and hence to the oceans where it is again returned to the atmosphere by evaporation; and some is absorbed by the soil to be used by plant life or to be added to the zone of saturation where it becomes ground water. This investigation is concerned with the hydrological cycle of the area only insofar as the ground water and surface water are concerned.

TABLE A-1

Generalized Section of Geology at Quehanna, Karthaus, Pennsylvania

Age	Formation	Thickness (feet)	Description	Water-Bearing Properties
Pennsylvanian	Pottsville	200±	Consists of massive coarse-grained gray to white sandstones with pebbles as large as hazelnuts. Caps hilltops. Probably represents the Olean member of formation.	Sandstones productive where found below drainage level, generally yield small to moderate supplies elsewhere.
Mississippian	Mauch Chunk shale	50±	Red and green argillaceous shale, with some sandstone. Not generally exposed, indicated by a terrace developed between Pottsville conglomeratic cliffs above and steep Knapp slopes below.	Not a water-bearing horizon, probably forms impervious strata retarding downward percolation of water.
Mississippian or Devonian	Knapp (Pocono)	600±	Succession of alternating olive-gray, gritty, micaceous sandstones and gray-green argillaceous shales. Some red beds occur near bottom of formation.	Productive consolidated rock where encountered below drainage level.

1. Ground Water. The source of ground water in the Quehanna area is precipitation. A part of this precipitation seeps down through the soil to the zone of saturation, the top of which is called the water table. Water in the zone of saturation moves laterally through permeable zones towards points of discharge. In the immediate area of investigation, recharge occurs primarily on the high interstream areas and discharge occurs as springs on the slopes of the deep stream gorges or as loss to the streams themselves where they intercept the water table.

The Pottsville formation capping the hilltops of this area absorbs and transmits downward a part of the precipitation that falls on it. Exception to this condition exists in a few areas where lenses of shale in the Pottsville formation form impermeable zones at the surface causing swampy and wet areas. In general, however, water moves downward fairly rapidly through the Pottsville area along joints and fractures and through the interstices of the rock itself. The Mauch Chunk shale, however which underlies the Pottsville formation is relatively impervious and, therefore, hinders the downward movement of water. As the water can no longer move downward, it moves laterally away from points of recharge to points of natural discharge along the slopes of the stream gorges. The top of the Mauch Chunk shale which is believed to occur at an elevation of slightly less than 1900 ft above mean sea level probably forms the bottom of the zone of saturation in the formations overlying the Mauch Chunk in this area. In areas where the strata have not been deformed and fractured by folding very little additional fresh water will be obtained below the top of the Mauch Chunk shale.

Where the Mauch Chunk shale has been fractured such as in the area of the Chestnut Ridge anticline and Sinnemahoning syncline in the northern part of the Quehanna area the shale does not serve as an impervious zone, but transmits water downward to recharge the underlying Knapp (Pocono) formation. In such areas, if sufficient water has not been obtained in the Pottsville formation, additional water may be available from the underlying Knapp (Pocono) formation. The syncline in the northern part of the area is the most promising location for groundwater supply. Where the Knapp (Pocono) formation is exposed at the surface, such as along the streams in the deeper gorges, the formation is recharged directly by precipitation and by water loss from the surface stream. In this latter case, the Knapp (Pocono) formation probably will yield moderate amounts of water to wells.

At least four wells have been drilled at the Quehanna site for water supplies. Two of the wells, No. 1 and No. 2, drilled near the main entrance by the Pennsylvania Drilling Company of Pittsburgh, Pennsylvania, were abandoned because of low production. The other two wells, No. 3 and No. 4, drilled by Kohl Brothers Drilling Co. of Harrisburg, Pennsylvania at about the center of the site were successful wells and are being used at these locations. The drillers logs of the No. 1 and No. 2 wells are given in Table A-2.

TABLE A-2

Drillers Logs of Test Well 1 and 2

Test Well No. 1 Thickness (in feet)	Depth (in feet)	Strata	Tentative Geologic Correlation	Test Well No. 2 Thickness (in feet)	Depth (in feet)	Strata
15	15	clay		15	15	clay
65	80	sand hard		15	30	sand
12	92	shale	Probably lower	25	55	shale
76	168	sand	Pottsville	25	80	sand
7	175	shale	formation	20	100	slate ar shale
10	185	red rock		40	140	sand
8	193	shale	Probably Mauch	40	180	slate
27	220	shale and sand	Chunk shale			
10	230	pink rock		55	235	red rocl
25	255	shale		12	247	sand
95	350	sand				
10	360	sandy shells				
75	435	sand				
45	480	slate and shells				
10	490	sand				
20	510	red rock	Probably Knapp			
15	525	sand	formation			
23	548	slate and shells				
42	590	sand				
5	595	slate and shells				
22	617	sand-hard				
18	635	red rock				
10	645	slate and shells				
15	660	red rock				
40	700	slate and shells				

The following are notes from the drillers (Mong and Hickey) logs on these two wells.

