Docket No. 50-39
File G.

CURTISS-WRIGHT CORPORATION
RESEARCH DIVISION
QUEHANNA, PENNSYLVANIA
AMHERST 3-4711

April 27, 1959

DOCKETED USAEC Division of Licensing & Regulation MAY 13 1959 Docket Officer

U. S. Atomic Energy Commission
Division of Licensing & Regulation
1901 Constitution Avenue
Washington 25, D. C.

Attention: Mr. Lyall Johnson, Chief
Licensing Branch
Division of Licensing and Regulation

Gentlemen:

Application is hereby made to amend CWRR Reactor Facility License No. R-36 to permit operation at power levels up to 1.7 megawatts with existing 10 plate fuel elements, and up to 4.0 megawatts with 19 plate fuel elements. In support of our application we are submitting 15 copies of CWR-4062, Hazards Evaluation Report, for 4 megawatt operation.

We have previously requested your office to issue a construction permit to modify our original facility, as described in our original Hazards Evaluation Report, CWR-431, and action has been withheld pending receipt of the attached hazards report. We have also asked that we be permitted to operate continuously at the 1 Mw level as soon as provision has been made to provide secondary cooling. This action has also been withheld until the current hazards report can be examined. We request that the section describing our new cooling system be examined first so that early action may be taken on our requests for the construction permit and for permission to operate continuously at the 1 Mw (thermal) level.

A summary of the major changes made in the revision of our Hazards Evaluation Report has been submitted as part of our application for construction permit.

Very truly yours,

CURTISS-WRIGHT CORPORATION
RESEARCH DIVISION

Wm. T. Lake

Wm. T. LAKE
Controller

Attachment:
(15) Copies CWR-4062

D-145

State of New Jersey :
: ss.
County of Bergen :

Wm. T. Lake , being duly sworn according to law, deposes and says that he is the Controller of Curtiss-Wright Corporation mentioned in the foregoing application, that he has read the said application and knows the contents thereof and that the same is true to the best of his knowledge, information and belief.

Wm. T. Lake

Sworn to and subscribed
before me this 1st day
of May, 1959.

William Hall
Notary Public

NOTARY PUBLIC OF NEW JERSEY
MY COMMISSION EXPIRES JULY 15, 1961

I, Roger W. Mullin, Jr., certify that I am the Secretary of the Corporation named, as applicant herein, that Wm. T. LAKE, who signed this application, was then Controller of said Corporation; that said application was duly signed for and in behalf of said Corporation by authority of its governing body, and is within the scope of its corporate powers.

Roger W. Mullin Jr.

CWR-4062

[illegible] No. 50-39

HAZARDS EVALUATION REPORT
CURTISS-WRIGHT RESEARCH REACTOR

April 1959

4 MEGAWATT OPERATION

CURTISS WRIGHT

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

DOCKET NO. 50-39

classification review not required [signature] 5/14/59

DOCKET NO. 50-39

HAZARDS EVALUATION REPORT
CURTISS-WRIGHT RESEARCH REACTOR

(Encl. w/ ltr. of 4/27/59)

April 1959

4 MEGAWATT OPERATION

CURTISS WRIGHT

DOCKETED USAEC Division of Licensing & Regulation MAY 1 1959 Docket Officer

CURTISS-WRIGHT CORPORATION
RESEARCH DIVISION
QUEHANNA, PENNSYLVANIA

HAZARDS EVALUATION REPORT
CURTISS-WRIGHT RESEARCH REACTOR

April 1959

4 MEGAWATT OPERATION

Prepared by:
Research Reactor Personnel
Nuclear Sciences and Engineering

Approved by:

G. A. Anderson
Chief
Nuclear Sciences and Engineering

TABLE OF CONTENTS

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 an amendment to our present 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 4000 kw, cooling is by forced circulation of water. Most of the necessary shielding is supplied by the water.

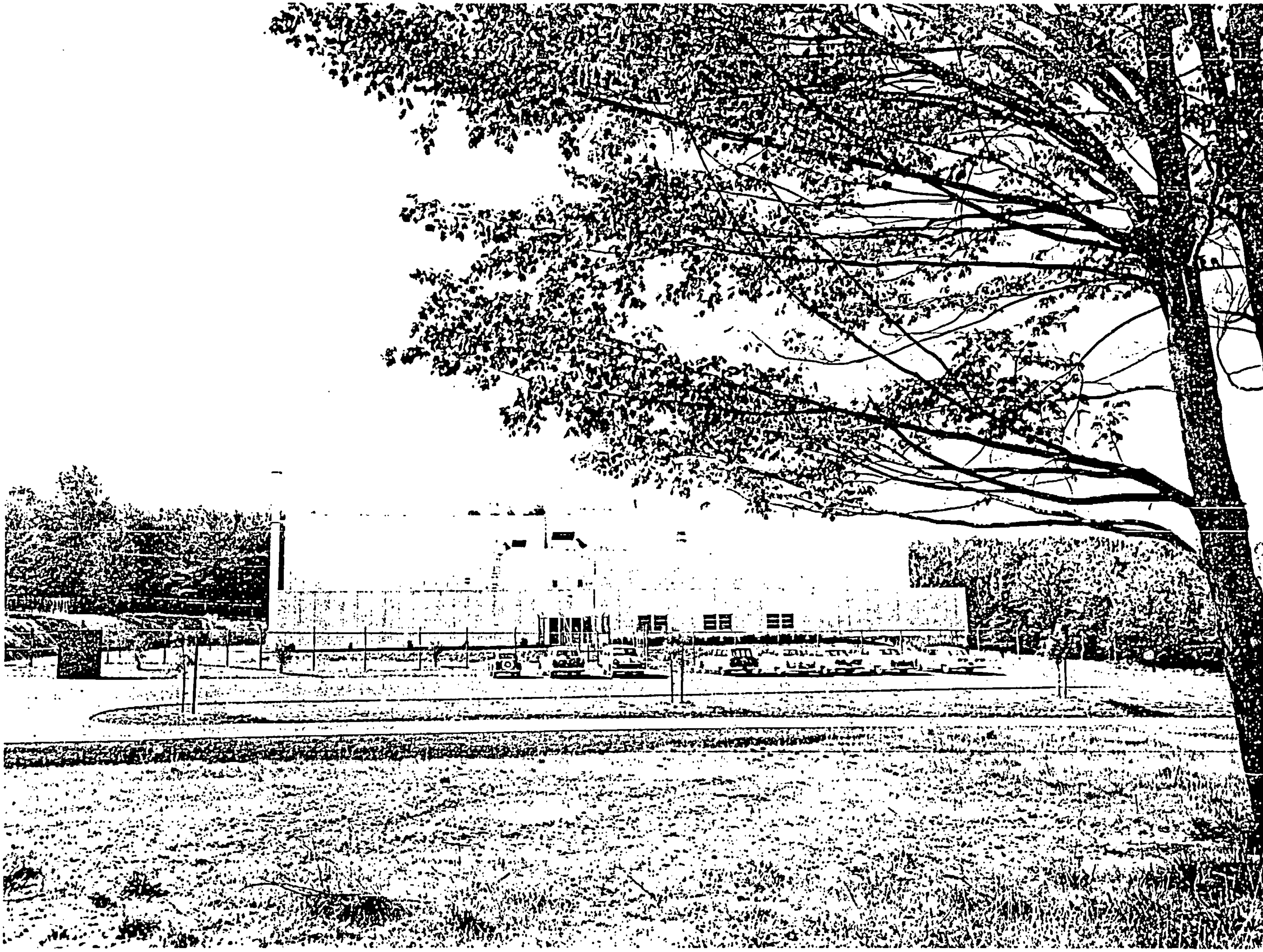
The reactor is located at the Curtiss-Wright Research and Development Center at Quehanna, Pennsylvania. This center is located on an 80 sq mi tract of land surrounded by a low population density area in the north central portion of the state. One of several research facilities at this site is the Research Reactor and Radioactive Materials Laboratory, the exterior of which is shown on page 8. 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.

In Section 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 omitted purposely from the description. The matter of operational procedures and safety devices is considered in Section II.

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.



RESEARCH REACTOR AND RADIOACTIVE MATERIALS LABORATORY

I. THE REACTOR, ITS SUPPORTING FACILITIES AND SITE

A. Reactor

Some of the more important characteristics of the reactor are tabulated in Table 1.

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 (10 plate element) 0.58 (19 plate element)
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
Coolant flow rate	700 or 1200 GPM
Coolant temperature rise	10.1°F (10 plate element) at 1.7 Mw 22.7°F (19 plate element) at 4 Mw
Biological Shield	light water, normal and high density concrete
Critical mass	2.2 Kg (graphite reflector) 2.8 Kg (water reflector)
Power level	up to 4000 kw
Average thermal flux	$3. \times 10^{13}$ n/cm^2/sec at 4000 kw with an H_2O reflector

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

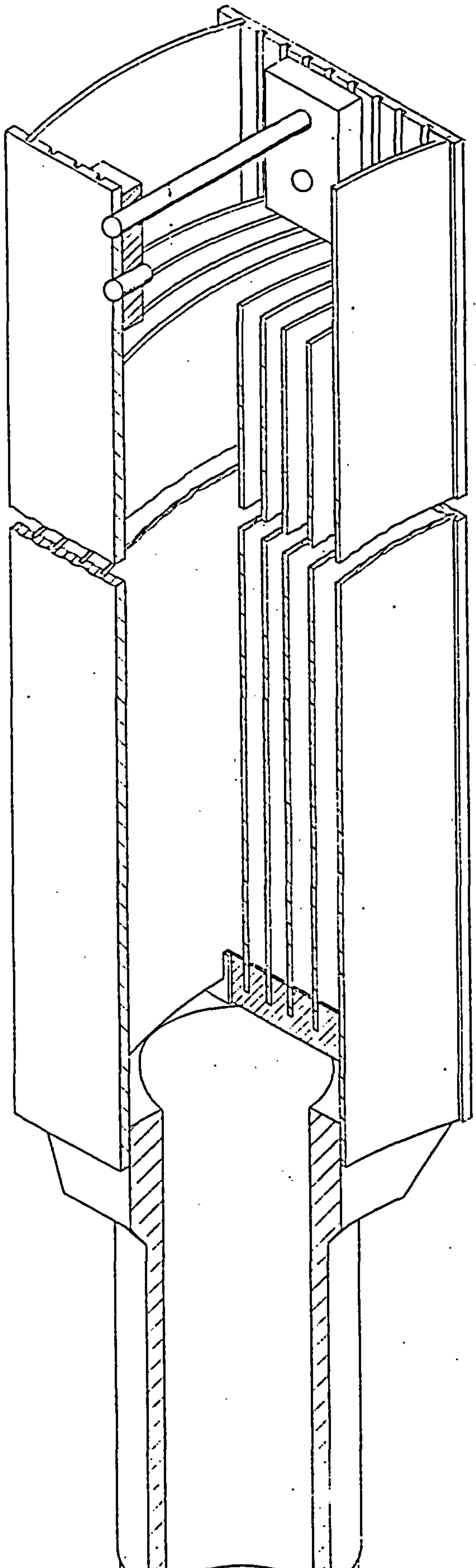
1. Elements

a. Ten Plate Fuel Element. 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 curved sandwich consisting of a 0.020 in. thick layer of aluminum-uranium alloy covered on each side by a 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, 23.5 in. in length, and will contain 17 grams of U-235. The finished plate will be approximately 2.8 in. wide, 24.625 in. long and 0.060 in. thick.

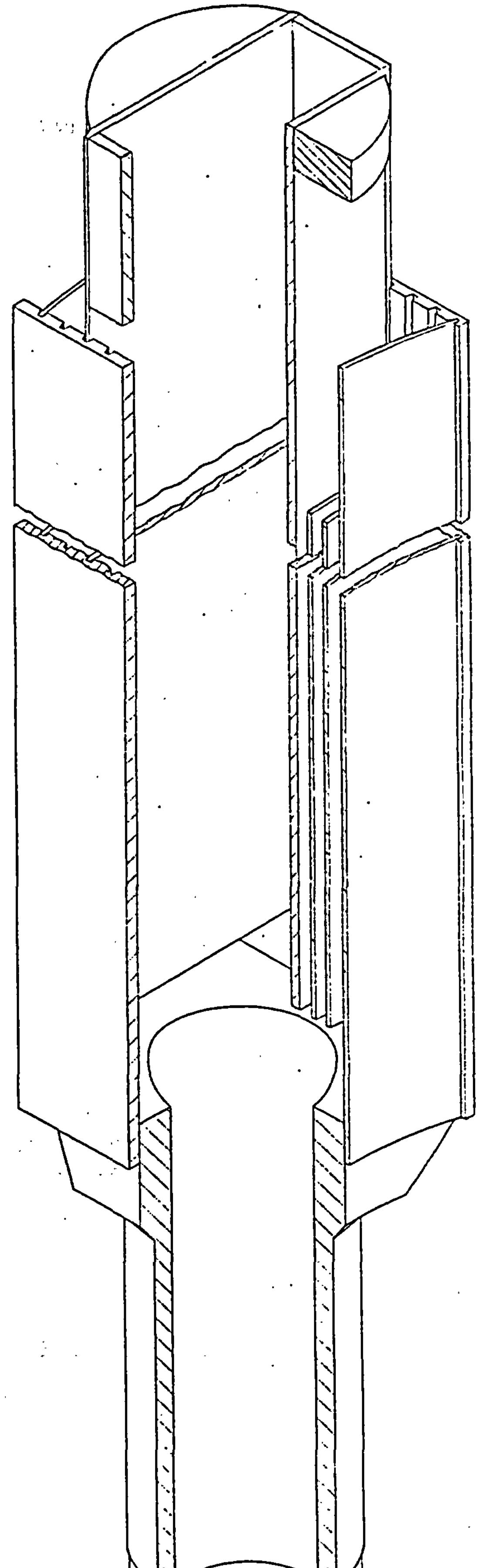
The fuel plates are fastened into groups of ten with aluminum side plates so that the finished element has an almost square cross-section measuring 3 in. by 3 in. At one end a male guide section of circular cross-section is attached, bringing the over-all length of an element to about 35 in. This guide piece is inserted into a hole in the grid plate which supports the entire fuel element array or core.

The element 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 on page 11. The method of inserting the element in the grid plate is indicated by the illustration on page 14. 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, proceeding from the control rod space toward the end plate, the channels between the plates have .6, .9, and .9 times the cooling flow area provided between plates in a standard element. Thus, since the .6 channel nearest the center of the element will cool only one fuel plate, an increase in cooling capacity has been provided to compensate for flux peaking in this vicinity due to the water island created by the control rod withdrawal. The design of the rod element also is shown on page 11.



FUEL ELEMENT



CONTROL ROD ELEMENT

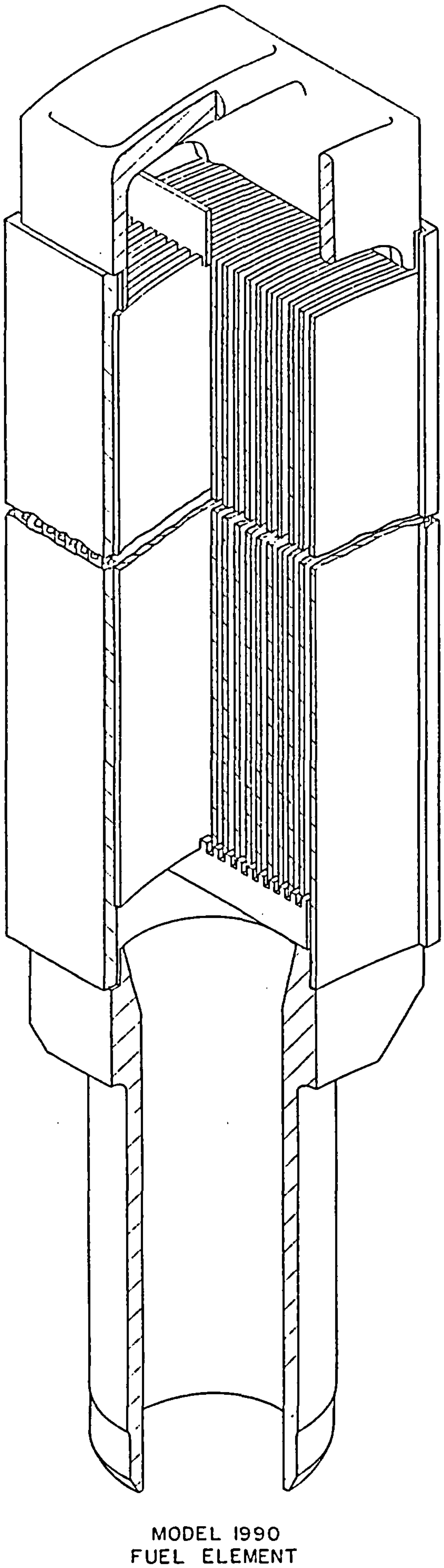
b. <u>Nineteen Plate Fuel Element</u>. The reactor core will be made up of eighteen or more Curtiss-Wright designed, MTR-type fuel elements. The standard elements will contain 19 fuel bearing plates. Each plate is a curved sandwich consisting of a 0.01 in. thick layer of aluminum-uranium alloy covered on each side by a 0.02 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, 23.5 in. in length, and will contain 10 grams of U-235. The finished plate will be approximately 2.8 in. wide, 24.625 in. long and 0.05 in. thick.

The fuel plates are fastened into groups of nineteen 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 (nosepiece) of circular cross-section is attached, bringing the overall length of an element to about 35 in. This guide piece is inserted into a hole in the grid plate which supports the entire fuel element array or core.

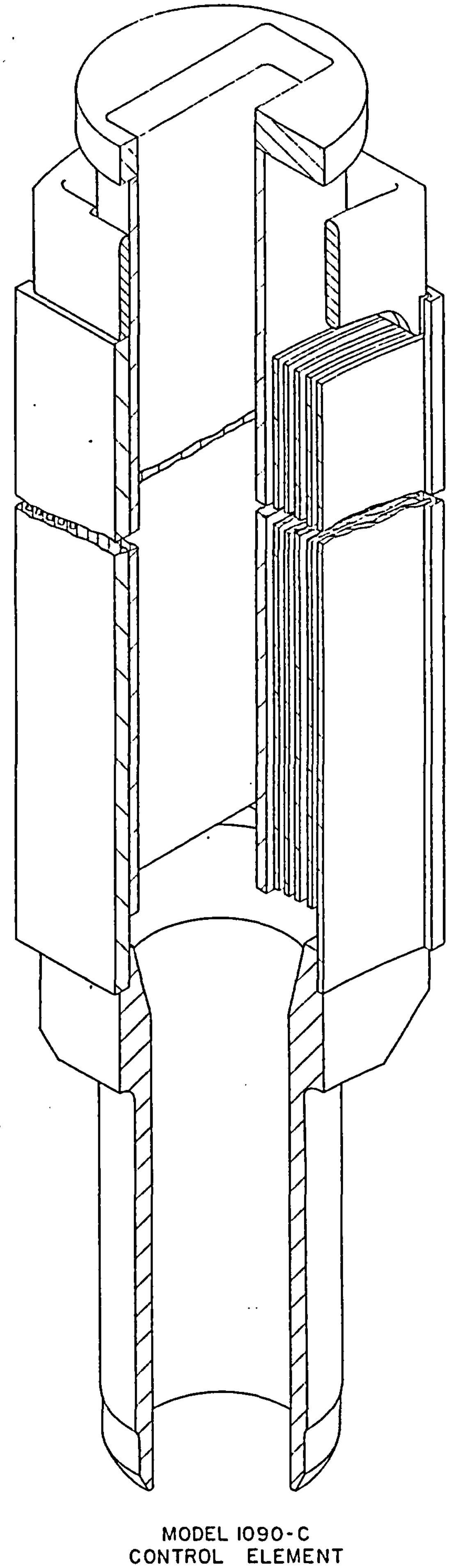
The element and grid plate are designed so that the fuel bearing plates are uniformly spaced throughout the core. Both ends of the elements are open so that cooling water may flow up or down between the fuel plates. The tolerances are set so that if all dimensions are off in the same direction there 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 on page 13 . The method of inserting the element in the grid plate is indicated by the illustration on page 14. In addition to the standard elements, there will be rod elements as shown on page 13. The rod elements have the central nine plates removed to accommodate the rod. The remainder of the plates are spaced so that, proceeding from the control rod space toward the end plate, the channels between the plates have the same cooling flow area provided between plates in a standard element. Thus, since the channel nearest the center of the element will cool only one fuel plate, an increase in cooling capacity has been provided to compensate for flux peaking in this vicinity due to the water island created by the control rod withdrawal. The design of the rod element also is shown on page 13.