Well No. 1

75 ft Some water
168 ft 19 bailers, 24 minutes
370 ft Some water
265 ft Water dropped to 90 ft from top.
365 ft Water dropped to 65 ft from bottom.
Checked with bailer made 37 gpm after being idle for one week, water level came back to 130 ft from top.

Well No. 2

60 ft-32 gallon, 17 minutes
140 ft-55 gallon, 12 minutes
Static level, completed.
Hole 131 ft

These logs indicate that the Mauch Chunk shale is acting as an impervious stratum that prevents the Knapp (Pocono) formation from being recharged. If the tentative geologic correlations are correct, there appears to be a saturated section of about 40 ft thickness overlying the shale. The Pottsville, however, either has a relative low permeability at the location or the wells were not properly developed since both wells were abandoned because of low yield.

Unfortunately no drillers logs were kept on wells No. 3 and No. 4, so little is recorded about subsurface conditions. The driller, Mr. Yohn, and the Drilling Company office in Harrisburg were contacted and the following information obtained:

	Well No. 3	Well No. 4
Diameter, in.	6	6
Depth, ft	250	241
Length of casing, ft	15	15
Water level (static), ft	75	72
Pump setting, ft	200	180
Pump, hp	3 Fairbanks Morse	3 Fairbanks Morse
Capacity, gpm	52	140
Pumping water level, ft	204	-----

A 24 hour pumping test conducted on August 15 and 16, 1955 on well No. 3 indicates a specific capacity of only .4 which represents the yield in gallons per minute per foot of drawdown. No test was recorded for well No. 4 but apparently the specific capacity is about three times as large as that for well No. 3.

The available information from these four wells indicates that there is a fairly good correlation between the subsurface data despite the fact that they are several miles apart.

2. Surface Water. As indicated earlier in this report, the surface streams at the Quehanna site have their origin on the interstream upland, but quickly enter deep gorges through the flat lying rocks to reach the master streams of the area. The master streams of the site are Sinnemahoning Creek and Mosquito Creek, both tributaries of the West Branch of the Susquehanna River, and both have cut their valleys some 900 ft below the interstream uplands of the Quehanna site.

Four stream-gaging stations are operated in the Quehanna area by the U. S. Geological Survey in cooperation with the State Pennsylvania Department of Forest and Waters. The location of these stations and data on the drainage areas and discharge, as reported by Mangan (Reference 6), are as follows;

1. West Branch Susquehanna River at Karthaus, Pa.
2. Driftwood Branch Sinnemahoning Creek at Sterling Run, Pa.
3. Sinnemahoning Creek at Sinnemahoning, Pa.
4. First Fork Sinnemahoning Creek near Sinnemahoning, Pa.

Station	1	2	3	4
Drainage, sq mi	1,462	272	685	245
Record available	1940-date*	1918 to date*	1938 to date*	1953 to date*
Average discharge, cfs	2463	485	1,150	328
Maximum discharge, cfs	50,900	47,800	50,000	5,670
Peak discharge, cfs 3-18-42 flood	135,000	-------	61,200	80,000
Minimum discharge, cfs	109	0.4	1.2	6.4

*Record as of June 8, 1955. Records since this date are available from U. S. Geological Survey, P. O. Box 421, Water Resources Division, Surface Water Branch, Harrisburg, Pennsylvania.

Below the junction of Mosquito Creek and the West Branch of the Susquehanna River the river is not used for public water supplies because of the acidity of the water. At Karthaus, Pennsylvania on August 8, 1944, it is reported (Reference 7) that the West Branch of the Susquehanna had an average flow of 850 cfs of water with a pH of 3.2. On April 17, 1945, at the same station, the pH was 3.60 when the flow was 2,490 cfs.

C. Earthquake Activity

Earthquake intensities given on the Rossi-Forel scale of intensities as follows:

1. Micro seismic shock-recorded by a single seismograph or by seismograph of the same model, but not by several seismographs of different kinds.

2. Extremely feeble-shock recorded by several seismographs of different kinds; felt by a small number of persons at rest.

3. Very feeble shock-felt by several persons at rest: strong enough for the direction or duration to be appreciable.

4. Feeble shock-felt by persons in motion: disturbance of movable objects, doors, windows, cracking of ceilings.

5. Shock of moderate intensity-felt generally by everyone; disturbance of furniture, beds, etc; ringing of some bells.

6. Fairly strong shock-general awakening of those asleep; general ringing of bells; oscillation of chandeliers; stopping of clocks; visible agitation of trees and shrubs; some startled persons leaving their dwellings.

7. Strong shock-overthrow of moveable objects; fall of plaster; ringing of church bells; general panic, without damage to buildings.