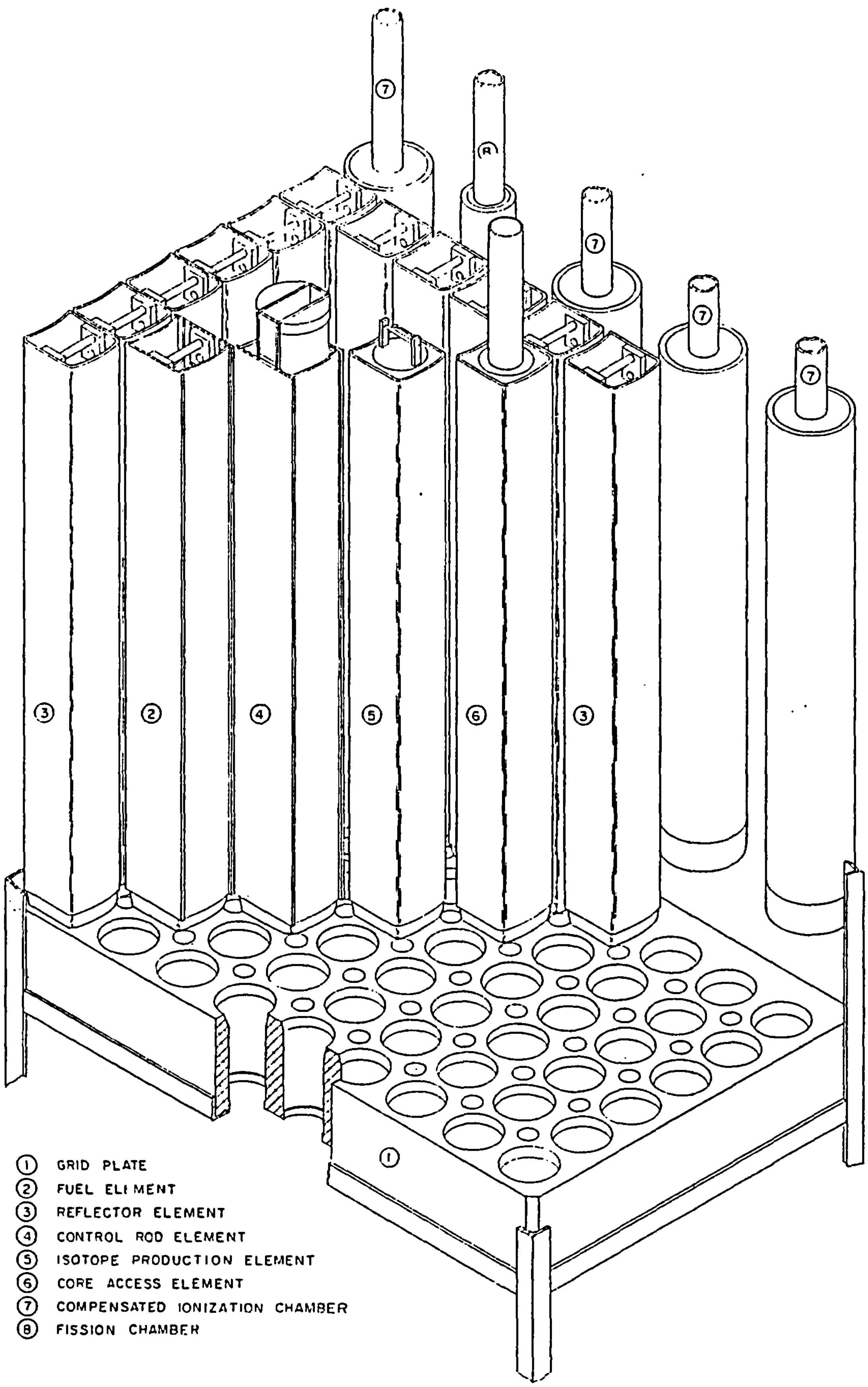
c. <u>Reflector Elements</u>. Curtiss-Wright has fabricated thirty graphite and fifteen BeO elements for use in the reactor. These will be completely encased in aluminum, and helium leak tested to insure proper



MODEL 1990
FUEL ELEMENT



MODEL 1090-C
CONTROL ELEMENT



REACTOR CORE

welding. The reflector elements have the same over-all shape and dimensions as fuel elements, but are sufficiently different in design and appearance so as to be clearly distinguishable from the fuel elements under the full depth of water in the pool. Moreover, each reflector element has a large identification number engraved on two sides. These numbers will positively identify each element so that they cannot be mistaken for fuel elements and vice versa. Reflector elements will be logged when inserted or withdrawn from core in exactly the same manner as fuel elements so as to have a complete exposure record. In addition, there are also seven graphite isotope production elements available for use in the reactor.

TABLE II

Inventory of Fuel Elements for the
Curtiss-Wright Research Reactor

Type	Sub-type	Number of Fuel Elements	Grams of U-235 per Element (Average)	Total Grams U-235	Number of Fuel Bearing Plates Per Element
10 Plate	Full	20	167.89	3357.86	10
10 Plate	Partial				
10 Plate	80%	2	135.79	271.57	8
10 Plate	60%	2	100.67	201.34	6
10 Plate	40%	2	67.15	134.29	4
10 Plate	20%	2	33.75	67.50	2
10 Plate	Rod Element	6	100.34	602.04	6
		34		4634.60	
19 Plate	Full	24	Approx 190	Approx 4,560	19
19 Plate	Rod Element	6	Approx 100	Approx 600	10
		30		Approx 5,160	

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 on page 17. 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 on page 18. 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 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.

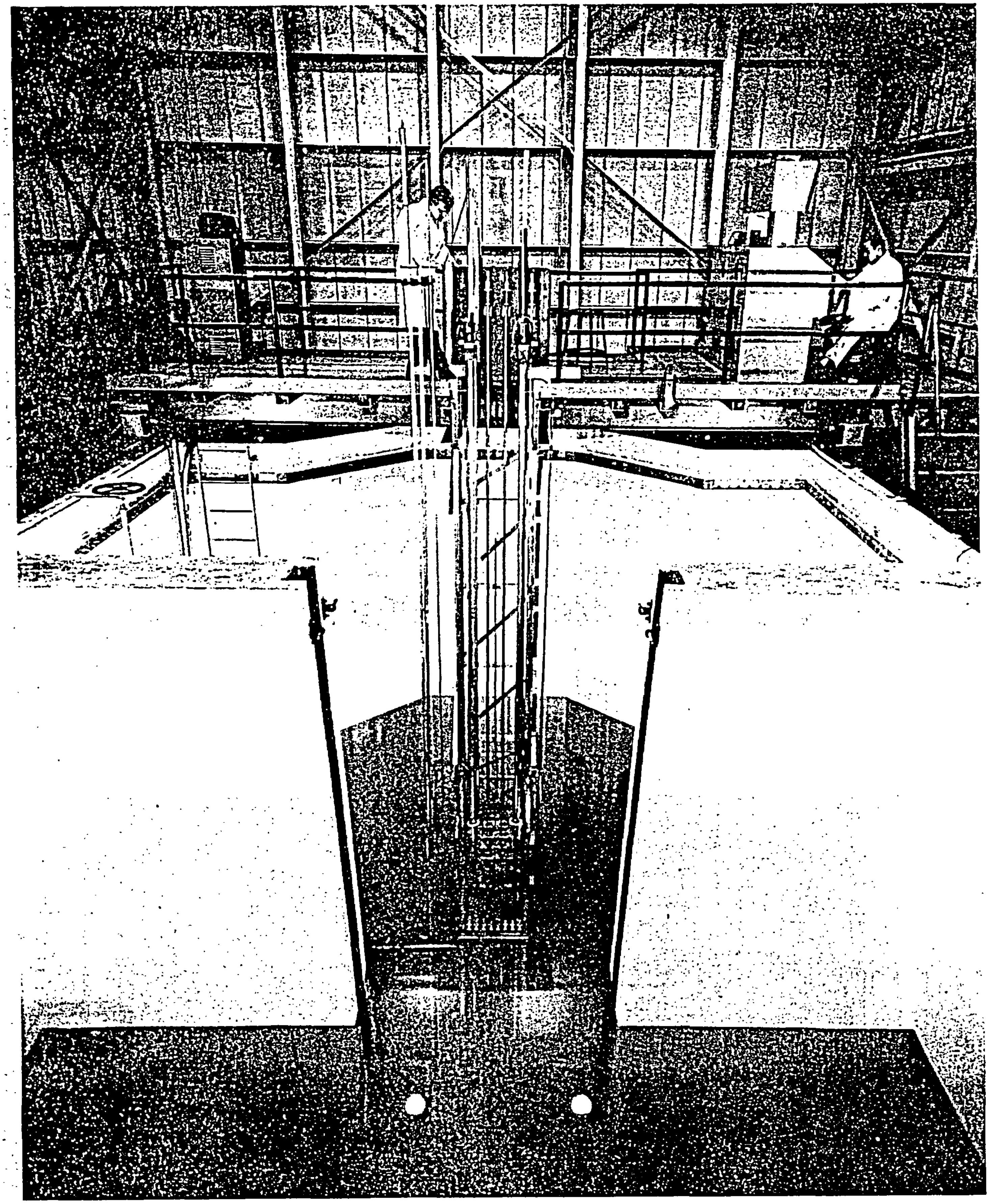
TABLE III

Characteristics of Control Rods for the Curtiss-Wright Research Reactor

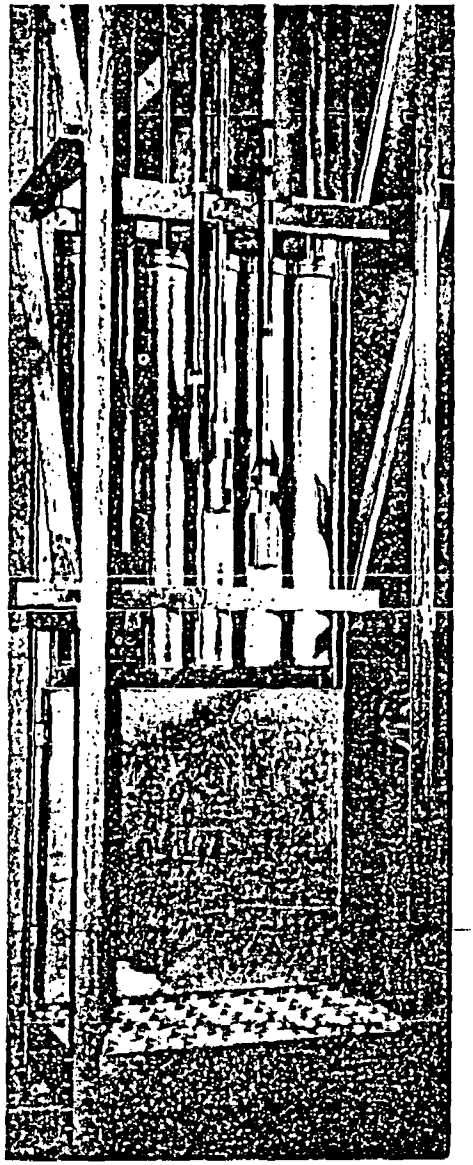
Type	Material	Number Used	Approx Worth *	Rate of Drive in./min
Regulating	Stainless Steel (.030")	1	0.29 - 0.41%	12
Regulating	Stainless Steel (.060")	1	0.5 - .70%	12
Safety-Shim	Cadmium-Boron Carbide	4	2.2 - 3.5 %	$6\frac{1}{4}$

* Actual worth dependent upon configuration

The regulating rods are made from flattened stainless steel tubes (one .030 in. and the other .060 in.) with a 2.25 x 0.800 in. cross section. Only one of the two rods is used for a given configuration. The safety-shim rods, with outside dimensions 2.25 by 0.875 in. are of laminated



General Arrangement of Grid Plate, Tower, and Bridge




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

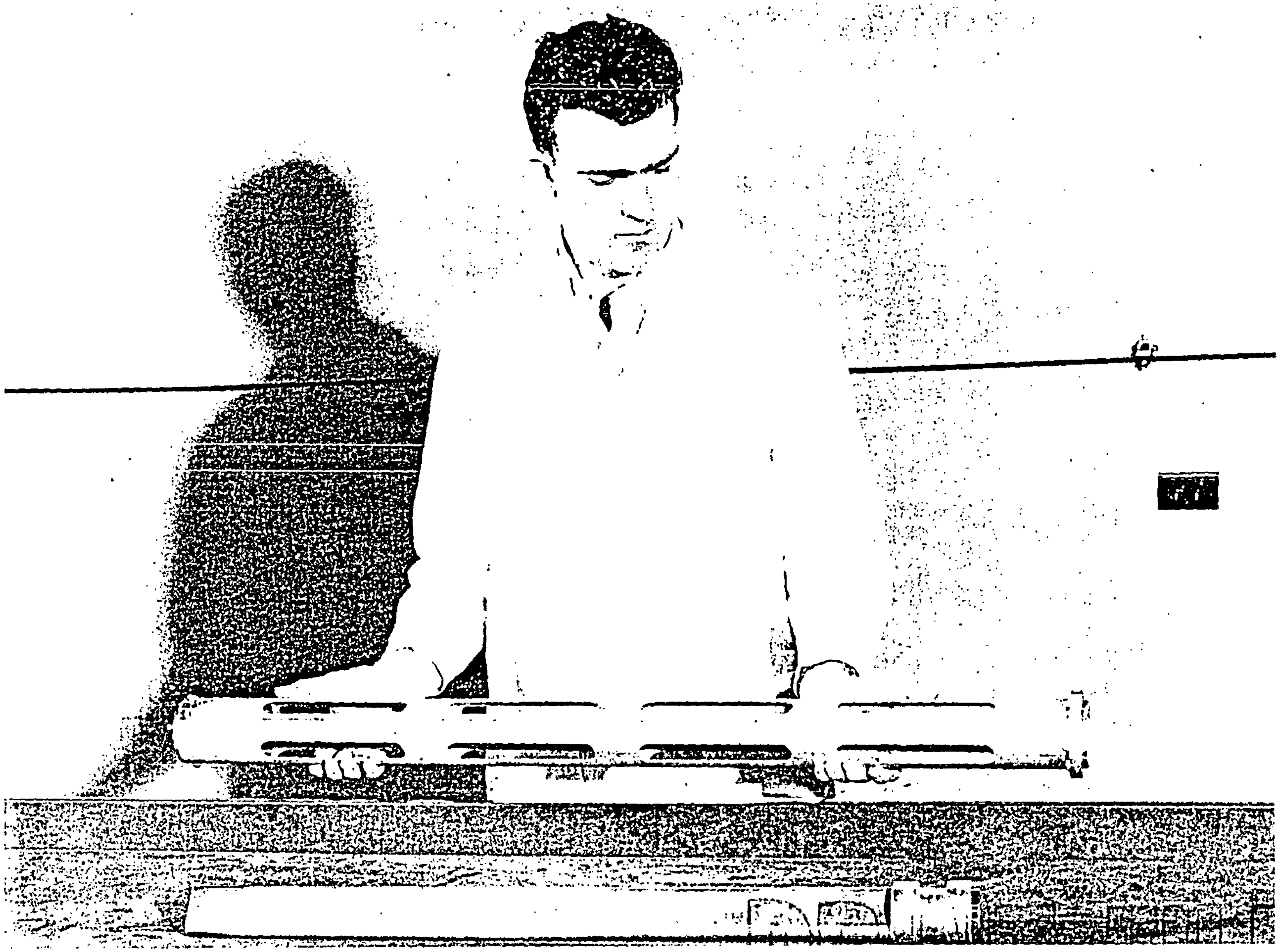
construction; a 0.065 in. stainless steel outer shell, a 0.032 in. cadmium inner shell, and the remaining cavity filled with boron carbide crystals to produce a minimum density of 1.5 gm/cm^3. The figure on page 20 shows a safety rod and its housing. The four safety-shim rods are magnetically coupled 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.

A one dimensional IBM-704 calculation indicates that reflector savings for a 3 in. thick layer of graphite backed by water are approximately the same as for a 3 in. layer of BeO again backed by water. Based on this, it is expected that the control rods will be worth approximately the same amount in a graphite and a BeO reflected core. A definite check of this situation will be made as soon as sufficient BeO elements are available to form a complete reflector. Table III gives the range of worth of the regulating and safety rods in a water-reflected and graphite-reflected core. These figures are based on Curtiss-Wright Reactor experimental values. As stated above, the values for a BeO reflected core are expected to be approximately the same as for graphite.

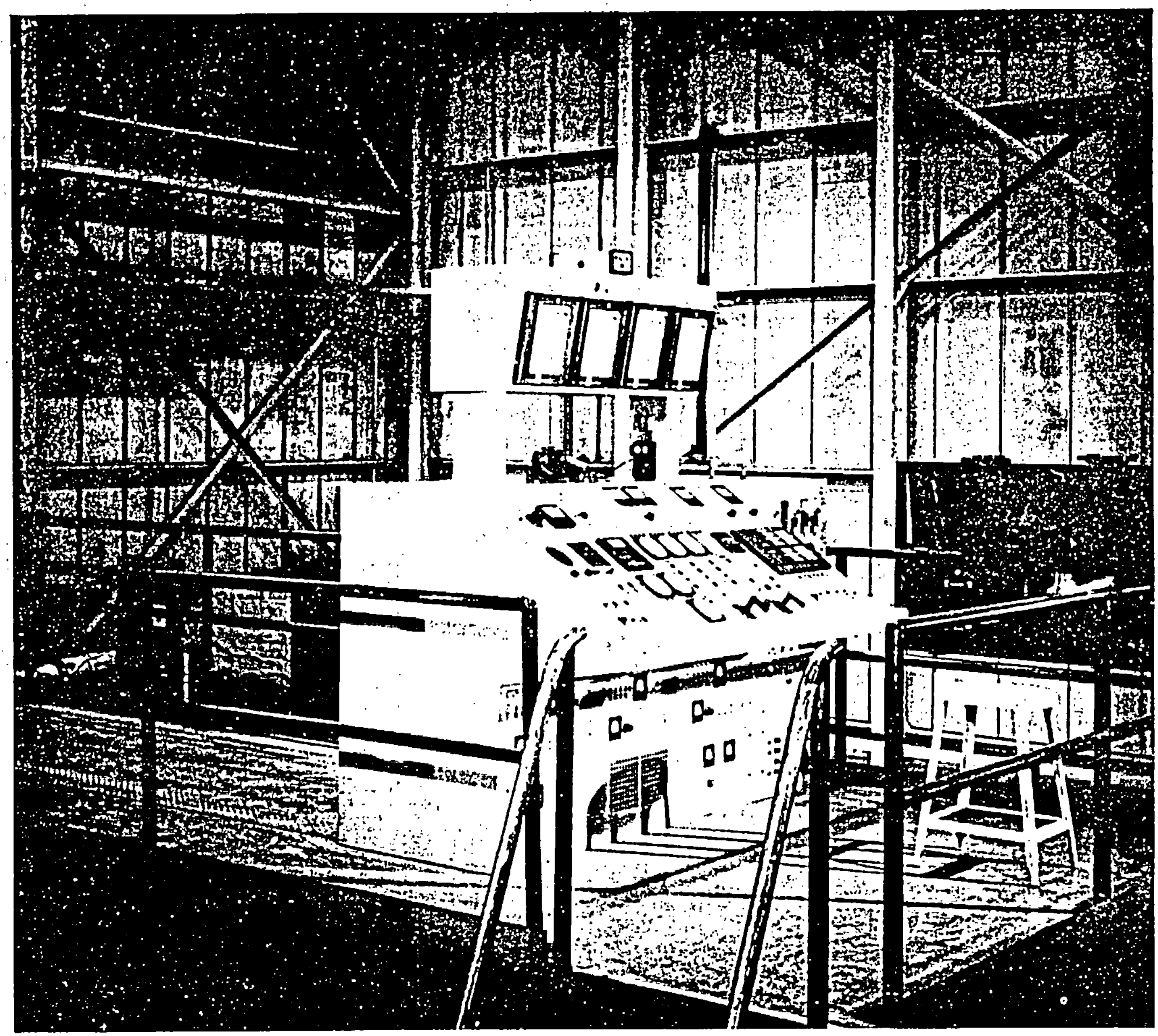
4. Control Console

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 are shown on page 21. The main unit is approximately 67 in. high with four miniature type recorders (Leeds and Northrup Type H) mounted 12 in. above it in such a manner that the operator, when seated at the console, may maintain both visual and aural communciation 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.



SAFETY ROD AND HOUSING



REACTOR CONTROL CONSOLE

Instrumentation and controls which will be added to the console for expansion to the 4 Mw level include:

a. Complete control and position indication system for fourth safety rod.

b. Meter - primary system flow rate

c. Meter - core effluent temperature

d. Meter - differential temperature across core

e. Meter - heat exchanger effluent temperature (primary)

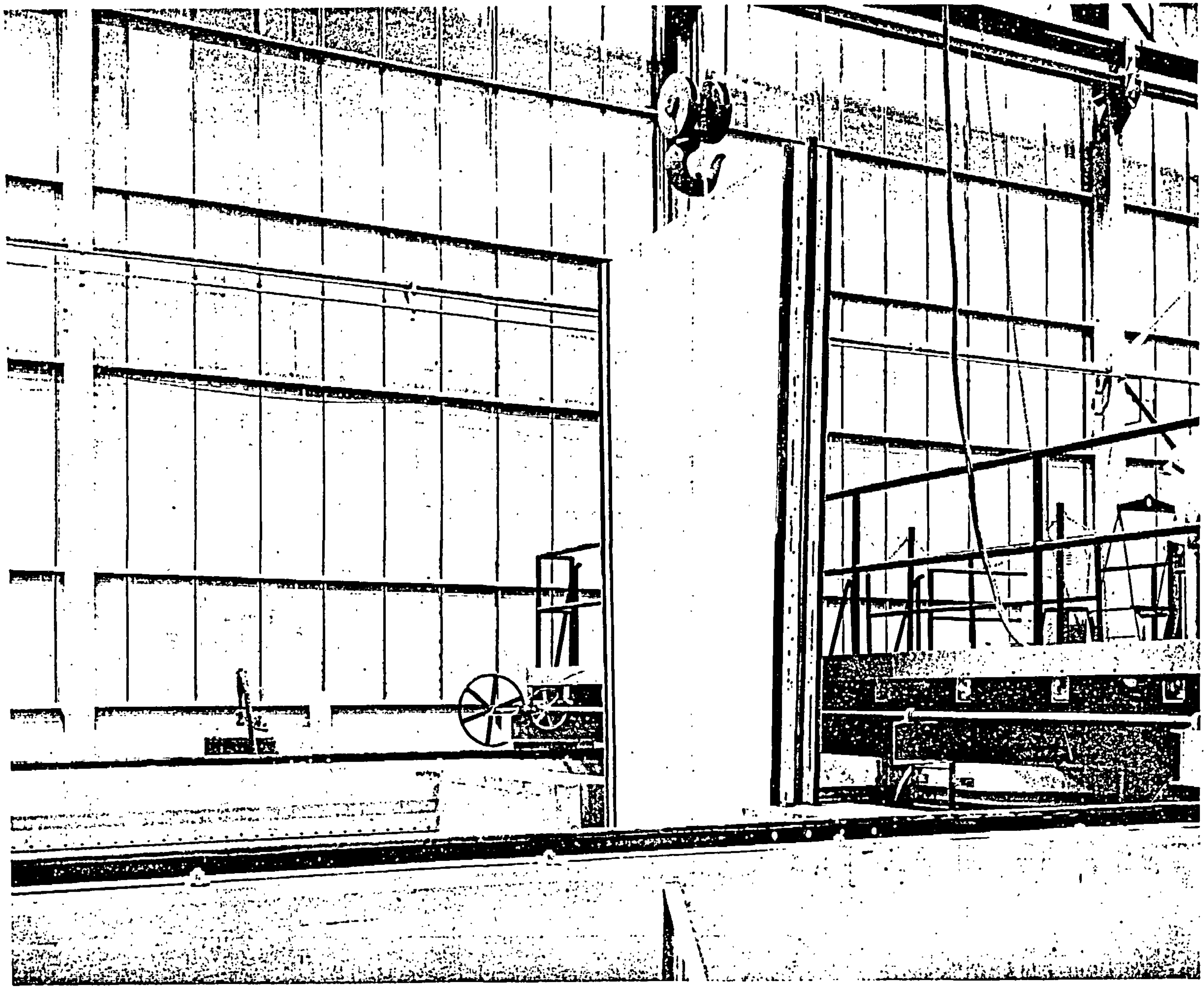
Other measuring and indicating devices for monitoring coolant system characteristics, as described in Section 6 below, will be mounted on a separate control panel.

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 26 ft long, located at the center of the concrete bulkhead. The figure on page 23 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, 18 in. thick, except at the beam hole end where the thickness is increased to 5 ft, 6 in. On the two long sides, North and South, and the East end of the pool, there is no excavation beneath the floor level as shown on page 28. Therefore, there is no possible way for personnel to approach the North, South, or East pool walls other than at the main floor level, where they are protected by a water shield at least 19 1/2 ft thick.

At the beam tube end of the pool (West) the floor is dropped to approximately the same elevation as the bottom of the pool. With the reactor in position against the beam tubes, the water thickness separating the core and area accessible to personnel is only 4 ft. Because of this, the shielding value of the 18 in. thick pool wall is augmented by 4 ft of dense concrete (> 280 lb per cu ft). This high density (ferrophosphorus aggregate) insert reaches 9 ft, 6 in. above the floor level in the beam room. The concrete sides and floor of the pool have



POOL GATE PARTIALLY REMOVED

received several coats of protective vinyl paint in order to prevent excessive leaching of minerals from the concrete into the demineralized water.

Test runs completed at one megawatt thermal power show the radiation level in the beam room to be the same order of magnitude as natural background.

There is no other area adjacent to any part of the reactor pool to which personnel could conceivably gain entrance.

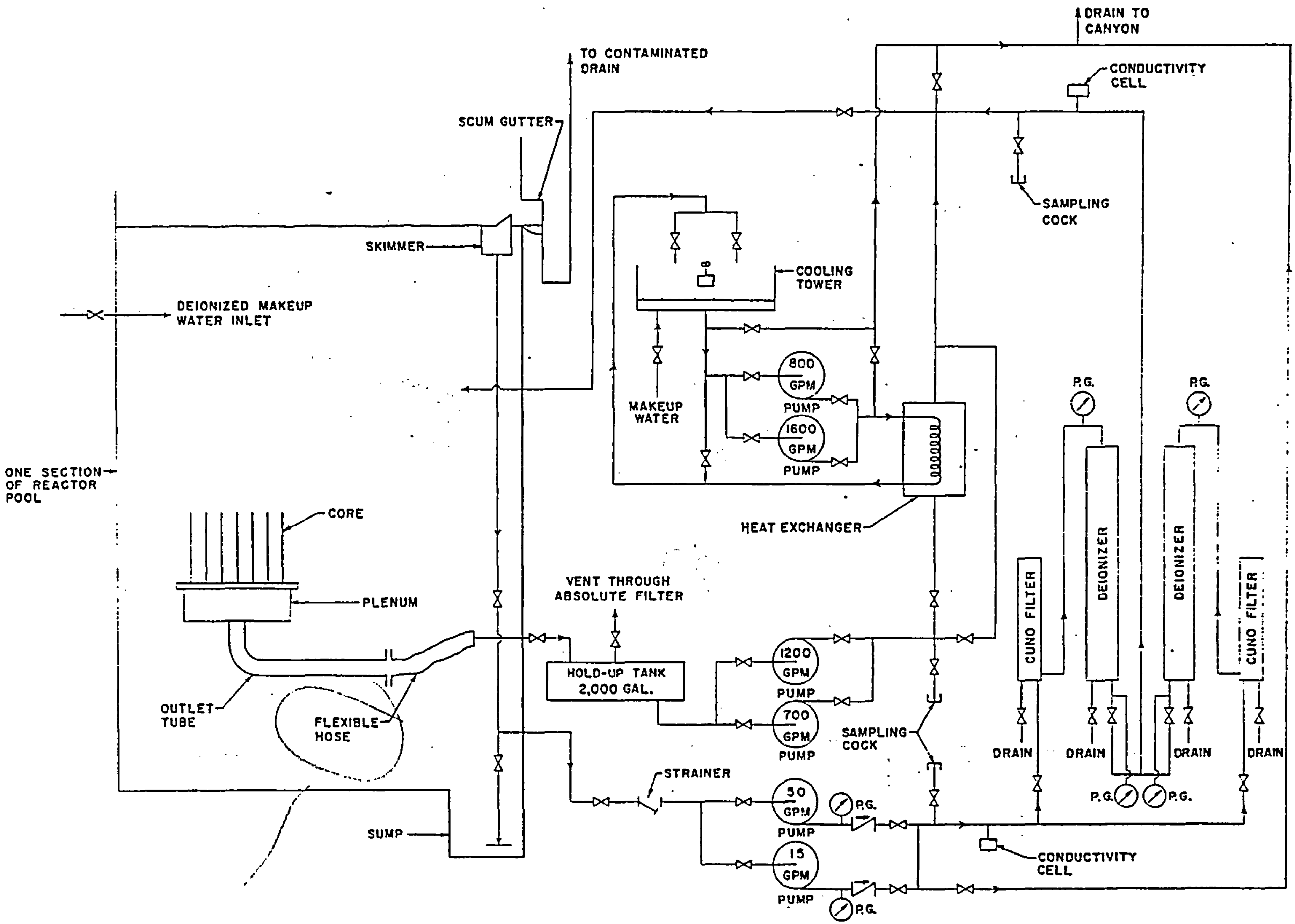
6. Cooling System

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 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.

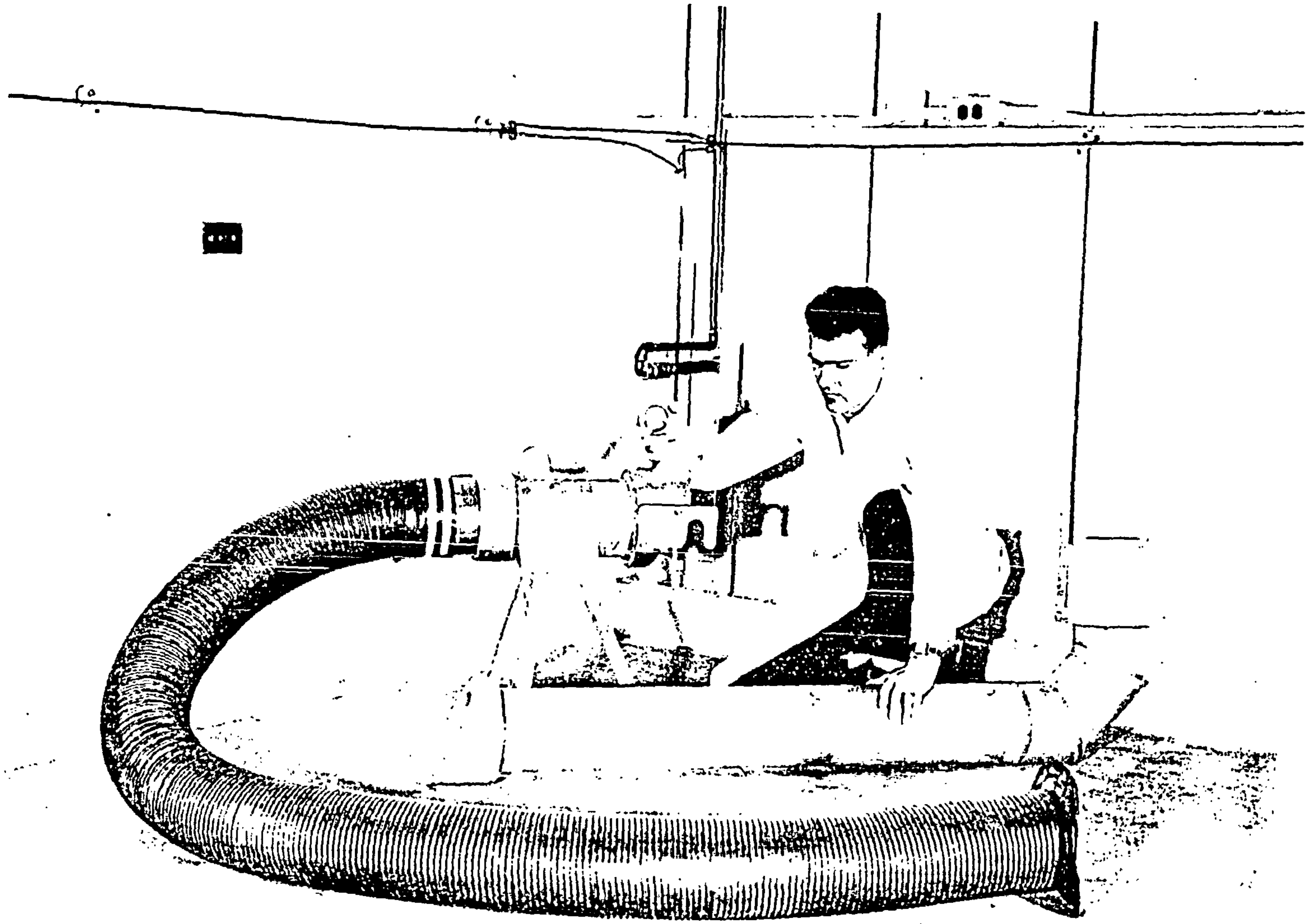
A schematic flow diagram of the cooling system is shown on page 25. Water will be pumped through the core at 700 gpm or 1200 gpm and into an aluminum plenum chamber attached to the underside of the grid plate. Grid plate holes not containing fuel elements, reflector elements, or irradiation capsules 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 14 ft above the pool floor before penetrating the wall. The coupling between the aluminum tube and the flexible section is shown on page 26.

After leaving the pool proper, the 6 in. line leads underground to a 2,000 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 pumps and the heat exchanger. At power levels between 100 kw and 4000 kw, the reactor can operate continuously with the bulk of the pool water at about $90^\circ F$. For 4 Mw operation, the temperature rise through the core will be approximately $22.7^\circ F$.

The secondary or cooling side of the heat exchanger forms a closed loop whereby the coolant water is pumped through the heat exchanger at 800 gpm, 1600 gpm, or 2400 gpm and returned to the cooling tower.



REACTOR POOL COOLING, CLEANING & PURIFICATION SYSTEM



FLEXIBLE TUBE AND COUPLING FOR FORCED COOLING

A large flapper valve is installed in the bottom of the plenum chamber. This will be operated normally 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. Upon receipt of a low flow signal, this valve is automatically opened.

The cooling system safety features include annunciation and/or readings at the control console or a graphic panel located in the immediate vicinity of the console. The console will indicate primary coolant flow, core effluent temperature, heat exchanger effluent temperature, core differential temperature, and excess coolant activity. The graphic panel will indicate core influent temperature, excess coolant activity, and flow for the primary system; and flow, heat exchanger influent and effluent temperature, wet and dry bulb cooling tower air temperature, cooling tower air flow, basin water level and temperature for the secondary system.

B. Supporting Facilities

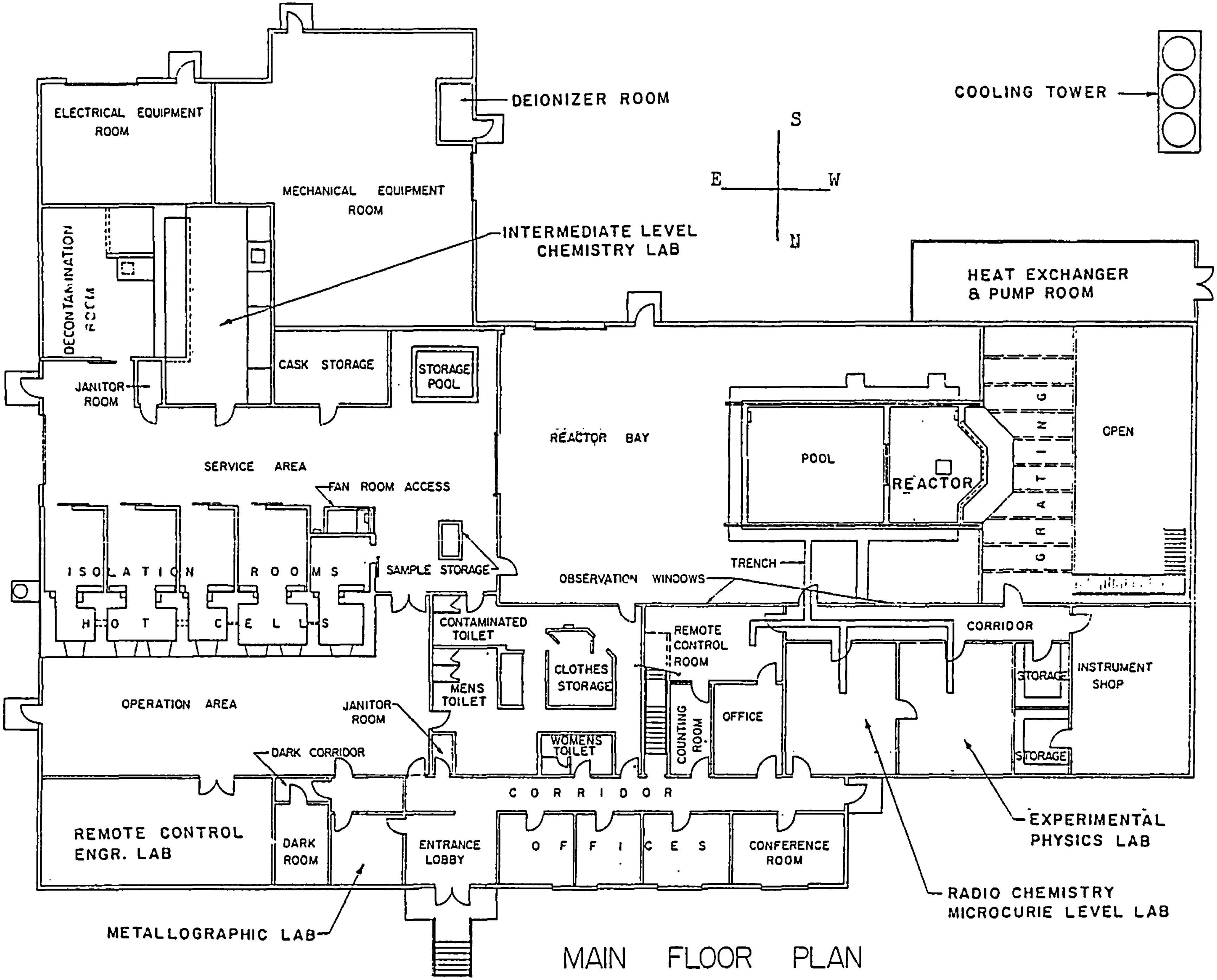
1. Reactor Building

The reactor pool is housed in a large bay 48 ft wide, 120 ft long 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.

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 three doors open 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 on page 28. Associated with the reactor portion of the building are an instrument shop, radiochemistry laboratory, remote control room and an experimental physics laboratory. 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.



MAIN FLOOR PLAN

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, intermediate level 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.

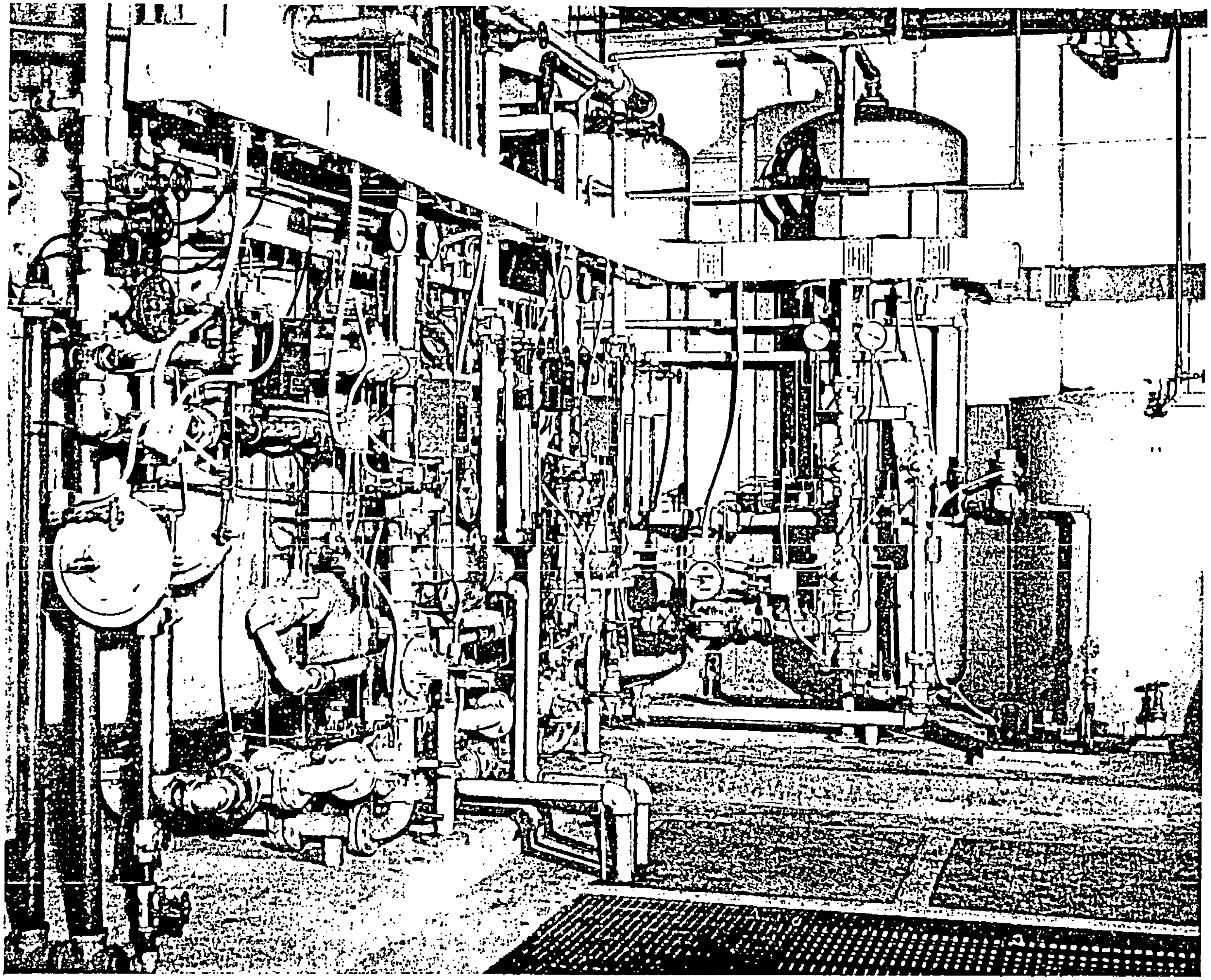
2. Pool Water Supply System

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 precoat filter, then through a water softener tank to remove the calcium and magnesium ions, and then through a mixed resin bed ion exchanger (deionizer) to remove the remaining anion and cation impurities. The effluent water contains fewer minerals than distilled water.