8. Very strong shock-fall of chimneys, cracks in the walls of buildings.

9. Extremely strong shock-partial or total destruction of some buildings.

10. Shocks of extreme intensity-great disaster; ruins, disturbance of the strata, fissure in the ground; rock falls from mountains.

Six earthquakes have been reported to have had their epicenter within the geographical boundaries of the State of Pennsylvania. Some of the stronger Canadian earthquakes and the New York shock of 1929 were also widely felt throughout the state. The shocks listed below for Pennsylvania according to Heck (Reference 8) were all local in nature and had low intensities. The shocks in Table A-3 are listed by date, each gives the location of the epicenter where known, and the intensity of shock at the epicenter.

TABLE A-3

List of Earthquakes with Epicenters in Pennsylvania

Year	Date	Locality	N. Lat.	W. Long.	Intensity
1800	March 17	Philadelphia	39.8	75.2	----------
	Nov. 29	Philadelphia	39.8	75.2	----------
1840	Nov. 11	Philadelphia	39.8	75.2	----------
1877	Sept. 10	Delaware River	40.3	74.9	4-5
1884	May 31	Allentown, Pa.	40.6	75.5	6-7
1889	March 8	Pennsylvania	40	76	6
1908	May 31	Allentown, Pa.	40.6	75.5	6

From the record it is apparent that the Quehanna site is not subject to frequent earthquake activity. In recorded history there has been no earthquake centered within 150 mi of the site. Because of the absence of earthquakes only reasonable care is necessary with regard to foundations and construction of buildings at the site.

D. Quehanna Operations

Curtiss-Wright Corporation has plans for diversified industrial operations at the Quehanna site. Only those operations which involve nuclear facilities or the use of nuclear material are of concern to this geologic and hydrologic investigation.

Assuming normal safety precautions are taken, the problems then become those which are accidental in nature. As the nature of an accident and type of contaminant cannot be forseen, the problem can best be evaluated when no relative values are assigned to the contaminant. Simply stated then, the problem is, what would happen to a radioactive liquid or solid material which might be set free to be dissipated by nature.

If the material is solid, it would remain in place as a point source of contamination and, as exposed to the elements, would gradually be neutralized. This decay process however, would provide a continuous source of contamination. If the material is liquid the accident would release a single slug of contamination.

In the event of an accidental spillage of a large quantity of radioactive liquid, a portion of the fluid would run off overland, ultimately reaching the tributaries of the West Branch of the Susquehanna River. That part of the fluid that enters the soil would move downward under the influence of gravity through the unsaturated zone to the water table. Upon reaching the water table it would, depending upon its specific gravity, still continue its downward movement, but would also move horizontally towards a point of discharge. The amount of percentage of a given volume of fluid that would enter the soil would depend upon the soil condition at that time. If, for example, an accident occurred during or immediately following a heavy rain storm or during a period when the ground was frozen all or most of the liquid would run off overland and little, if any, would enter the soil. On the other hand, an accident occurring under the right climatic conditions would result in all or most of the fluid entering the soil.

In the case of overland movement of radioactive fluids, the time factor is short in moving a contamination from one area to another, whereas in the case of underground movement the time factor would be many many times larger.

At the reactor and radioactivity laboratory area, any fluid that enters the soil would percolate downward to the water table then move laterally to points of discharge. Since the strata at these areas are essentially flat lying and probably not fractured it is believed that no contamination would penetrate the Mauch Chunk shale which forms the lower confining member of the aquifer and would probably determine the elevation of points of discharge from the aquifer.

The reconnaissance of the Quehanna site indicates, without detailed information on subsurface conditions, that this is a reasonably safe area for operations which do not deal with products of extremely high radioactivity. As the contemplated operations do not involve the storage or disposal of radioactive wastes it is reasonable to assume that any contamination that would affect the ground-water would be the result of an accident or of some undetected leak in a fluid system containing radioactive material.

APPENDIX I

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APPENDIX II -- Meteorological Appraisal of the Quehanna, Pa. Site of the Curtiss-Wright Corporation

Prepared by Office of Meteorological Research
U. S. Weather Bureau
May 3, 1956
by
D. H. Pack

A. Introduction

The purpose of this report is to review the meteorology of the Quehanna, Pa. area for use in site evaluation and engineering application. The area under consideration is an approximate circle of 5 mi radius located 30 to 40 mi northwest of State College, Pa. and encompassing portions of Clearfield, Elk, and Cameron Counties. It has been proposed that this site be utilized for the development of a number of facilities including certain nuclear developments. In order that this report may be as useful as possible, consideration was given to the meteorological parameters of interest not only to the nuclear facilities but also to other contemplated installations.

B. Local Topography

The dominant topographical features of this section of Pennsylvania are the series of parallel ridges oriented northeast-southwest and rising 500 to 1,500 ft above the intervening valleys. In the immediate area of the site the ridge orientation is less pronounced and the terrain is very irregular. The site proper is on a rolling plateau with elevations generally between 1,900 and 2,000 ft mean sea level. The plateau is penetrated by a number of deep, relatively narrow, ravines or valleys radiating outwards from near the center of the site in almost all directions except to the west. The range in elevation of the site is from about 2,300 ft msl on several small knolls at the eastern edge, to 1,000 to 1,200 ft msl at the bottom of some of the deeper ravines where they cross the site boundaries. The terrain surrounding the site to distances of 20 to 25 mi has about the same character and range in elevations and there are no marked sheltering effects from any near by higher ridges.

C. Source of Data

Although no meteorological data exists for the proposed site itself complete meteorological records have been taken for a number of years at the Philipsburg, Pa. Airport (Black Moshannon) which is 27 mi southeast of the Quehanna site. The topography at and surrounding the

Philipsburg Airport is quite similar to the Quehanna site. The Philipsburg elevation is about 1,963 ft msl and is located on the top of a plateau very much resembling the Quehanna site. For most purposes, the meteorological data which have been collected at Philipsburg will be adequate for the preliminary evaluation of the Quehanna site.

D. Climatological Review

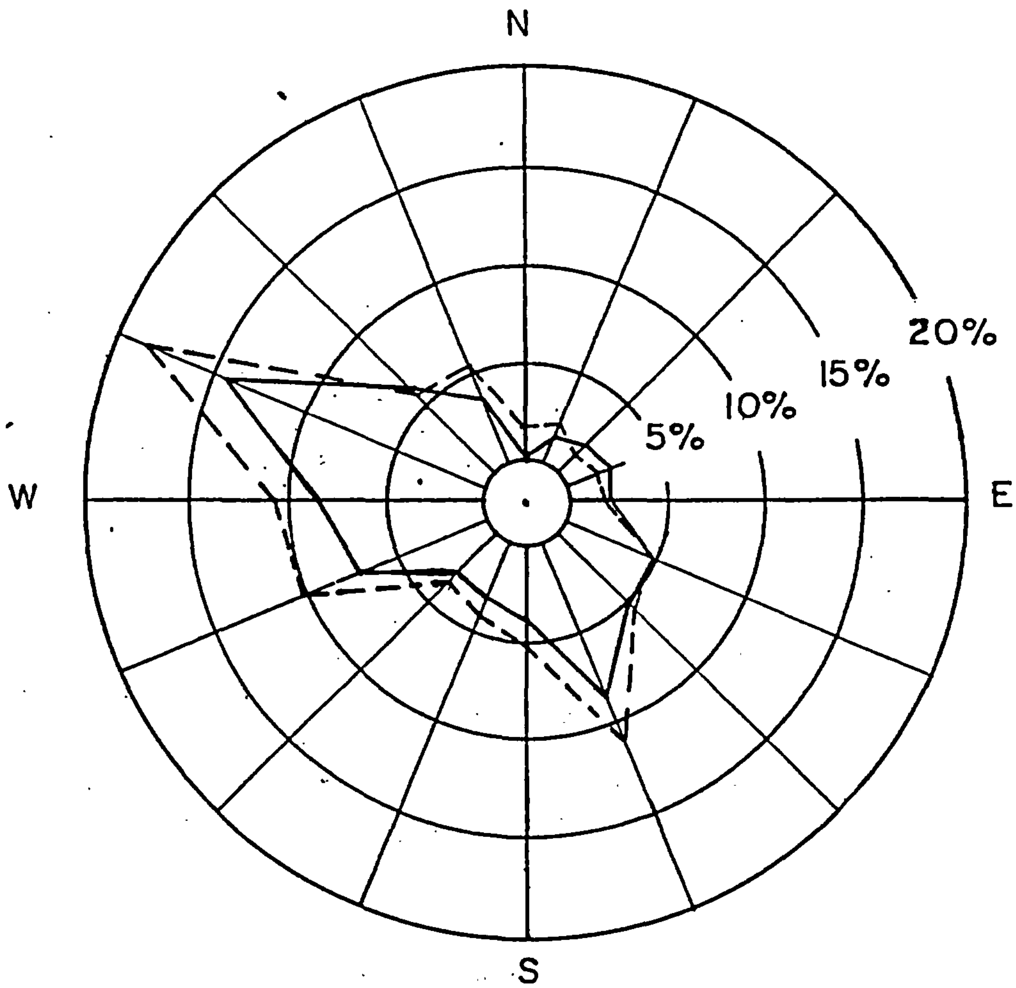
The general climate of Pennsylvania is a modified continental type with occasional intrusions of Alantic maritime air from the east. The area has generally adequate rainfall without extreme variations from year to year. Temperatures have, in general, a continental range with hot summers to cold winters, ranging from over 100°F to less than -30°F. The prevailing wind across the area is westerly although the detailed wind movement is very greatly influenced by the small scale topography. More specific analyses of the individual elements, particularly those affecting diffusion of material by the atmosphere, follows.