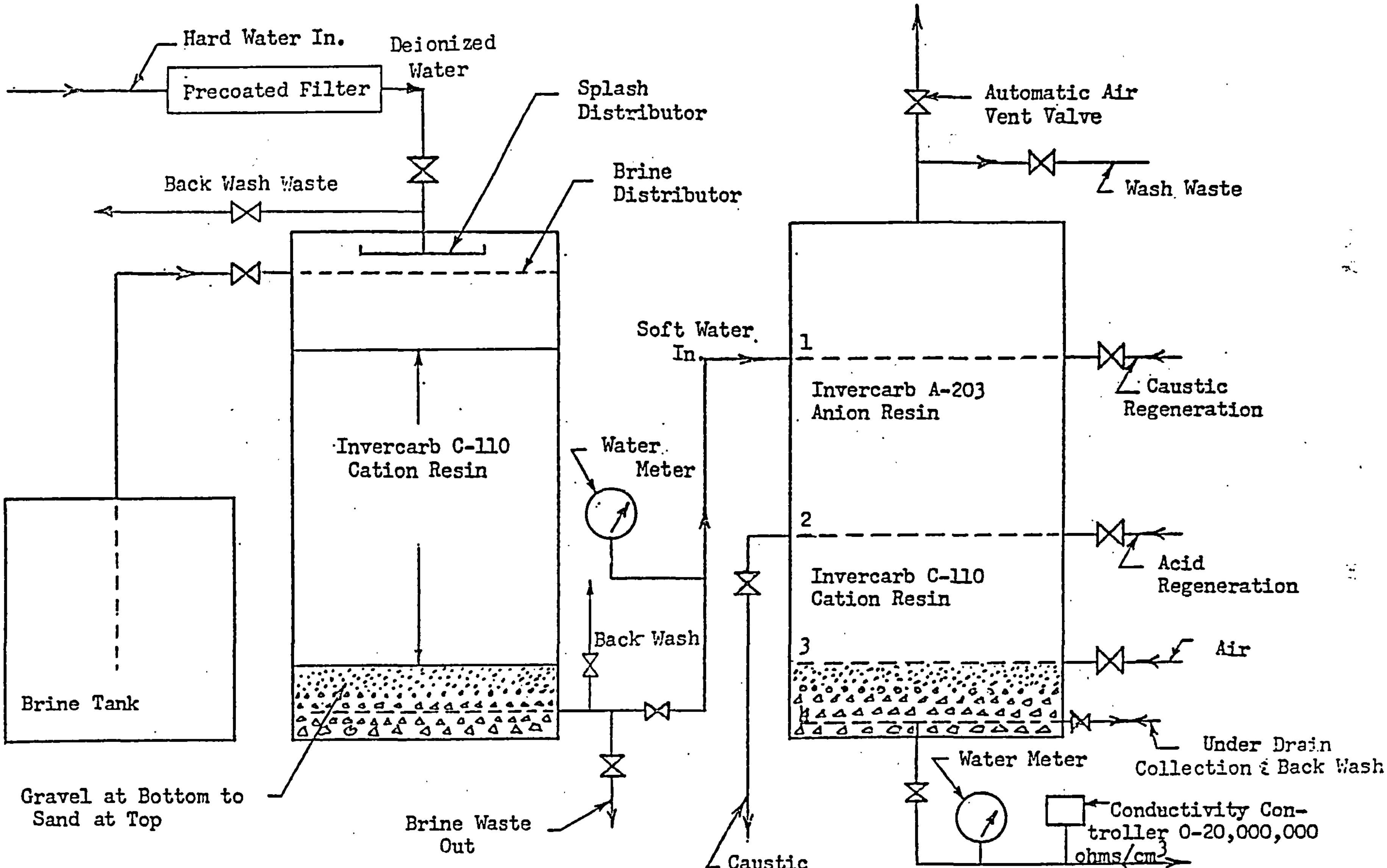
The equipment consists of two softener units and two deionizer units with the associated equipment required for operation of the system and regeneration of the resins once they have become exhausted. Normally, one softener and one deionizer will be in use while the second pair is acting as a standby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

The equipment shown on page 30 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 72 hours. A block diagram of the system is shown on page 31.

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



POOL WATER SUPPLY PURIFICATION SYSTEM



BLOCK DIAGRAM
POOL WATER SUPPLY SYSTEM

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	$Ca(H\ CO_3)^2$	71.4
Magnesium Bicarbonate	$Mg(H\ CO_3)^2$	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 and Cleaning 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.

As the figure on page 25 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 50 gpm with a pressure loss of not more than 15 psi. There is also a 15 gpm pump in parallel with the 50 gpm pump for the purpose of draining the pool. This purification rate 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 and cuno filter, located in the line prior to the pump and deionizer respectively, 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.

In the process of circulating water through the deionizers, water can be pulled from a reactor pool sump or a skimmer, individually or simultaneously. Both sections of the pool contain their individual sump and skimmer. The skimmer is a 6 x 12 x 48 in. stainless steel box, open at the top and located at the surface of the water. It can be adjusted vertically so as to allow the surface water to flow over the edge and into the box. Therefore, the foreign particles on the surface of the water can be continuously removed in conjunction with purification. A portable underwater cleaner is available for cleaning the pool sides and floor.

The efficiency of the ion exchange column in use is determined by means of conductivity cells and sample-cocks located before and after the columns. The cartridges which contain the resins are removable and will be replaced when exhausted. No attempt will be made to regenerate the resins. Used cartridges will be handled as radioactive waste.

The ion exchange assemblies shown on page 34 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.

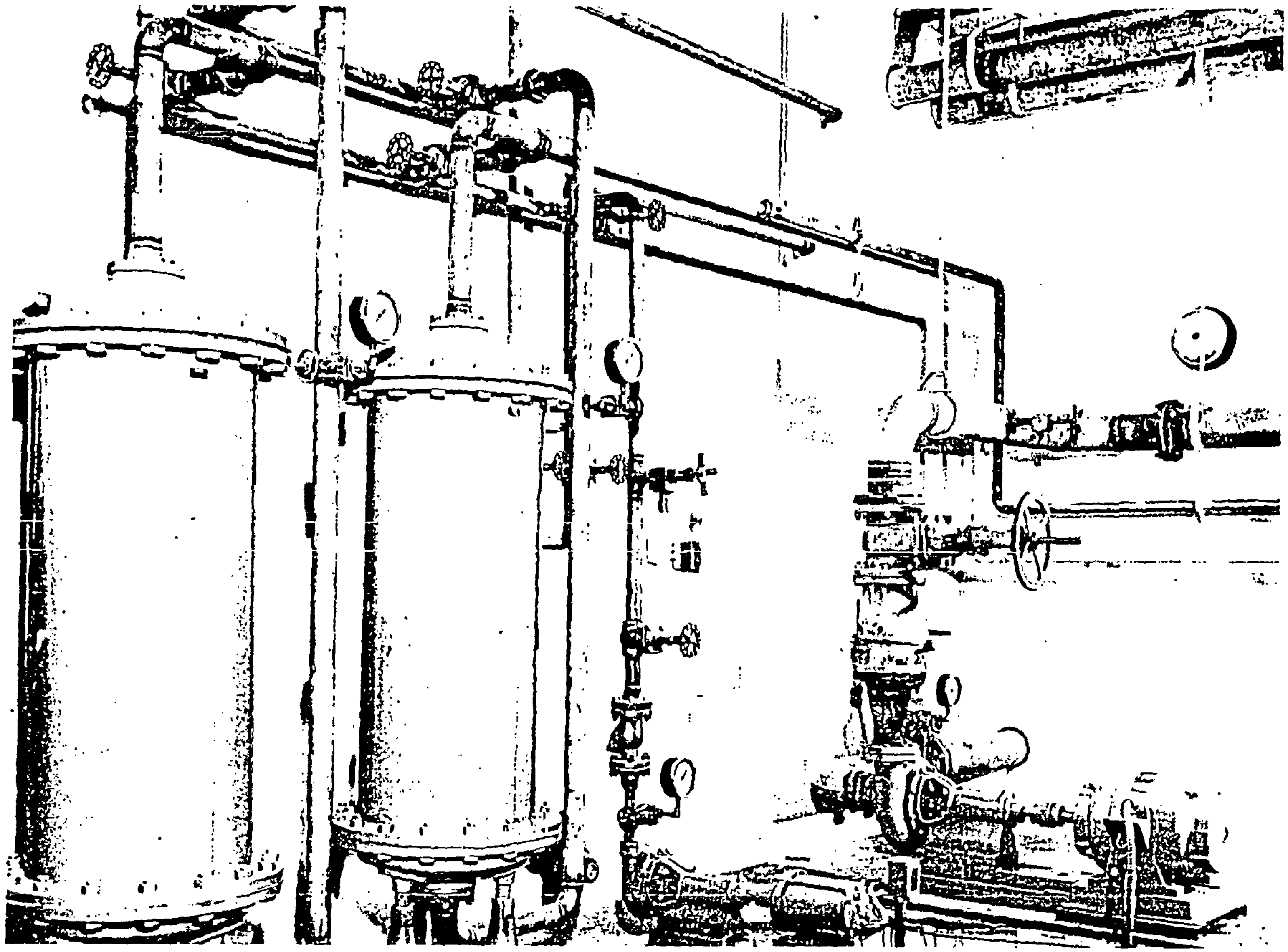
4. Beam Tubes

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.

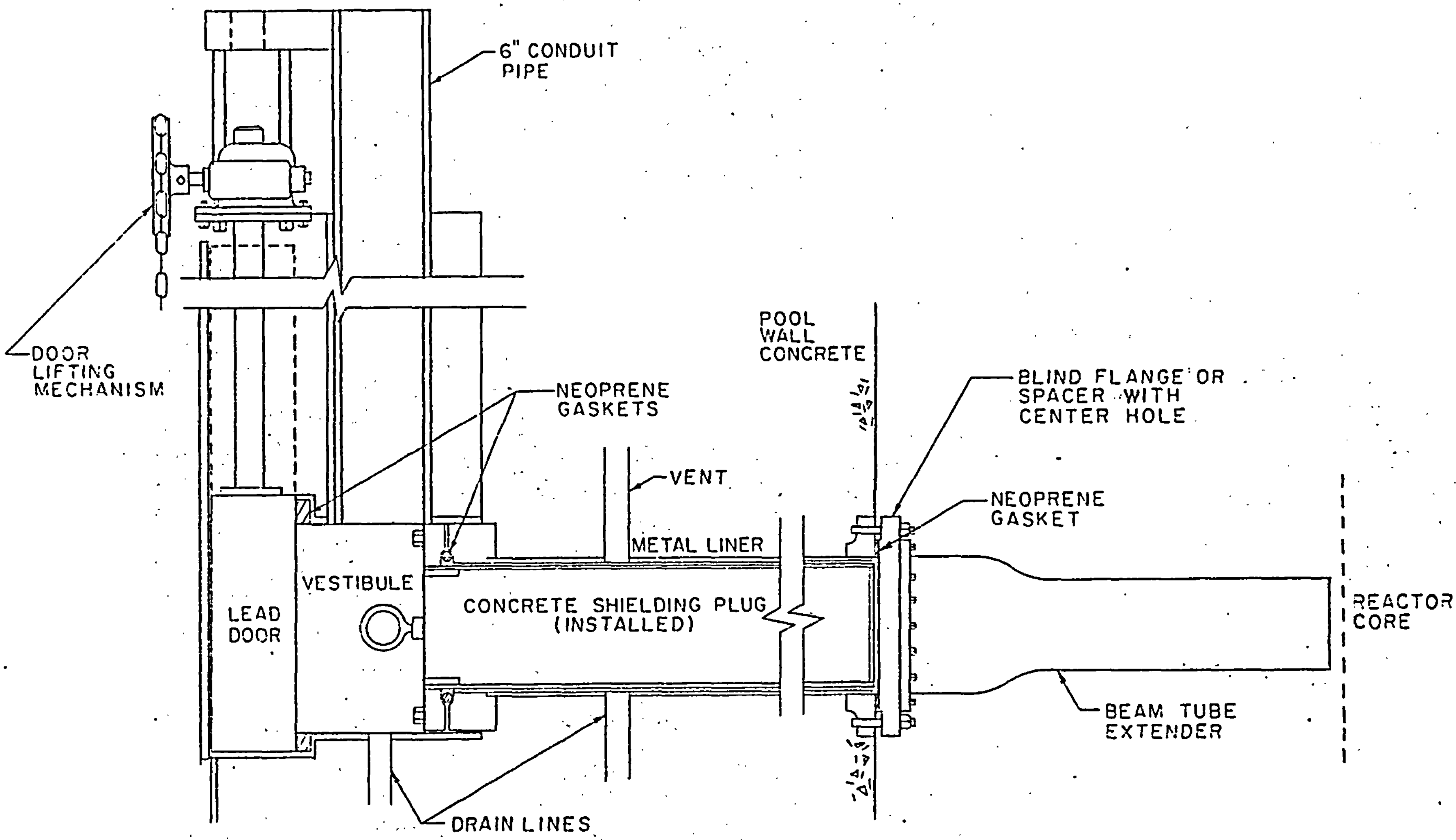
The drawing on page 35 is a cross sectional view of a typical beam tube. The first section, constructed of boral (aluminum-boron-carbide-aluminum sandwich) forms a liner for the 8.0 in. I.D. hole through the concrete wall and extends from the pool wall outer face to its inner face. The second section, constructed of aluminum, extends from 8.0 in. I.D. at the inner face of the wall, tapers to 6.06 in. I.D. over its entire length 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. The flange plate or spacer which connects the extender to the pool wall is 8 - 9/16 in. I.D. and 13 - 1/2 in. O.D. The photograph on page 36 was taken during construction, showing the beam tubes in place.

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

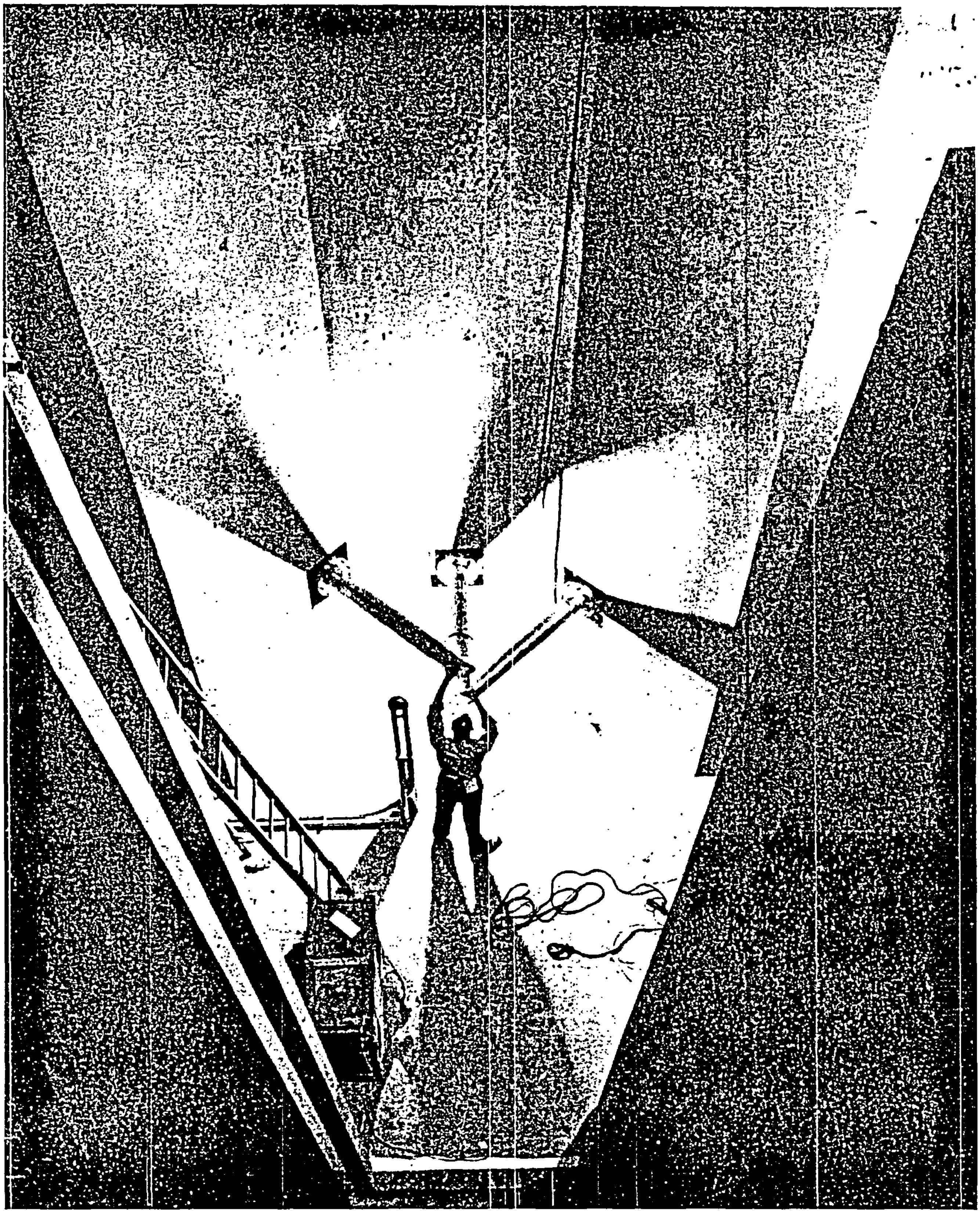
The first section of the beam tube may be sealed by a water tight blind flange, in which case the pool must be drained in order to install a beam tube extender. Without the extender in place, the neutron flux at the inner face of the beam tube is small and the hole will be useful for irradiations at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



PUMP ROOM AND ION EXCHANGER ASSEMBLIES



TYPICAL BEAM TUBE



BEAM TUBE END OF POOL.

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 than 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. At 4 Mw and considering all tubes dry, the center line thermal neutron flux (n/cm^2-sec) for a 4 by 5 loading in all three tubes is as follows:

	At Core	30 in. From Core
Beam tube #1	9×10^{12}	$.3 \times 10^{12}$
Beam tube #2	14×10^{12}	$.8 \times 10^{12}$
Beam tube #3	13×10^{12}	$.4 \times 10^{12}$

The flux at the exit apertures is approximately 4×10^9 n/cm^2-sec.

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 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.

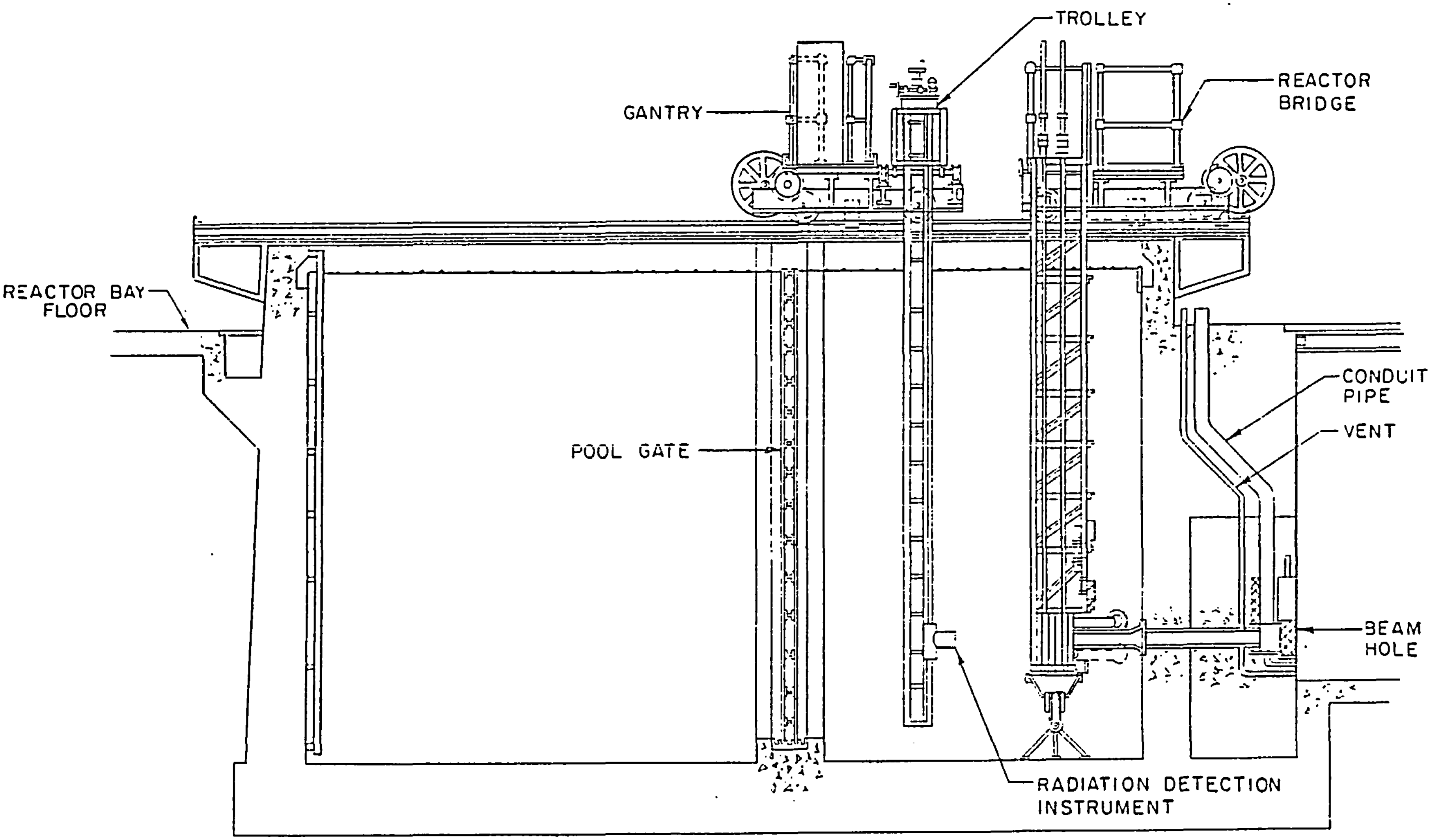
Concrete shielding plugs are available for each hole when not in use. They are contained in a 1/4 in. thick aluminum outer casing.

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 on page 35. 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.

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



REACTOR POOL FACILITIES — VERTICAL SECTION 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. A schematic diagram of the system is shown on page 40.

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 on page 36.

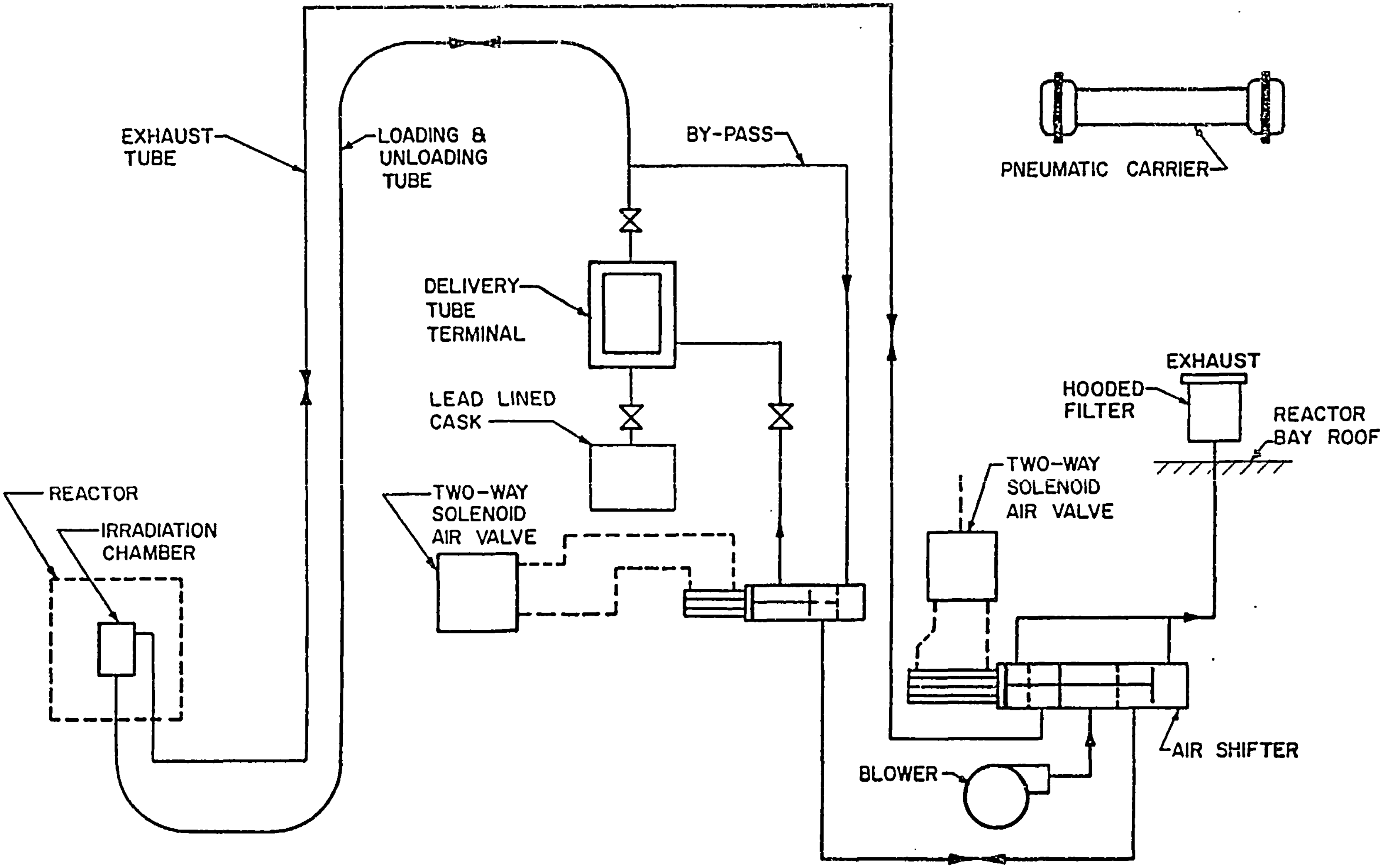
The blower has a capacity of 80 cfm of air at a pressure of 2 psi. This air is used for movement of the pneumatic carrier to and from the irradiation chamber at a speed of about 30 ft/sec. It is used also for cooling the carrier once it is in the irradiation position.

The delivery tube terminal is located on the east wall of the beam room together with the control panel. This is shown on page 41.

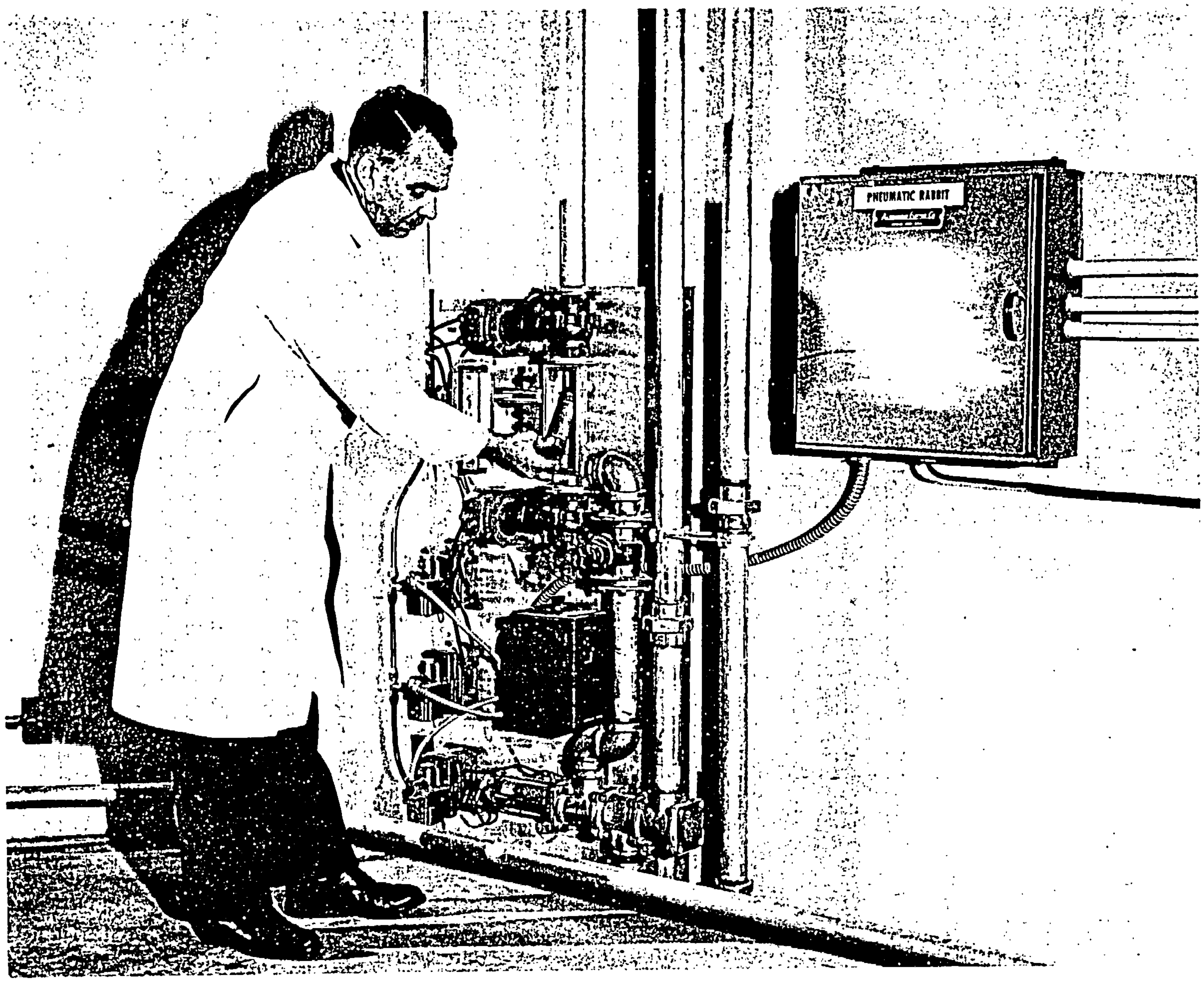
The pneumatic carriers are similar in design to the department store type. They are expendable and suitable for use in temperatures up to 137°F. The main body and heads are constructed of Micarta, and the two air discs of Fabreeka. The heads are threaded to the body portion of the carrier and it is necessary 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, fabricated of 2S aluminum, is attached to the bottom of the cart and extends down to the reactor core level. An elevator traverses the length of the tower, thus permitting objects to be positioned at any desired vertical distance from the reactor (see page 38.) Four degrees of freedom, described in Table V, are available for positioning objects relative to the reactor core.



PNEUMATIC CONVEYOR SCHEMATIC



PNEUMATIC RABBIT TERMINAL AND CONTROL PANEL

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, page 46.

The capacity of the plant was based on an estimated flow of 300 gal/day, of which about 10% might actually require treatment.

The figure on page 47 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 change room showers, reactor area, the personnel decontamination sink in the change room, and skimming of the reactor and storage pool. 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 processed through the evaporator.

Because of the design of the system and the use for which it is intended, off-gases are not considered to be a problem under normal usage. The system does not receive high level wastes. These will be segregated as they are generated and will be handled in small batches. The waste treatment system normally will receive only wastes with essentially no dissolved active gases. Any operation which would generate significant fission gases would be called to the attention of the health physics group and the entire operation would be closely monitored, including the disposal of wastes.

The contents of any tank which is found to contain an unusually large amount of activity would be analyzed radiochemically. It would, therefore, be impossible for a large amount of dissolved fission gas or any other activity which had accidentally been allowed to enter the collection system to reach the evaporator undetected.

All pressure relief and vent lines from the evaporator and vacuum-receiver tanks pass through an absolute filter before being discharged to the atmosphere. The performance of this filter in removing particulates is monitored by sampling the effluent air stream from the filter during the evaporation process.

If the analysis of the waste solution before treatment has revealed the presence of significant amounts of fission gases which might be driven off during evaporation and would not be removed by the absolute filter, the monitoring would be extended to outside the waste disposal building. The evaporation process would not be undertaken until meteorological conditions were such that the operation could be conducted safely and within the limits set by Part 20, CFR.

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 distillate 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 an ejector operated by compressed air. This causes the liquid in the evaporator to boil at a relatively low temperature (approximately 80°C). The 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 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. <u>Criterion for Release of Liquid Wastes to the Environment</u>. Radioactive liquid wastes, when sampled and found safe for release, are released to the area stream system. The figure on page 48 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.

The illustration on page 49 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.

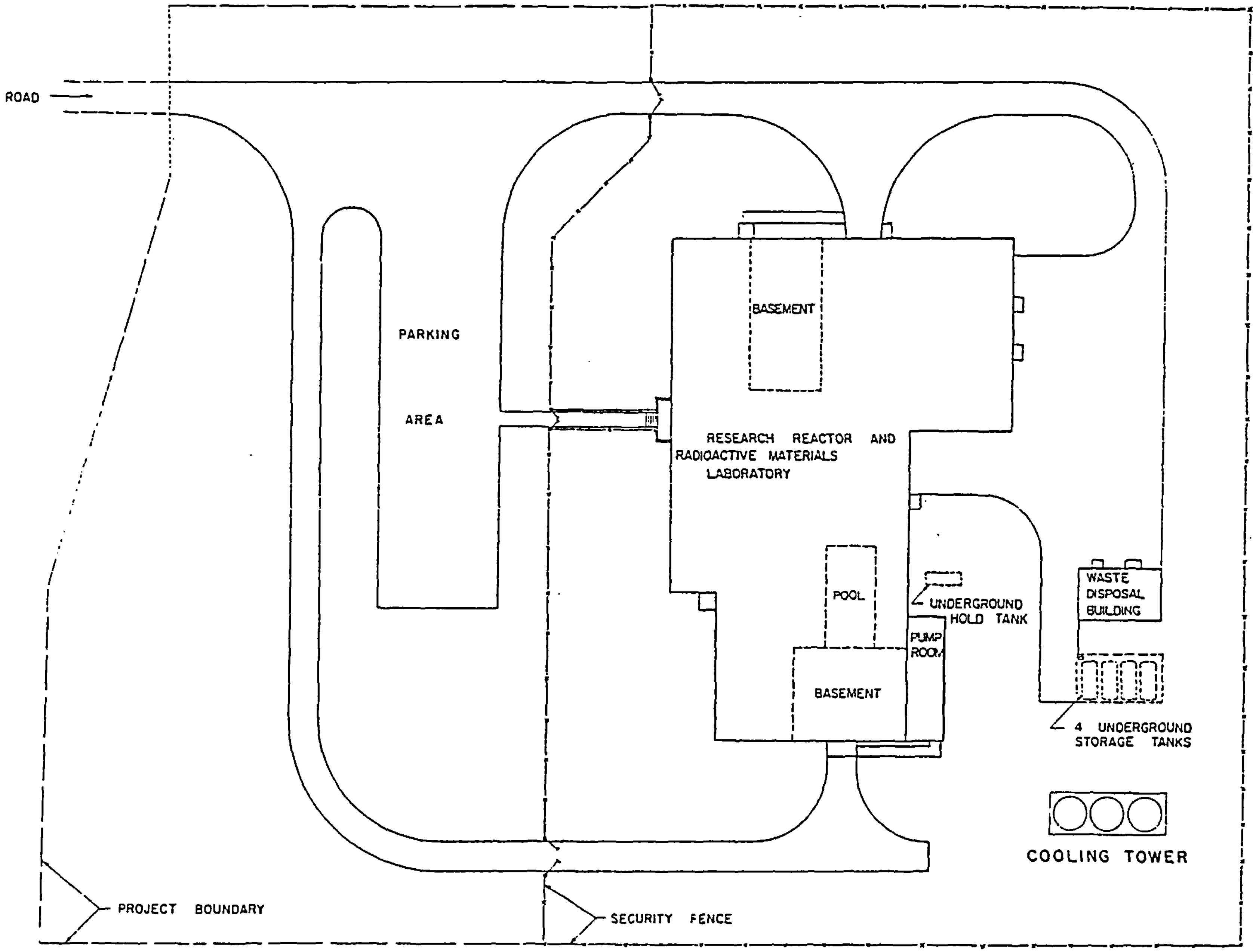
Wastes released from the liquid waste treatment plant are carried by underground pipe to Meeker Run. Waste water released from the reactor pool flows via an underground pipe to a point where the contours of the terrain give positive flow to Reactor Run and thence to Meeker Run. The end of the pipe terminates over crushed stone "rip-rap" for erosion control.

The flow of spring water following Route "A", shown on page 48, is a little over 30 gpm, so wastes following Route "B" 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.

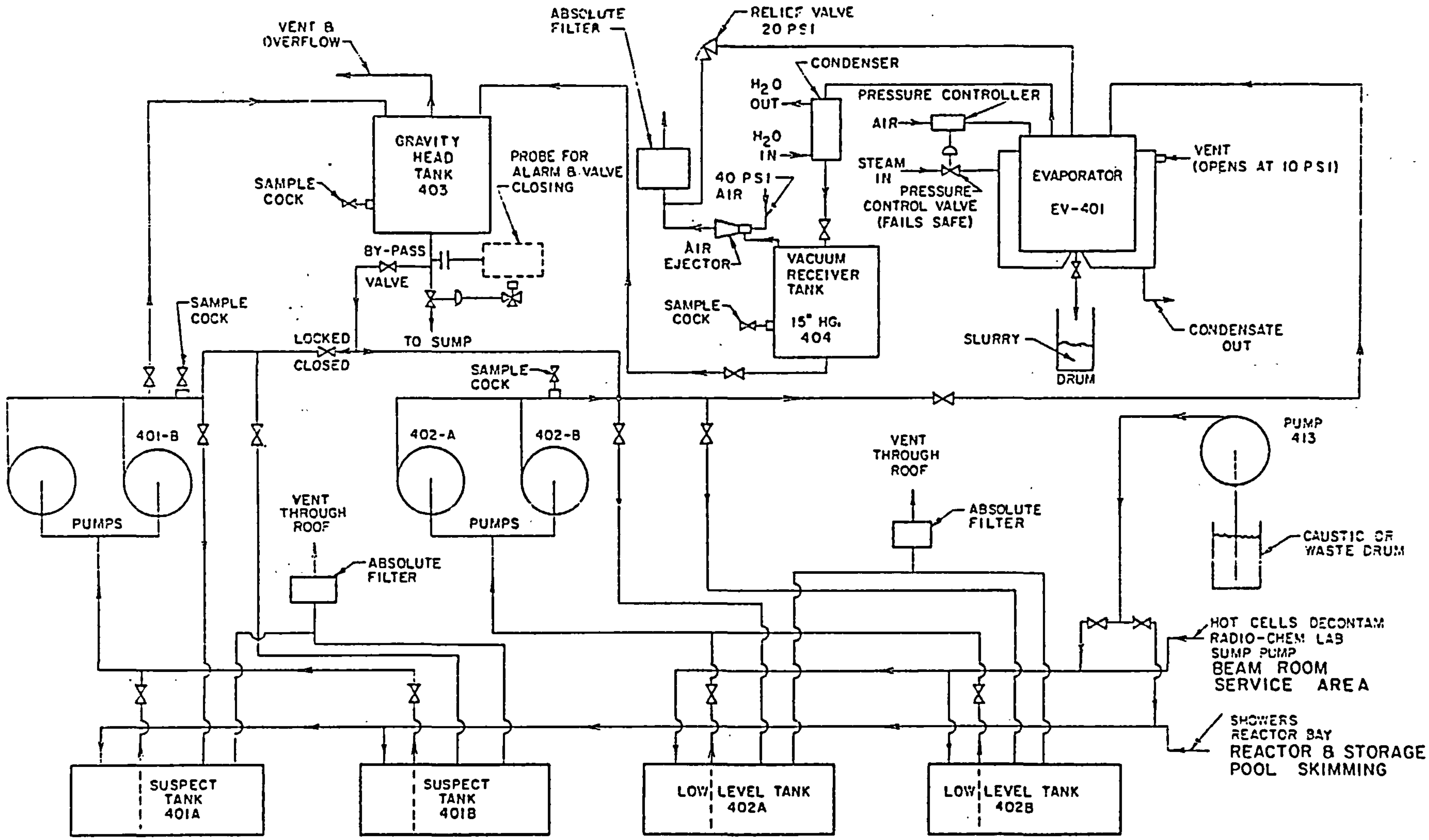
Pennsylvania State Industrial Wastes Permit #1907 allows the release of liquid wastes at or below the maximum permissible concentration for continuous use in unrestricted areas for the most hazardous radioisotope in the mixture.