1. Surface Wind Direction. The hourly wind observations for an 8 year period, 1948-1955, for the Philipsburg Airport were studied in detail. Table A-4 presents the annual percentage frequency of the wind direction at various times and under various weather conditions. It is immediately evident that there is little variation of the most frequent winds from day to night, during periods of precipitation, and also when the visibility was equal to or less than 6 mi. These figures show that, on the average, the distribution of wind directions will be about the same regardless of the type of weather that is occurring. A detailed examination of the seasonal variations show that this holds true for all four seasons. The only major variation with season is that the west and northwest winds are more frequent during the winter as would be expected and that the highest wind velocities occur during the spring. The exposed nature of the area results in somewhat higher wind speeds than would occur in locations near sea level. Wind speeds for the daylight hours vary from a maximum average of 12.4 mph in the spring to 8.2 mph during the summer months. The nighttime speeds are somewhat lower with the highest average speed of 10.1 mph occurring in the wintertime and the lowest speed of 4.2 mph occurring in the summer. The frequency of calms follows the same pattern. The maximum number of calms occur during the summer night when 39% of the time the wind is less than 1 mph. Figure A-1 shows the remarkably constant prevailing wind directions with various wind conditions somewhat more graphically than does the table. It can be easily seen that about 30 to 40% of the winds are from the west-southwest through west-northwest and generally speaking, 20% of the remaining winds are from southeast to south quadrant regardless of the weather conditions occurring at that time.

TABLE A-4

Annual Frequency of Wind Directions (Percent) and Average Speed (Mph)

Direction (Windspeed ≥ 4 mph)	Daylight (07-1700 EST)	Night (18-0600 EST)	During Precipitation	During Low Visibility
N	1.7	0.9	0.8	0.8
NNE	2.1	1.4	1.4	1.3
NE	1.2	2.1	1.1	1.5
ENE	1.6	2.3	2.0	2.4
E	1.7	2.0	2.3	2.5
ESE	4.9	4.9	7.1	7.1
SE	5.5	5.1	6.8	7.3
SSE	11.1	8.6	11.7	12.2
S	5.1	3.8	4.2	5.0
SSW	4.1	3.1	5.5	4.3
SW	3.5	3.8	2.4	3.2
WSW	10.1	7.0	9.2	8.0
W	10.6	8.3	10.2	8.6
WNW	18.8	14.3	21.9	13.1
NW	5.7	6.1	5.2	3.4
NNW	5.4	3.4	3.2	2.4
3 mph and calm	10.1	30.9	8.5	23.8
Average speed, mph	10.4	7.3	11.8	8.3



--- DAY ≤ 3 mph AND CALM 10.1%
AVERAGE SPEED 10.4 mph.

——— NIGHT ≤ 3 mph AND CALM 30%
AVERAGE SPEED 7.3 mph

FIG. A I ANNUAL FREQUENCY AND WIND DIRECTION

Highest wind speeds are generally observed with westerly winds. The maximum wind speed observed during this period of record was 50 mph with an instantaneous peak gust reaching 60 mph. It should be noted, however, that the wind observations were taken from an instrument not equipped with a recorder. Thus, higher velocities may have occurred without being observed. It is estimated that rare wind gusts which might reach as high as 80 to 90 mph are not improbable.

In addition to the high wind gusts previously discussed, mention should be made of the possibility of tornado occurrence at this site. While tornadoes are not particularly common in this section of Pennsylvania an analysis of 35 years of record shows that during this period five tornadoes occurred in the area covered by the five counties surrounding the Quehanna site. Because of the usually short path covered by any single tornado and the small width of this path, it would be impractical to assign probabilities of a tornado striking any particular building or installation. However cognizance should be taken of the fact that this phenomenon can occur at this site.

The data on winds occurring with precipitation was included in order that one might consider the effect of washout of possibly airborne contaminants. The wind frequency during periods of low visibility was included as a method of estimating the wind direction during periods of atmospheric stability. It is of considerable interest to note that since these do not differ markedly from the day or night wind frequencies no special consideration of variation in weather conditions seems necessary in considering the transport of pollutants by the wind.

Another point of uniformity that can be noticed in the wind at the area is the distribution of wind speeds with various weather conditions. Table A-5 illustrates the annual frequency of various wind speed classes. It is noted that by far the largest proportion of the winds are between 4 and 12 mph averaging over 50% in all circumstances. The second largest occurrence is in the 13 to 24 mph category, although an appreciable fraction of the time the winds are calm, particularly during the night or in periods of atmospheric stability as represented by the low visibility condition.

2. Winds Aloft. It is not expected that under normal conditions the winds more than a few hundred feet above the surface will be of particular concern at this site. However, these were examined for the Pennsylvania area and, as might be expected, the general flow is from the west and northwest with the velocities increasing steadily as the elevation above the surface increases.

TABLE A-5

Annual Frequency of Wind Speeds (Percent)

Mph	Calm	1-3	4-12	13-24	25-31	32-46	> 47
Daylight	6.8	3.4	59.8	27.3	2.2	0.4	< 0.1
Night	23.0	7.9	51.2	16.3	1.3	0.3	< 0.1
During precipitation	5.1	3.4	52.4	34.4	3.8	0.9	< 0.1
Visibility $\leq$ 6 mi	16.9	6.9	55.1	19.1	1.7	0.4	< 0.1

3. Precipitation. The period of record at the Philipsburg Airport is too short to permit the computation of "normals", but from comparison of actual amounts of precipitation at Philipsburg with that of nearby stations the average annual precipitation at this site is estimated to be between 40 and 45 in. per year. The period with most precipitation is generally May through July and the least amounts are recorded in November and December. The range of average precipitation is from about $2\frac{1}{2}$ in. per month at minimum periods to around $4\frac{1}{2}$ in. per month at the time of the rainy season. Table A-6 shows the average number of days with precipitation equal to or greater than certain specified amounts. From this table it can be seen that precipitation amounts equal to or greater than a tenth of an inch will occur about 25% of the days in a year. Heavy amounts of half an inch are much less frequent. It should be noted, however, that precipitation is extremely variable. This is borne out by the range of precipitation occurrence which is presented in Part B of Table A-6. The central Pennsylvania area, including the Quehanna site, is subject to storms producing heavy precipitation. These storms may occur in any season in the year but high intensity short duration rainfall can be expected with considerable frequency during the spring and summer months with the passage of thunderstorms over the area. Table A-7 is a listing of the maximum precipitation recorded during the period of record at the Philipsburg Airport between 1944 and 1955. However, heavier storms have occurred in the immediate vicinity particularly the very heavy rainfall accompanying the storm of July 17 and 18, 1942. This storm was centered somewhat to the north and to the east of the Quehanna site; however, the rainfall intensities for various geographical areas are included in Table A-8 to give some estimate of precipitation amounts that are possible. A storm of this intensity would be very rare. Lesser rainfall amounts but still quite heavy are not too uncommon and recordings of an inch or more in a six-hour period occur most often during the months of May through August. An analysis of the precipitation record shows that during the year 1948 no such amounts were recorded while in 1954 an inch or more in six hours was recorded on nine separate days.

Much of the wintertime precipitation will be recorded as snow. In the absence of direct measurement of the site the variation of snowfall from point to point due to the irregular topography is such that no exact estimate of seasonal snowfall is advisable. However, it can be expected that somewhat more than 40 in. of snowfall will be recorded each winter although much of this can be expected to melt off and not accumulate throughout the entire season. Heavy snowfalls in the short period are not uncommon and the Philipsburg Airport has recorded four snowfalls in 13 years that exceeded 10 in. in 24 hours. The maximum snowfall recorded during any one 24 hour period was 12.6 in.

TABLE A-6

(a) Average Number of Days of Precipitation

Inches	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Annual
≥ .10	8	10	9	9	10	8	7	6	6	6	8	6	89
≥ .50	1	2	3	3	3	2	2	2	2	3	2	2	27
≥ 1.00	*	*	1	1	1	1	1	1	1	1	1	1	9

(b) Range of Precipitation Occurrences

≥ .10	from 72/year to 113/year
≥ .50	24/year to 38/year
≥ 1.00	3/year to 15/year

*Less than 1/2

TABLE A-7

Maximum Precipitation

Duration (Hours)	Amount	Date
1	1.38	Aug 1947
2	1.96	May 1944
3	1.97	May 1944
6	2.98	May 1953
12	4.20	Nov 1950
24	4.20	Nov 1950
48	4.68	Nov 1950

TABLE A-8

Maximum Average Rainfull Depth
Storm of 17-18 July 1942 New York-Pennsylvania

Area (Miles^2)	Duration (Hours)			
	6	12	18	24
Station	30.7	34.3	35.5	35.5
1	29.3	32.0	33.8	34.2
5	26.4	28.6	30.5	31.0
10	24.7	26.7	28.7	29.2
20	22.8	24.8	26.8	27.4
50	19.7	21.9	24.1	24.6
100	16.4	19.4	21.8	22.4

4. Atmospheric Stability. Measurements of the vertical temperature distribution are not made in the Quehanna area nor are such measurements available from any locale near enough to be considered truly representative. However, measurements made at other locations have shown a high degree of correlation between low wind periods, restricted visibility and the occurrence of inversions. Conversely, high wind speeds and good visibility are indicative of lapse conditions and good diffusion weather. Examination of the length of time visibility was equal to or less than 6 mi provides some rough measure of the occurrence of stable conditions. These conditions occur about one-third of the time. Further examination of the wind record shows that low wind speeds of less than 4 mph occur at about an equal percentage of the time so that roughly it can be estimated between 30 and 45% of the time stable conditions will occur. These occurrences will be mostly during the nighttime hours. A considerable variation can be expected over the site however, with inversions being much less frequent at the top of the plateau than at the narrow deep ravines penetrating the site. Inversion duration in the ravines may be half again as long as on the plateau top provided that their orientation is such that they are protected from the sun in the early morning and late afternoon.