Laboratory wastes having a maximum permissible concentration of less than 10^{-4} microcuries per milliliter for continuous use in unrestricted areas, are collected separately, absorbed in vermiculite, and disposed of as solid wastes. This means that the storage tank solutions may be disposed of directly to Meeker Run at a radioactive concentration of 10^{-4} uc/ml or less.

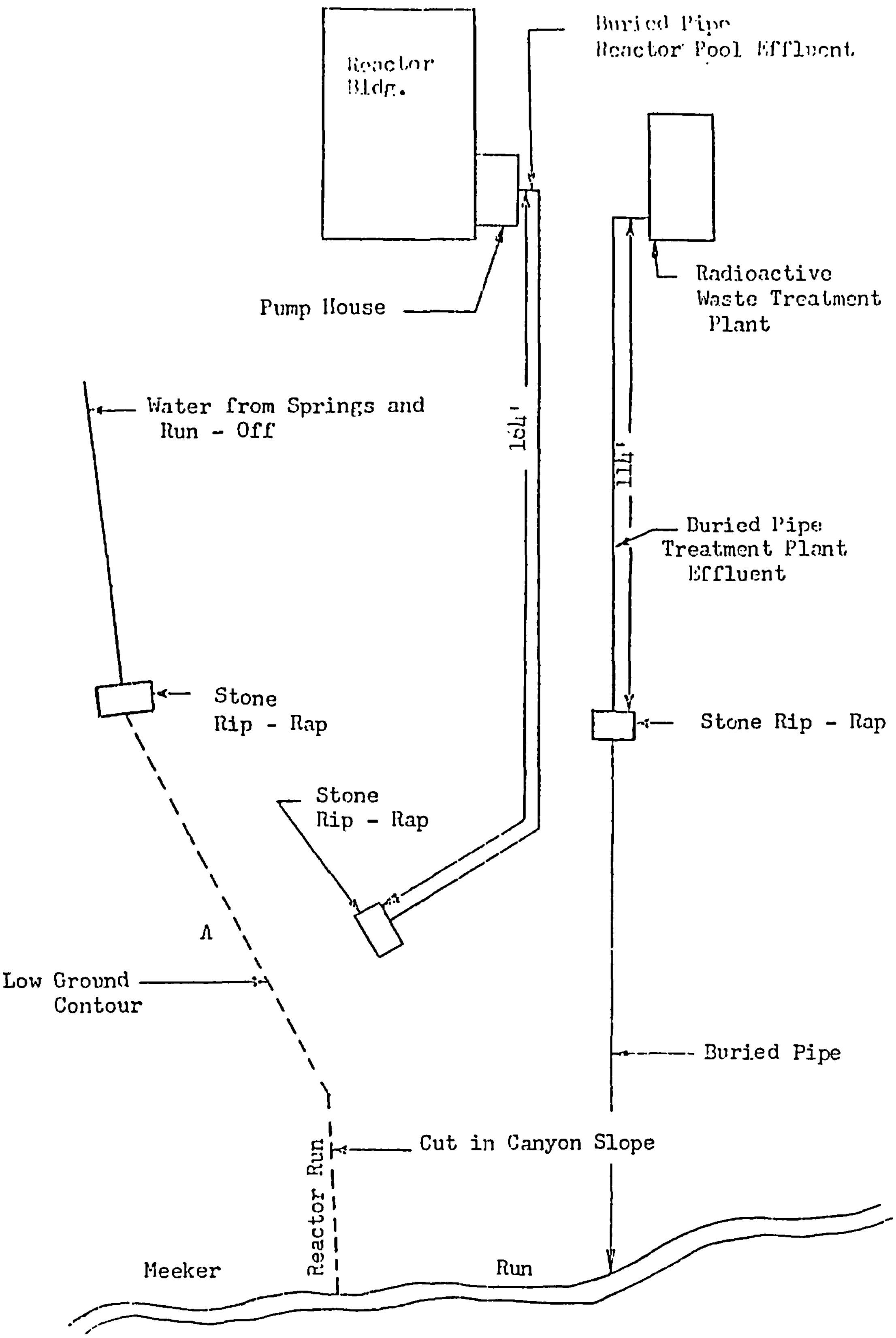
The following table is used to determine the average daily quantity of activity that may be discharged over any 365-day period.


ROAD
PARKING
AREA
BASEMENT
RESEARCH REACTOR AND
RADIOACTIVE MATERIALS
LABORATORY
POOL
UNDERGROUND
HOLD TANK
PUMP
ROOM
BASEMENT
WASTE
DISPOSAL
BUILDING
4 UNDERGROUND
STORAGE TANKS
COOLING TOWER
PROJECT BOUNDARY
SECURITY FENCE

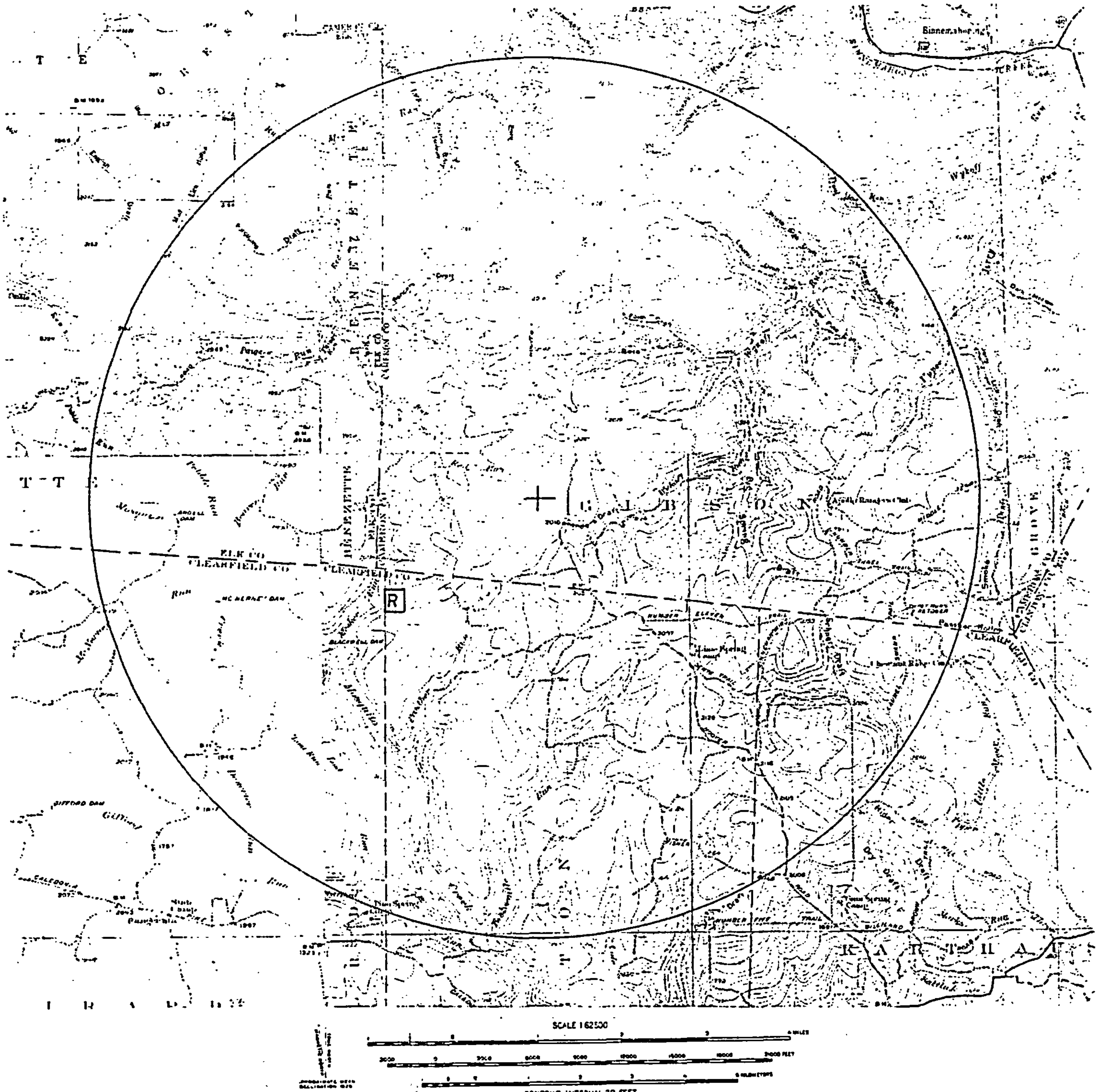
SITE LAYOUT



WASTE TREATMENT PLANT



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



(Latitude N41° 13' 21" and Longitude W78° 13' 48")

THE QUEHANNA SITE

R is Reactor

MPC for Continuous Use in Unrestricted Areas	Average Quantity of Radioactive Material That May be Discharged
10^{-8} uc/ml	0.4 uc/day
10^{-7} uc/ml	4.0 uc/day
10^{-6} uc/ml	40.0 uc/day
10^{-5} uc/ml	120.0 uc/day
10^{-4} uc/ml and greater	120.0 uc/day

The maximum quantity discharged in any one day is never more than four (4) times the quantity listed in the above table.

Investigations indicate that the average flow in Meeker Run, directly opposite the facility, is about 1000 gpm. As indicated above, no advantage is taken of the dilution afforded by this stream.

8. Ventilation

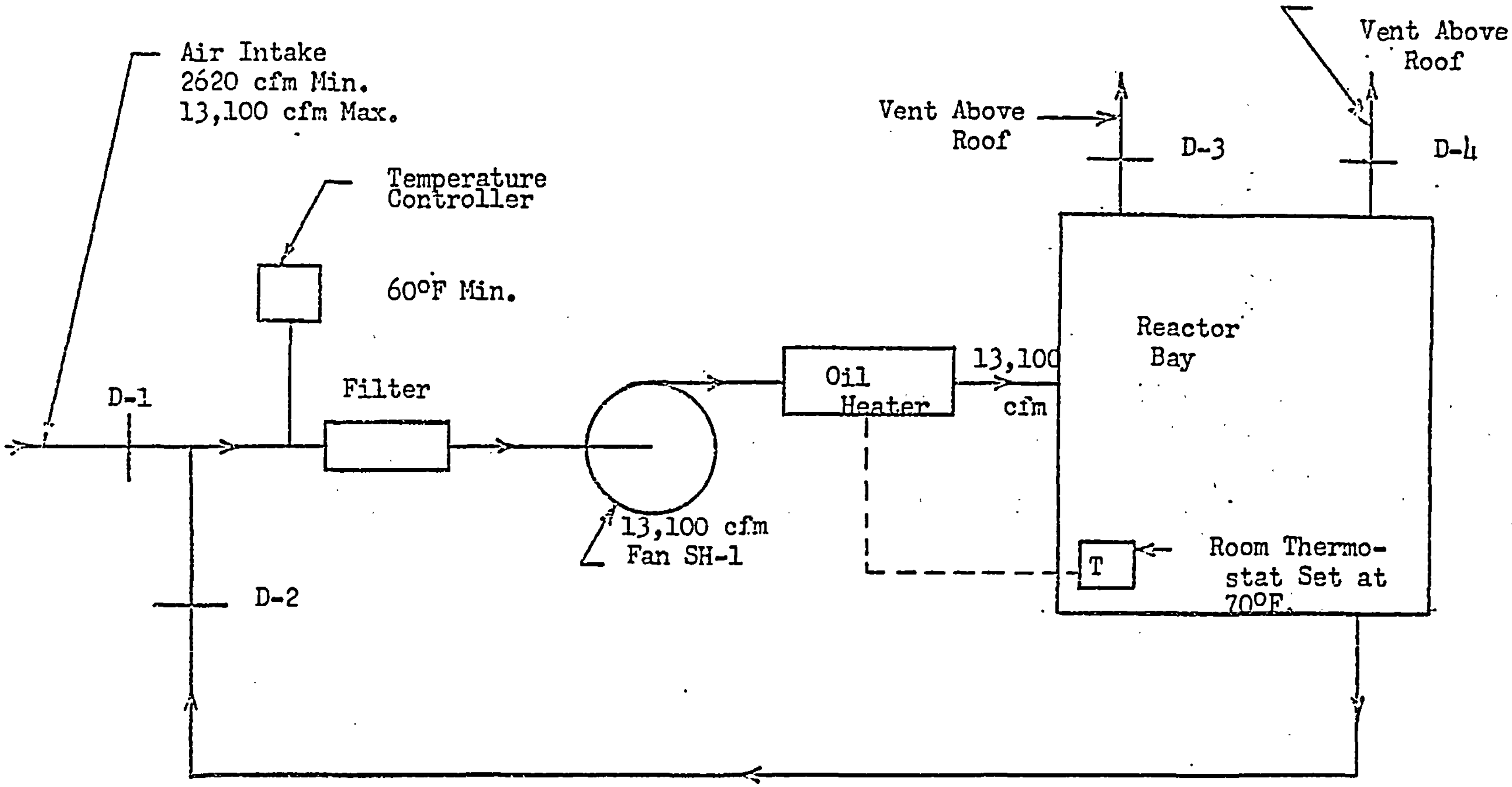
Heat and ventilation for the reactor bay are supplied by a single overhead space heater. This oil fired unit provides a minimum of 20% outside air and at least two air changes per hour.

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 three 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.

Page 51 shows a schematic diagram of the system. The supply fan is suspended from the ceiling of the reactor bay and draws its air from 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.



REACTOR BAY SUPPLY & EXHAUST

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 would 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

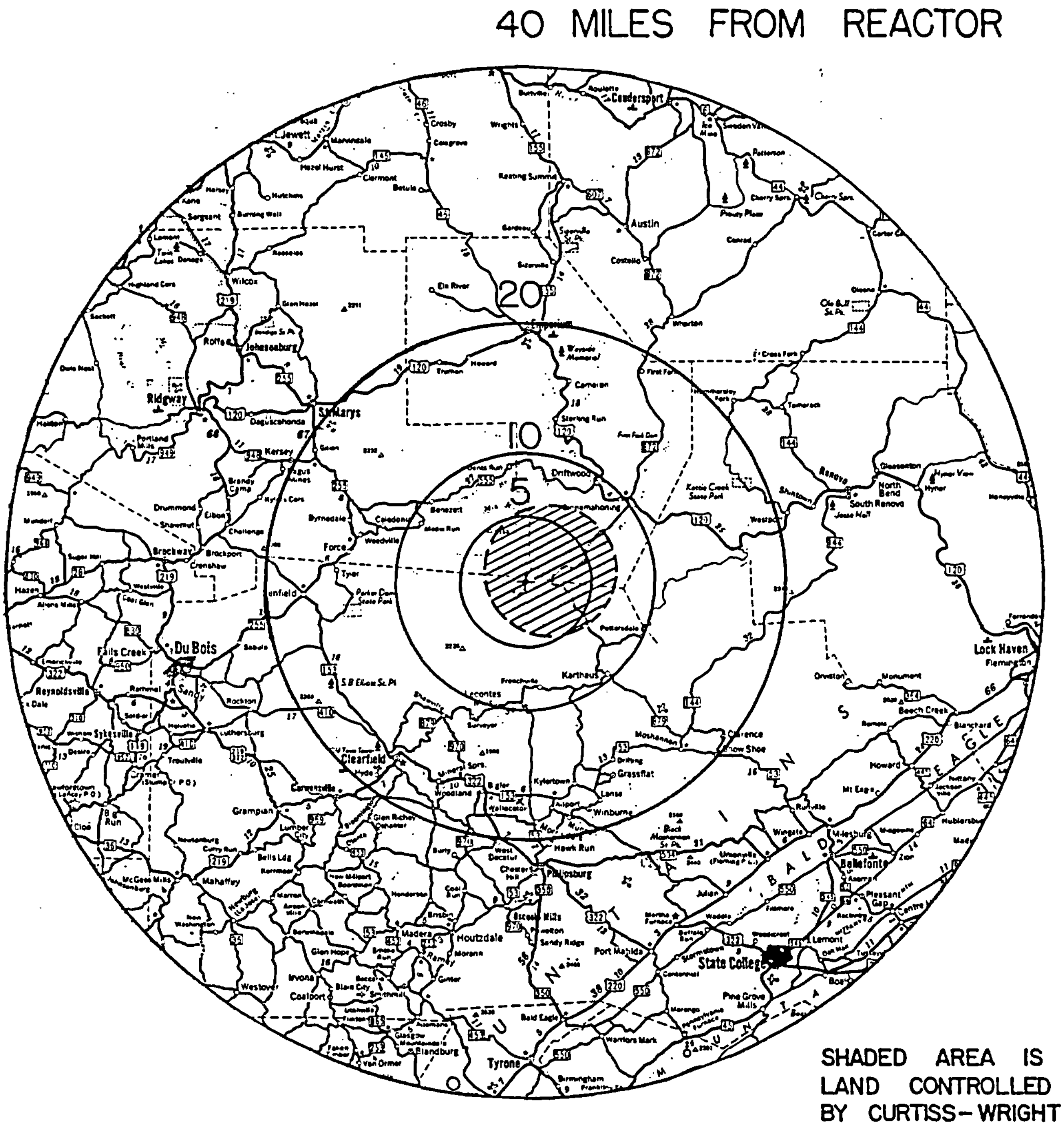
The Radioactive Materials Laboratory and Research Reactor is located on an 80 square mile tract of land which Curtiss-Wright has bought or leased for 99 years in order to build a research and development center for the Corporation. This tract, in the form of a 16-sided figure approximating a circle of 10 mi. in diameter, lies in North Central Pennsylvania (see page 53). It encompasses portions of Elk, Cameron, and Clearfield Counties.

As the figure on page 49 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 is one of the larger streams which merges with Mosquito Creek before it leaves the Curtiss-Wright site.

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 in 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 water is considered further in Section IV-B.

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



LOCATION OF REACTOR RELATIVE TO ENVIRONS

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 a 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.

A precise description of the location of the actual "Reactor site" itself (comprising an area of about 2.5 acres) is at present being obtained by accurate survey. As soon as this is available, it will be forwarded to the Commission for incorporation into the Curtiss-Wright Reactor License and as an addendum to this report.

TABLE VI

Towns Nearest Reactor Site*

	Population		Population
Bald Hill	250	Lecontes Mills	600
Benezette	200	Morrisdale	655
Bloomington	200	Moshannon	500
Blue Ball	900	Munson	450
Brockport	415	Oak Grove	300
Byrnedale	500	Oshanter	300
Caledonia	300	Orviston	400
Chester Hill	954	Penfield	600
Clarence	530	Philipsburg	3,988
Clearfield	9,357	Quail	600
Crenshaw	550	Renovo	3,751
Curwensville	3,332	Rockton	250
Driftwood	289	Saint Marys	7,846
Drury Run	300	Sinnemahoning	450
Emporium	3,646	Snow Shoe	670
Force	506	South Renovo	862
Frenchville	558	Sterling Run	225
Glen Richey	425	Tyler	300
Hawk Run	651	Wallaceton	440
Hyde	750	Weedville	600
Karthaus	[illegible]	Westport	201
Keewaydin	200	Woodland	650
Kylertown	500	Winburne	700
Lanse	350	Pine Glen	182

*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

Four safety-shim rods and one regulating rod will be used in controlling the reactor at the 4 Mw level. Each of the four safety rods, made of boron carbide and cadmium, will control a reactivity worth that is not expected to exceed 3.5% (highest value measured is 3.3% for a graphite reflected core). 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 0.025% per second (when the total rod worth is 3.5%). The safety rods may be withdrawn individually, or two may be withdrawn simultaneously as a gang. The two gang switches, each controlling two rods, are interlocked to prevent simultaneous operation. The maximum rate of insertion of reactivity, then, should never exceed 0.050% per second.