While inversions form nearly every night there is nothing in the available records which could be interpreted as signifying that the Quehanna area experiences any unusual poor stability conditions. On the contrary, the fairly high elevation and the high average wind speed would indicate that good atmospheric dispersion would be obtained during the majority of the daylight hours.

5. General Weather Conditions. The average monthly temperature for the Quehanna area will range from about 65° in July to a low of nearly 22° in January, with an estimated annual temperature of around 44°. The area can expect generally cool nights with considerable temperature extremes between the hilltop areas and the valleys. The nighttime minimum temperatures in the vallys will be much lower than on the plateaus. Table A-9 lists the occurrence of various weather phenomena in terms of the average number of days per year. These data were obtained from the 8 years of record at the Philipsburg Airport. Two items of particular interest are the large number of days per year when the minimum temperature will fall below 32°. This may become important in construction activities since almost half of the year some protection will have to be made against freezing temperatures. The second item is the occurrence of thunderstorms. Although a straight numerical average would indicate approximately three thunderstorms per month the majority occur in the spring and summer and average from six to ten per month during these seasons.

TABLE A-9

Occurrence of Weather Phenomema
(Average Number of Days per Year)

Thunderstorms	Visibility $\leq$ 1/4 mi	Snow $\geq$ 1 in.	Temperature $\leq$ 32	Temperature $\leq$ 0
33	69	16	157	8

TABLE A-10

Percent of Time Visibility $\leq$ 1 mile together with Ceiling $\leq$ 500 Ft

Winter	11%
Spring	7%
Summer	3%
Fall	6%

Average Annual 6.6%

Since there is a possibility of operating aircraft to and from the Quehanna site a tabulation was made of the number of occurrences of ceilings 500 ft or less together with visibilities of 1 mi or less. Table A-10 lists these data. From this it can be seen that the winter and spring seasons have the most frequent occurrence of adverse flying conditions. A comparison of the low visibility wind tabulations shows that the majority of adverse weather occurs with south-southeast or west-northwest winds.

E. Conclusions

The major conclusion to be drawn from this preliminary study is the skewed distribution of wind direction that apparently is relatively unchanged by the occurrence of various types of weather. Since the most frequent wind direction is west-southwest and the second most frequent south-southeast, it should be possible to orient facilities at the site to avoid cross contamination or interference of one by the other. Quantitative estimates of diffusion will have to be based on more precise data obtained from the site itself since there will undoubtedly be anomalies created by drainage flow into the valleys from the plateau.

APPENDIX III -- DERIVATION OF THE INHALATION DOSE EQUATION

$$D = \frac{A\ 3.7 \times 10^{10}(\text{d/sec/curie})E\ 1.6 \times 10^{-6}(\text{ergs/Mev})86{,}400(\text{sec/day})}{M \times 93\ (\text{ergs/gm/rep})}$$

D = Dose in rep/day

A = Cureis in critical organ

E = Average energy per disintegration

M = Mass of critical organ, gm

$$D = 5.5(10^7)\frac{AE}{M}$$

$$A = \text{TID}\left(\frac{c - \text{sec}}{m^3}\right) B\left(\frac{m^3}{\text{sec}}\right) f\ e^{-\lambda t}$$

TID = Total integrated dose

B = Breathing rate; taken as 20L/min or $3.3(10^{-4})m^3/\text{sec}$

f = Fraction of inhaled activity reaching critical organ

λ = Effect decay constant (day^{-1})

$$\text{TID} = \frac{2Q}{\pi C^2\ ux^{2-n}}$$ from Suttons equation. (Reference 10)

The integrated dose from time of inhalation to time t is

$$\text{Integrated dose} = \int_0^t Ke^{-\lambda t}\,dt$$

$$= \frac{K}{\lambda}\left[1 - e^{-\lambda t}\right]$$

The integrated dose in rep is therefore

$$D = \frac{1.16(10^4)Q}{C^2 \quad ux^{2-n}} \frac{Ef}{M\lambda} \left[1-e^{-\lambda t}\right]$$

Using the values of E, f, M and λ from Handbook 52 (Reference 12)

Isotope	$\frac{Ef}{M\lambda}$
Sr-90 + Y-90	0.123
Sr-89	1.31 (10^{-3})
I-131	1.83 (10^{-2})
Y-91	8.4 (10^{-4})
Ba^{140} + La^{140}	5.27 (10^{-4})
Ce^{144} + Pr^{144}	4.85 (10^{-3})

A breathing rate of 20 L/min is used which represents the rate for a "standard man" working. A non-working standard man breathes at 10 L/min. The values given in Handbook 52 for isotope retention are crude since retention, for example, depends on the particle size inhaled and also experimental data is often meager. In addition, Handbook 52 assumes an exponential loss of the isotope from the body whereas in fact the loss can follow a power function. However these uncertainties are probably not greater than those involved in estimating diffusion.