Several stainless steel regulating rods, as described on page 16, are available for use in the reactor. In no case will a regulating rod be used if it is worth more than 0.70% Δ k/k. The regulating rod is mechanically coupled to its drive mechanism. The maximum withdrawal rate, initially designed for 25 in. per minute, will be changed to approximately 12 in. per minute in order to reduce the reactivity change introduced by an "impulse" from the automatic control system. Experience has indicated that more precise regulation and a reduction in wear of the mechanism can be realized by use of the reduced rate. In its most effective position, the maximum rate of change of reactivity of the regulating rod will never exceed 0.010%/sec.

The position of all rods is continuously indicated to within ± 0.05 in. at the reactor control console by an electrical transmitting system. Microswitches, 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 regulating rod 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.

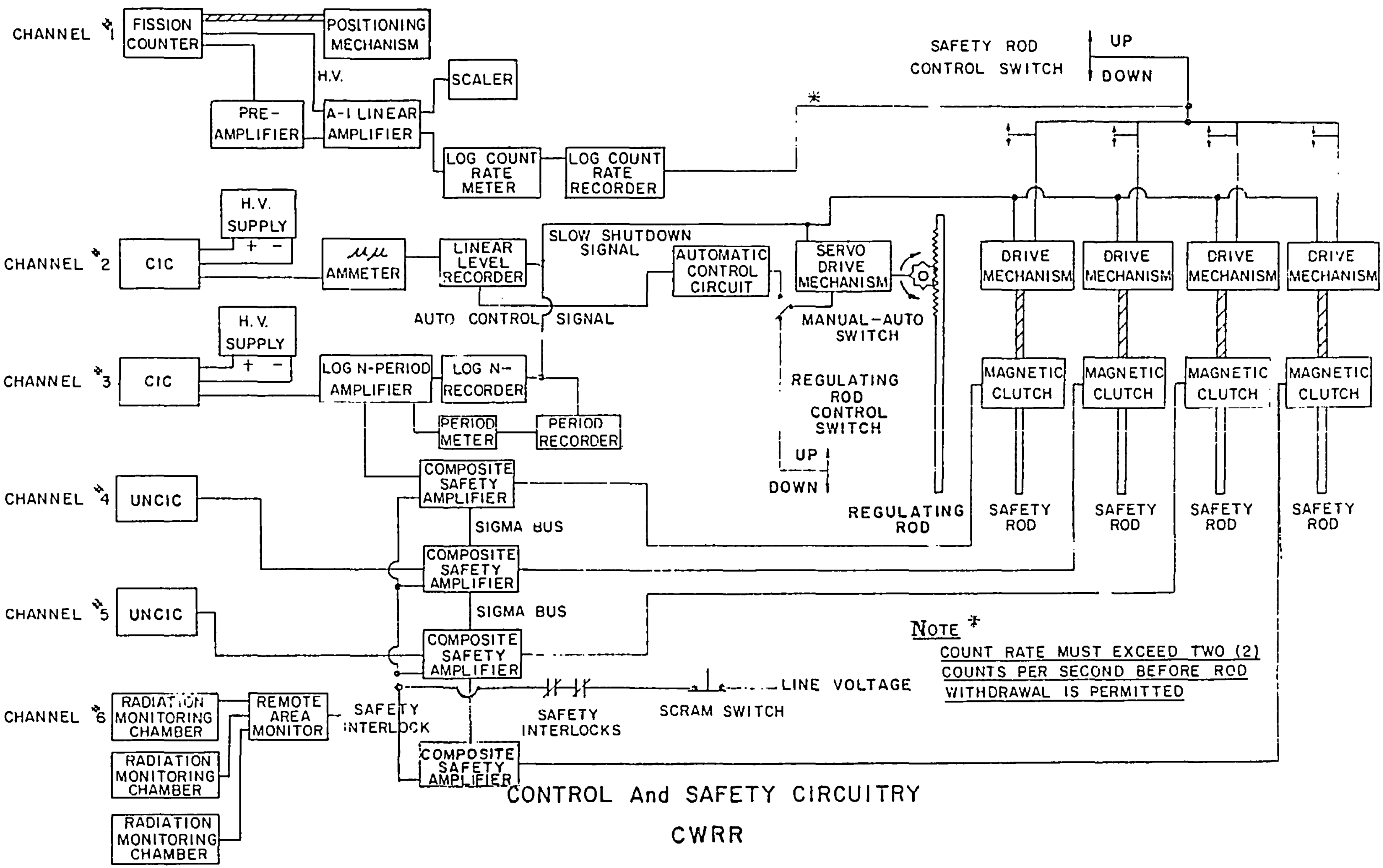
Four gamma compensated ionization chambers and a U-235 lined fission counter serve as the flux level detecting devices. A block diagram of these units and their associated circuits is shown on page 58.

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 in 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 a drastic electronic reduction of the current to the safety rod coupling magnets, and requires a minimum of time to effect this reduction (response time of approximately 30 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 of three ionization chambers - two sensing linear level and connected directly to their respective safety amplifiers, and the remaining one connected directly to the Log N system which furnishes a period signal to the safety amplifiers. The slow scram requires an appreciably longer time (approximately 100 milli-seconds) 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 VII, the slow scram is initiated by movement of the reactor bridge during operation and by manual operation of the scram pushbutton.

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. Two coolant activity monitors will be used to detect the presence of fission product activity in the coolant stream. The first monitor utilizes a BF_3 counter to detect delayed neutrons in the pump room downstream of the holdup tank. The second monitor employs two ion exchange beds. The first bed removes positive ions from a sampling line connected to the pump effluent stream. The second bed collects the negative ions, and the first layers of these resins are monitored by a gamma scintillation probe. From the second bed the sampling line is returned to the influent side of the pump. If a preset activity level is exceeded in either of the two monitors a slow shutdown will be initiated.

Flow rate and temperature monitors are described in Section IA 6.



CONTROL And SAFETY CIRCUITRY

CWRR

TABLE VII

Reactor Safety System

Situation	Detector	Unit Initiating Action	Resulting Action	Annunciation
140% Neutron Flux	Compensated Ion Chamber	Safety Amplifier	Fast Scram	Yes
Period-Fast Scram	Compensated Ion 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	Compensated Ion Chamber	Log N & Linear Recorder	Slow Shutdown	Yes
Period-Slow Shutdown	Compensated Ion Chamber	Period Recorder	Slow Shutdown*	Yes
Coolant Flow Interrupted	Pressure Transducer	Electronic Relay	Slow Shutdown	Yes
Core Differential Temp.	Resistance Thermometer	Electronic Relay	Slow Shutdown	Yes
High Radiation Level	Ionization Chamber	Radiation Monitor	Slow Shutdown and Evacuation Alarm*	Yes
Coolant Activity Excessive	BF_3 Counter and Sodium Iodide Crystal	Delayed Neutron and Ion Exchange Fission Product Monitors	Slow Shutdown*	Yes
Core Effluent Temperature	Resistance Thermometer	Electronic Relay	None	Yes
Safety Amplifier Indication	Safety Amplifier	------------------	None	Yes
Deionizer Depleted	Conductivity Cell	Solu Bridge Controller	None	Yes
Safety Circuit Bypassed	Reactor Engineer	Key Switch	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 Chamber	Log Count Rate Recorder	Prevents Withdrawal* of Safety Rods	No

* Indicates that safety circuit may be defeated (bypassed).

The performance of the deionizer will be monitored continuously by conductivity cells measuring the specific resistance of the pool water. This will provide information so that the resins can be replaced before becoming seriously depleted. An area monitor is installed in the pumproom to warn personnel of excessively high radiation levels before they enter the room.

E. 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 of experimental work will be divorced from operational considerations. The experimental program will be reviewed by the Project Engineering Staff and recommendations will be referred to the Chief of the Research Reactor Division, who will then decide what experiments will be conducted and with what priority. All projects approved by the Project Engineering Staff will be referred to the Health Physics officer for his approval and may be further referred at his discretion to the Curtiss-Wright Nuclear Hazards Committee.

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 compromise safety.

b. Supervise all reactor operating personnel.

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

d. Insure records and logs of reactor operation are kept up to date and in proper form, including fuel inventory.

e. Issue regular monthly reports on reactor operations.

f. 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 Nuclear Hazards Committee (CWNHC) covering any such incident and the corrective action taken.

g. Cooperate with Research Reactor Division Chief in scheduling and coordinating experimental programs and service irradiations.

h. 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.

3. Curtiss-Wright Nuclear Hazards Committee (CWNHC)

The Curtiss-Wright Nuclear Hazards 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 responsibilities of this committee shall be as follows:

a. Review all requests for proposed reactor projects which are forwarded to it. This review shall encompass only matters concerning health, safety, and legal requirements 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. Project Engineer

In order to assure that experimental equipment being designed and constructed for installation in the Curtiss-Wright Reactor conforms fully to all applicable engineering and safety requirements. Project Engineers will be assigned to review all future reactor projects.

5. Reactor Engineer (Shift Supervisor)

A reactor engineer shall be a person holding a valid AEC operator's license and having sufficient experience to intelligently perform the following duties:

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. 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.

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

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

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

g. 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.

6. 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 Reactor Operation Section Head or a Reactor Engineer during startup, shutdown, or 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 in the immediate vicinity of the reactor console. Changes in power level will be accomplished only under the direct supervision of a reactor engineer.

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 the 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 estimated 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 in Appendix IV. After completing the checklist, the form will be signed by the operator and the shift supervisor and then filed.

A complete checkout of the reactor will be required whenever the reactor has been shutdown for more than three hours. The nature of the checkout for reactor shutdown periods of less than three hours shall be at the discretion of the reactor engineer.

4. Start-Up (Cold)

A cold start-up is, in general, one in which the reactor has been shutdown long enough to reach equilibrium conditions, require a checkout, or one designated as such by the reactor engineer. Under conditions of a routine cold start-up, the following procedure will be followed unless the nature of the experiment requires a modification as prescribed by the reactor supervisor.

a. Make sure that the Reactor Checkout Procedure form has been properly completed and signed. Select the proper stamp and enter the appropriate information in the Reactor Log Book; refer to page 67.

b. Raise the safety-shim rods 6 in. and inspect the core to make sure that the rods and rod drives are operating properly. Also make sure the loading, samples, and source are as they should be.

c. Announce intention to start-up to entire building over the intercommunication system and log start-up time.

d. Raise safety-shim rods until shim range is reached, closely watching the 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 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. Record the appropriate information in the log book; see stamp on page 67.

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>

Any change in power and at hourly intervals, the operator will select the proper stamp and record the appropriate information. See stamp on page 68.

7. <u>Shutdown</u>

For routine shutdown, intention to shutdown will be announced on the PA system, and the reactor taken off of "AUTO." Log the shutdown time in the log book.

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

Start-Up Stamp

Loading No.____________________ Date____________________________

Experiment____

Purpose____

W. O. Number____

Operators____

Start-Up Time____________________ Time Reached Power_______________

Power: Linear Rec.______________________ Log N Rec.________________

Rod Positions: SR-1____________ SR-2____________ SR-3______________

SR-4____________ RR______________

Coolant: Core Effluent Temp._____________ HX Effluent Temp.__________

Core Diff. Temp.________________ Flow______________________

Samples at Core____

____

Hourly Reading and Alteration of Power Stamp

Time Changed Power__________ Hourly or Reached Power Time__________

Power: Linear Rec.__________ Log N Rec.__________

Rod Positions: SR-1__________ SR-2__________ SR-3__________

SR-4__________ RR__________

Ram Channels: 1__________ 2__________ 3__________

Coolant: Core Effluent Temp.__________ HX Effluent Temp.__________

Core Diff. Temp.__________ Flow__________

Magnet Currents: 1__________ 2__________ 3__________ 4__________

Integrated Power__________ Operator__________

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 complete the shutdown checkout procedure as described in Appendix IV unless instructed to do otherwise by the reactor engineer.

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, 277/480-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 the 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. The reactor automatically scrams whenever normal power fails.

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," "hand-crank," "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 19½ ft. of water will normally cover the center of the reactor. Data from the CWRR at one Mw operation gives a dose rate due to the penetration of gamma rays from the core of 2.5 mr/hr at the water surface, and a negligible close rate at the console. The neutron flux was negligible.

Continual recirculation of the pool water through a mixed-bed ion exchanger at 50 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 continuously monitored and activity buildup will be detected long before it becomes an external hazard. The continuous 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 occasionally 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 pump room will be continuously indicated 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 1200 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 be opened automatically to allow free convective cooling of the core, thus allowing afterheat to be removed without boiling.

The 2000 gal. hold-up tank will provide sufficient time dealy 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 4000 kw, to less than .4 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 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 a detector 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 seven miles 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 on page 46. During a normal operating shift, Gate 1 will be 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 authorized 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, mezzaine 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 page 74 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.

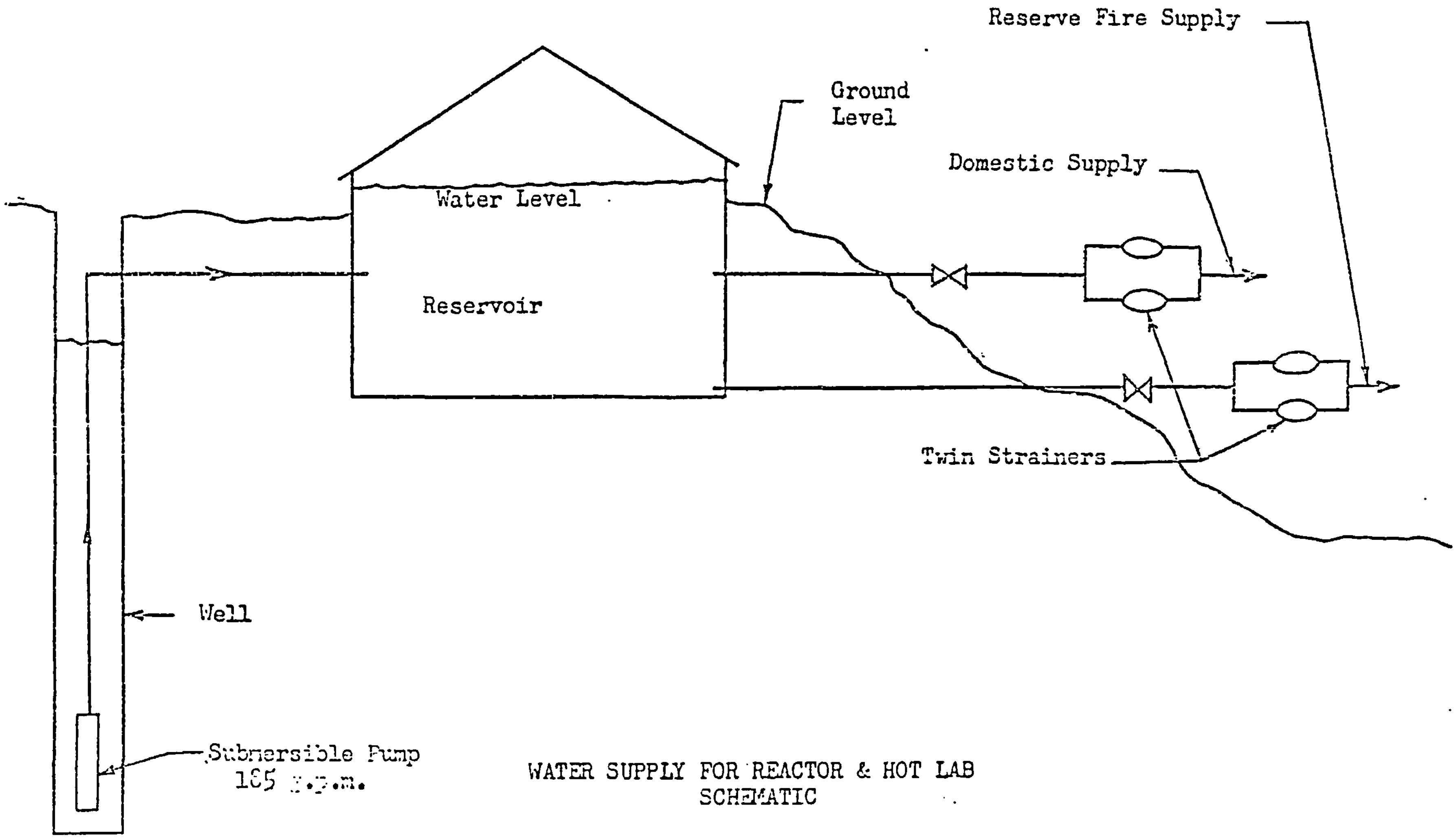
A water supply for the reservoir is obtained from a spring-fed well. A submersible pump set at a depth of 460 ft discharges its effluent to the reservoir at a point located below the frost line. The pump is rated at 185 gpm with a 500 ft head and is operated automatically by means of a high and low level 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. <u>Fuel Management</u>

The fuel elements when not in a core configuration are 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 deterrent against theft is the residual activity of the fuel elements which, when removed from the pool, will set off the building alarm.

The reactor went critical April 29, 1958 and the low power experiments were completed in the first week in December 1958. The reactor was operated intermittently at 100 kw in December 1958 and January 1959. Some high power



WATER SUPPLY FOR REACTOR & HOT LAB
SCHEMATIC

tests were conducted in February and March at 1 Mw intermittent operation. Needed maintenance of the pool and reactor structure are scheduled for April and May. During this time, the reactor will be connected to the proposed 4 Mw cooling system. Routine operation at 1 Mw will begin in May and the power will be raised to 4 Mw by the end of June. It is anticipated that an average of 20% burnup can be achieved. At this rate the first fuel will probably be returned for reprocessing during the first quarter of 1960.

A careful log is kept so that the percent burnup of any given fuel element may be readily determined at any time. Elements will be rotated between the reactor's various flux areas and the gamma pool. By rotating elements, about 10 will be replaced initially and 8-12 every 6 to 12 months thereafter, depending on utilization. During this time our fuel inventory will be increased to compensate for the partially burned up elements in our possession. 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. The elements will then be loaded under water into suitable containers and sent to a reprocessing plant.

H. Emergency Procedures

Emergency procedures have been published which 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 (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 an 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 to the environment. Vehicles equipped with the 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. (See Appendix V for Emergency Procedures.)

I. Health Physics

A permanent health physics organization has been set up and is operative in the building which houses the reactor. The group consists, at present, of a health physics supervisor and one radiation surveyor. Additional radiation surveyors will be added to provide for multiple shift coverage.

The general duties of this group are broken down as follows:

1. The issuance of film badges and pocket chambers for personnel monitoring and the determination and recording of any radiation exposures which might be received.

2. The control of external and internal radiation hazards.

3. The control of radioactive contamination.

4. Investigations of overexposure to personnel.

5. The disposal of radioactive waste.

6. The radioactive laundry.

7. Routine daily, weekly and monthly radiation surveys.

8. Inventory control of all by-product material.

9. Calibrations of radiation measuring instruments.

10. Medical program including blood sampling and counting, bioassay analyses and medical urinalyses.

11. The environmental survey program including routine sampling and radioassay of stream waters, vegetation, soil, and air.

12. Giving assistance advice and cooperation to the supervisors of radioactive work in order to protect personnel against indiscriminate handling of radioactivity.

13. Assisting in the education of workers in fostering a better understanding of the hazards of radiation.

14. Ensuring that all shipments of radioactive material conform to the standards as set by the Inter-State Commerce Commission.