APPENDIX IV - REACTOR CHECKOUT PROCEDURE

1. Make sure bridge is locked in position. ____

2. Turn on lights which illuminate core. Make sure that source is in proper position, and that loading is as indicated by the reactor log book. ____

3. Set zero on micro-microammeter. Make sure that this instrument is on the most sensitive range. ____

4. Read and record CIC power supply voltages.
   1. + ____ - ____
   2. + ____ - ____

5. Record reading on power integrator. ____

6. Check Jordan Rams system. ____

7. Energize recorders. Manually standardize recorders after recorders servo systems are operating. Check chart supply. ____

8. Calibrate log count ratemeter by turning switch to "Calibrate" position and adjusting until recorder indicates 60 cps. Return selector switch to "Use" position. ____

9. Make sure that fission chamber is in "Insert Limit" and that count rate recorder is indicating at least 2 cps. ____

10. Record gain setting on A-1 amplifier, and PHS setting on both A-1 and scaler.
    A-1 Gain ____
    A-1 PHS ____
    Scaler PHS ____

11. Withdraw fission chamber and make sure that count rate drops. Reinsert fission chamber. ____

12. Energize electromagnets by turning magnet key switch. Observe magnet current meters on safety amplifiers to make sure current is flowing. ____

13. Test #2 (power level) safety system by withdrawing all shim-safety rods about three (3) inches and inserting a probe in the opening marked "Scram" on #2 safety amplifier panel. All rods should drop. ____

14. Test #3 (power level) safety system in same manner. ____

15. Calibrate LOG N - Period amplifier:
(a) Turn selector switch (located on chassis of instrument) to ground position. LOG N meter should read 0.001 (extreme left black mark on meter). To adjust, use adjustment marked "Ground." ________

(b) Turn selector switch to "Lo Calibrate." The meter pointer should now be aligned with the left edge of the red mark on the left side of the meter. To adjust; use adjustment marked "Calibrate." ________

(c) Turn selector switch to "Hi Calibrate." The meter pointer should now be aligned with the left edge of the red mark on the right side of the meter. To adjust, use adjustment marked "Gain." ________

(d) Due to interactions between adjustments, if any adjustments were made in the steps above, the entire sequence should be repeated until no further adjustment is required. ________

(e) Turn selector switch to "Ground." ________

(f) Disconnect ion chamber input to LOG N - Period Amplifier and connect cable from Pile Period Simulator. ________

(g) Turn selector switch to operate. ________

(h) Use the Pile Period Simulator to check the calibration of the Pile Period circuit. Readings on the meter should be within 10% of all settings on the simulator. ________

(i) Raise shim-safety rods 3 inches and set 1 second period on simulator; all rods should drop. ________

(j) Return selector switch to "Ground" and reconnect ion chamber to LOG N - Period amplifier. ________

(k) Turn selector switch to "Operate." ________

16. Make sure that there are no keys in any of the interlock defeat switches. ________

17. Check condition of annunciator lamps by pushing "Test" button. ________

18. Make sure lamps in rod position indicating system are glowing (dimly). ________

19. Read and record magnet currents. #1.________ #2.________ #3.________

20. Prepare cooling system for operation. Select convection cooling for 100 kw or less and forced cooling for over 100 kw.
Do NOT attempt to change from one type of cooling to the other after the reactor has been started up, as this will automatically shut down the reactor. ________

(a) Convection cooling:
   1. Open plenum chamber door. ________

(b) Forced cooling:
   1. Close plenum door.
   2. Start pump.
   3. Read and record water temperature and flow rate after flow has stabilized.

Water Temp. ________

Flow Rate ________

21. Make sure that all annunciator indications are normal. ________

Operator ______________________ Date ______________________

Supervisor ______________________ Time ______________________

ABBREVIATIONS

BSR	Bulk Shielding Reactor
cfs	cubic feet per second
c/m^2	curies per square meter
d/sec/curie	disintegrations per second per curie
fis/sec	fissions per second
ft	feet
gm	grams
gpm	gallons per minute
k_{eff}	effective neutron multiplication rate
kg	kilogram
kw	kilowatt
lb	pounds
m	meters
Mev	million electron volts
mi	miles
min	minutes
ml	milliliters
mp	melting point
mph	miles per hour
MPL	maximum permissible limit
mr/hr	milliroentgens per hour
m/sec	meters per second
MTR	Materials Testing Reactor
$n/cm^2/sec$	neutrons per square centimeter per second
rem	roentgen equivalent man
rep	roentgen equivalent physical
sq mi	square miles
TID	total integrated dose