III. ACCIDENTS INVOLVING THE REACTOR

A. Reactor Power Excursions

1. Reactivity Requirements for Reactor Operation

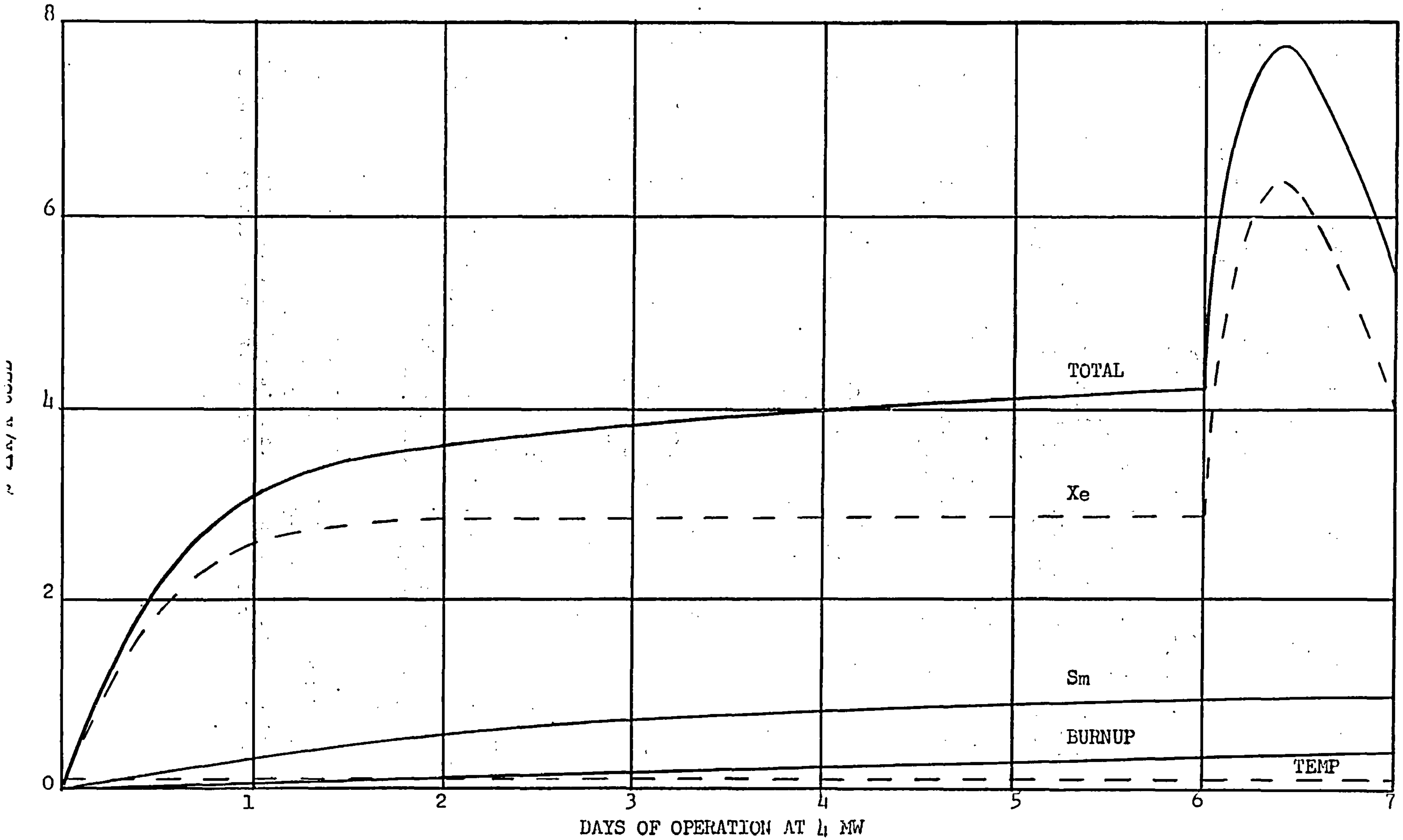
One source of danger inherent in the operation of a reactor of the swimming pool type is the sudden introduction of a large amount of reactivity. It is pertinent, therefore, to state the maximum amount of reactivity which would ever be available for rapid addition. Table VIII lists the reactivity requirements for operation at 4000 kw.

TABLE VIII

Estimated Reactivity Requirements

Source	Reactivity Required
Negative temperature coefficient (-5×10^{-6} Δ k/k per °C)	.001
Poisons (Xe, Sm, etc.)	.048
Experimental requirements	.002
Adequate rate of change of power level	.002
Addition of smallest increment of reactivity available	.003
Xenon override	.010
Total Requirement	.066

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 4000 kw is shown on page 79. It is apparent that starting with 6.6% Δ 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 shutdown 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.



CHA GE IN REACTIVITY AS A FUNCTION OF OPERATING TIME

The 1% excess reactivity for xenon override will allow the operator approximately an hour in which to start-up the reactor after a shutdown. The .001 negative temperature coefficient is larger than would be assumed from previous "bath" type temperature coefficient measurements. CWRR measured the coefficient of the reflector and the "bath" coefficient, and from this separated the effect due to the core alone (Ref. 1). The coefficient for the core is about four times as large as had been indicated by the "bath" coefficient.

In Table VIII the 0.2% $\Delta k/k$ is allowed for easily removed experiments, such as small irradiations and activations. Experiments involving more than 0.2% $\Delta k/k$ will be reviewed by the Curtiss-Wright Nuclear Hazards Committee after being submitted on the form shown on page 81. After approval, the experiment will be installed in a partially unloaded core and the reactor will be brought to power by a critical experiment. Under no circumstances will an experiment worth more than 1.5% be installed in the reactor. The total worth of experiments installed will never exceed 3.2% $\Delta k/k$ including the allowance for small experiments and those beam tubes which are in use.

The decision as to whether an experiment is worth more than 0.2% $\Delta k/k$ and less than 1.5% will have to be based on calculations and experience with similar experiments. However, the worth of all experiments will actually be measured after they are installed and corrective action will be taken if they are found to exceed the appropriate limit.

It is of interest to note that the reactivity available decreases very rapidly during the first few hours of operation at 4000 kw due primarily to xenon buildup. As a result, $\Delta k/k$ is reduced from 6.6% to less than 3.5% in the first day of operation. For most of the operating lifetime of the reactor, 3.5%, rather than 6.6%, can be considered the maximum $\Delta k/k$ to produce a power excursion.

2. <u>Reactivity Worth of Beam Tubes</u>

Another possible method of causing a stepwide 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-in. 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-in. tube

CWRR IRRADIATION REQUEST

Date ______________

Requested By ________________________________ Work Order ________________

Description of Proposed Operation (to include type of facility, type of radiation, dose rate integrated dose, chemical and physical form of material, type of container, special handling procedures required, etc):

Approval by Project Engineering Staff ________________________________

Approval by Health Physics Officer

Does experiment require approval of Nuclear Hazards Committee   Yes   No

Approved by ________________________________
Health Physics Officer

Approval by 0480 Division Chief ________________________________

Approval by Nuclear Hazards Committee (if required) ________________________________
Chairman,
Nuclear Hazards Committee

and 0.10% for an offset 6-in. tube. The total worth of the aluminum in three tubes was found to be 0.25%. The measured worth of the beam tubes in the Curtiss-Wright Research Reactor with a water reflector are as follows (see Ref. 4):

Beam Tube Condition			
1	2	3	% $\Delta k/k$
	(Reactor away from tubes)		----
Flooded	Flooded	Flooded	0.04
Flooded	Empty	Flooded	0.53
Empty	Empty	Empty	1.00

The total worth of the three tubes is approximately 1% $\Delta k/k$. 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 4000 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. 5) and the more recent Spert-1 tests (Ref. 6) 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 the 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 sub-cooled excursion was of the order of five times as great as for the saturated case. Nevertheless, the reactor still could withstand a step Δ k/k addition of 1.5% without mechanically damaging the fuel elements or approaching the melting point of aluminum cladding. According to Ref. 7, 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 airborne material.

The step experiments with sub-cooled 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% Δ 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 also was used to study response to ramp additions of reactivity, i.e., the control rod was withdrawn at a constant rate until a pre-determined Δ 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. 8, 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% Δ 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. 9), and Edlund and Noderer's analysis (Ref. 10) indicate that with the moderator sub-cooled, 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 going into diverging power oscillations. It is planned to operate the CWRR at a bulk water temperature not exceeding 100°F.

4. Conclusions

As indicated above, a step function addition of 1.5% Δ k/k probably can 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 Curtiss-Wright Research 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 0.01% per second and the total Δ k/k added would be 0.70% for the heaviest rod. 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 four 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 would be .10% per second. However, the rods are interlocked so no more than two may be withdrawn for a removal rate of .05% per second. At this rate, it would take over 50 seconds of continual withdrawal of the two possible 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.

In the Borax experiments (Ref. 5), 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 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 larger fraction 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 micrometeorological 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 0.7 Mev is assumed, and if the reactor is assumed to have operated for a long time, then the activity may be expressed as

$$Q\gamma = 7.8\ (10^6)\ t^{-0.2}\ \text{curies/Mw}$$

$$Q\beta = 14.8\ (10^6)\ t^{-0.2}\ \text{curies/Mw}$$

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

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

If continuous operation at 4 Mw is considered, the time required to reach the maximum average burnup of 20% is $1.5(10^7)$ 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 4 Mw)
Sr-89	.048	53 d	$1.45\ (10^7)$
Sr-90 + Y-90	.059	25 y	$2.7\ (10^3)$
Y-91	.059	61 d	$1.7\ (10^5)$
Ba-140 + La-140	.063	12.8 d	$2.1\ (10^5)$
Ce-144 + Pr-144	.061	282 d	$7.2\ (10^4)$
I-131	.029	8.14 d	$1.0\ (10^5)$

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 20% 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 3%; Sr-89, Y-91, Ba-140 + La-140 by 66%; and Ce-144 + Pr-144 by 32%; 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 C_x, C_y, 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. 11) 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 conditions 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
C_y, C_x	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 throughout its passage can be expressed as

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

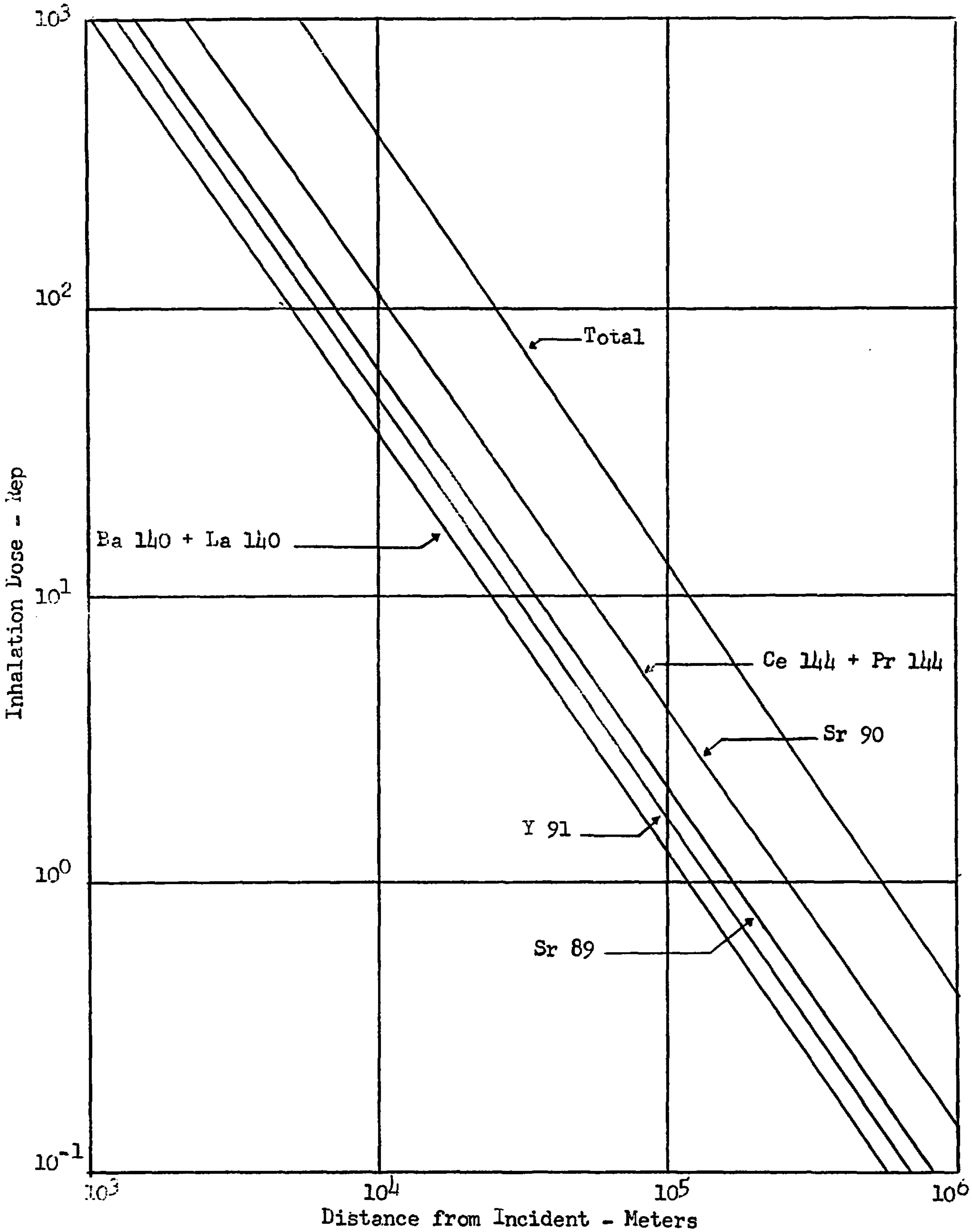
Where D = dose in rep to the critical organ
M = mass, gm of the critical organ
f = fraction of inhaled isotope reaching the critical organ
λ = the effective decay constant of the isotope in the critical organ, in day^{-1}
Q = source strength, in curies
E = average energy of the isotope per disintegration, in Mev
t = time after inhalation over which the dose is 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. Pages 90 and 91 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. Page 92 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. Page 93 shows the accumulated bone dose as a function of time after the incident.

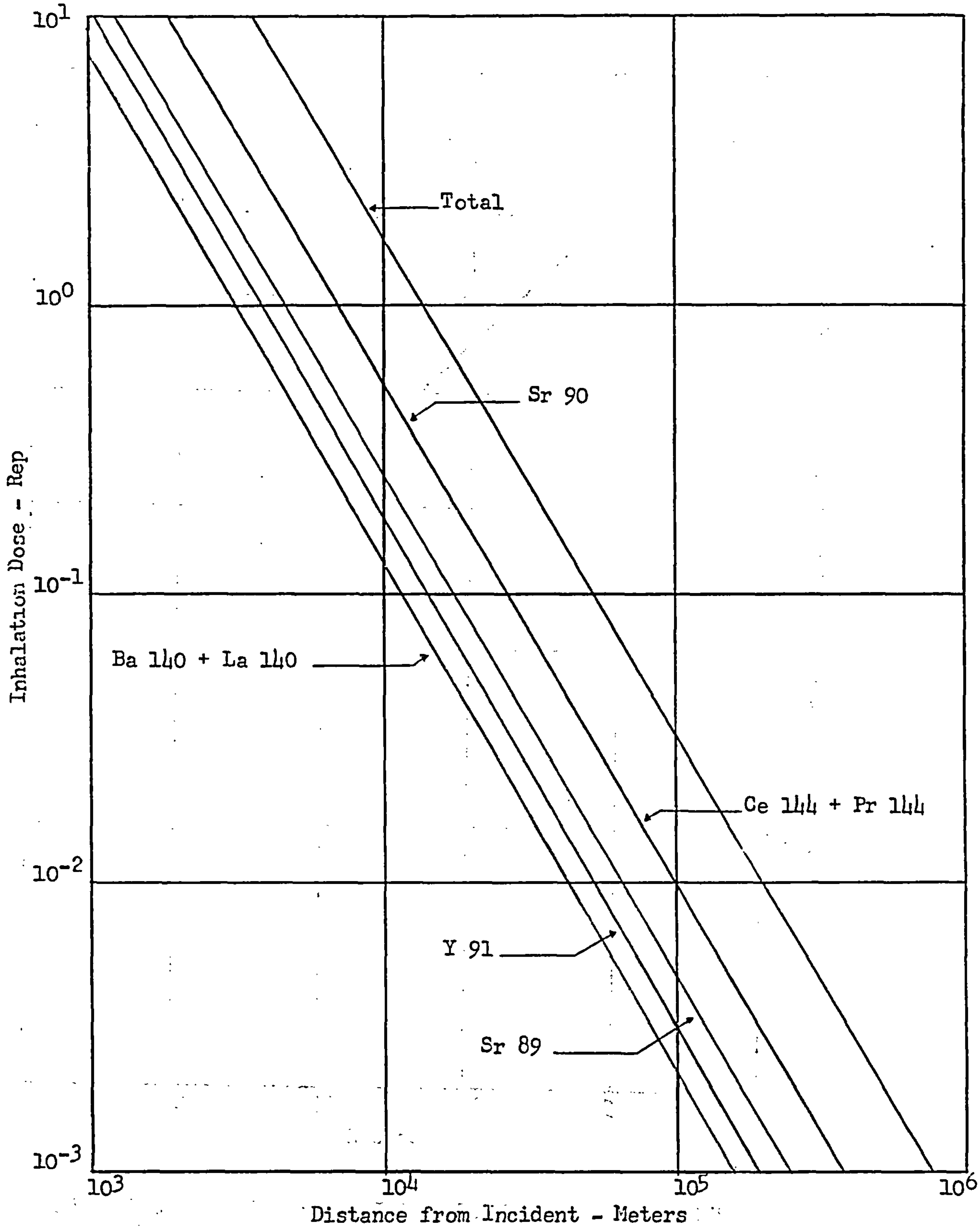
4. External Dose from the Passing Cloud

a. External β-Dose. The external β-dose to a person standing at the cloud axis throughout its passage is given by Ref. 12 as

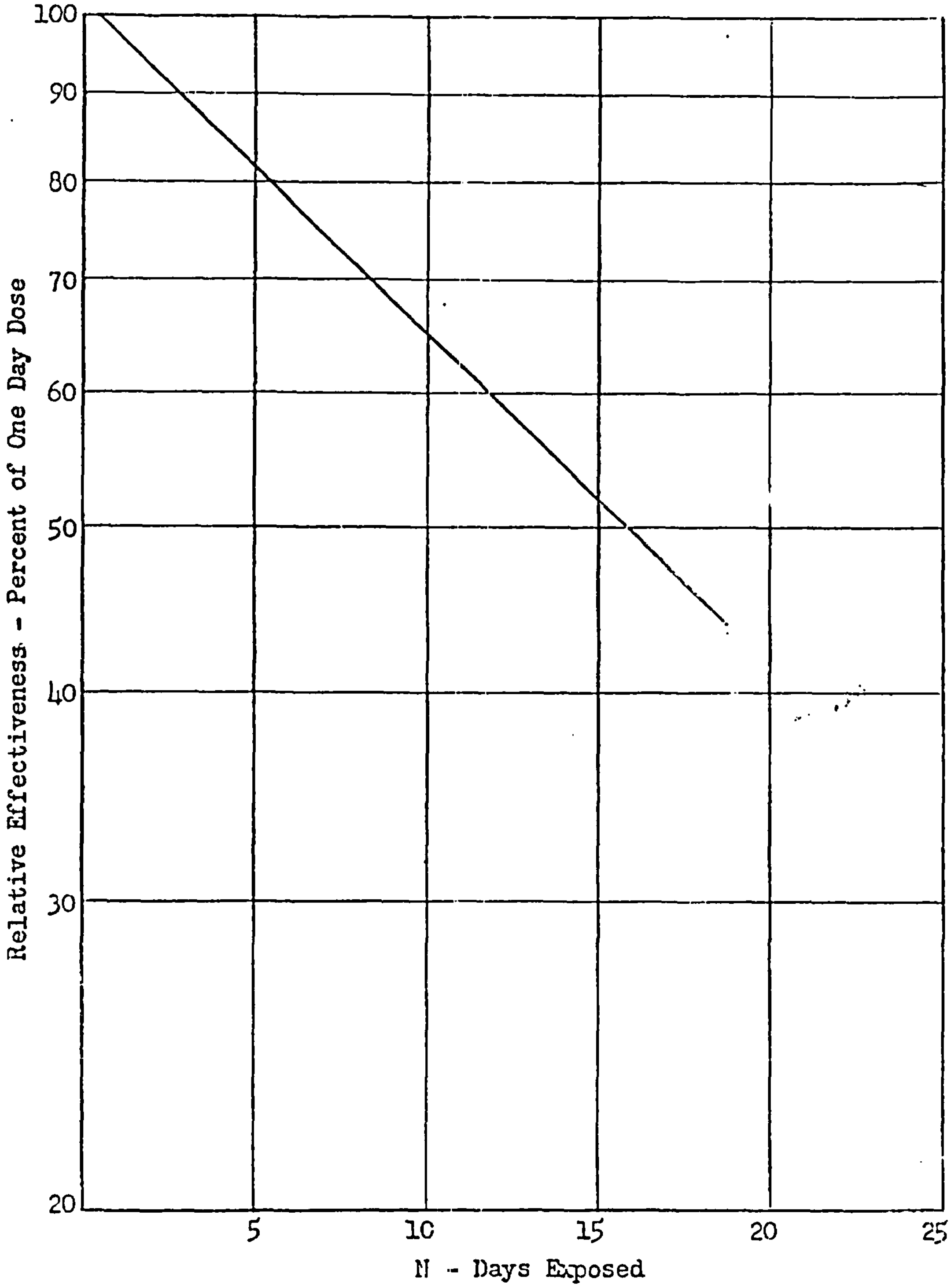
$$D_\beta = \frac{Q_\beta \exp\left[\frac{h^2}{C_z^2 \; x^{2-n}}\right]}{\pi u \; C_y C_z x^{2-n} \quad 1.06(10^{11})} \quad \text{rep}$$



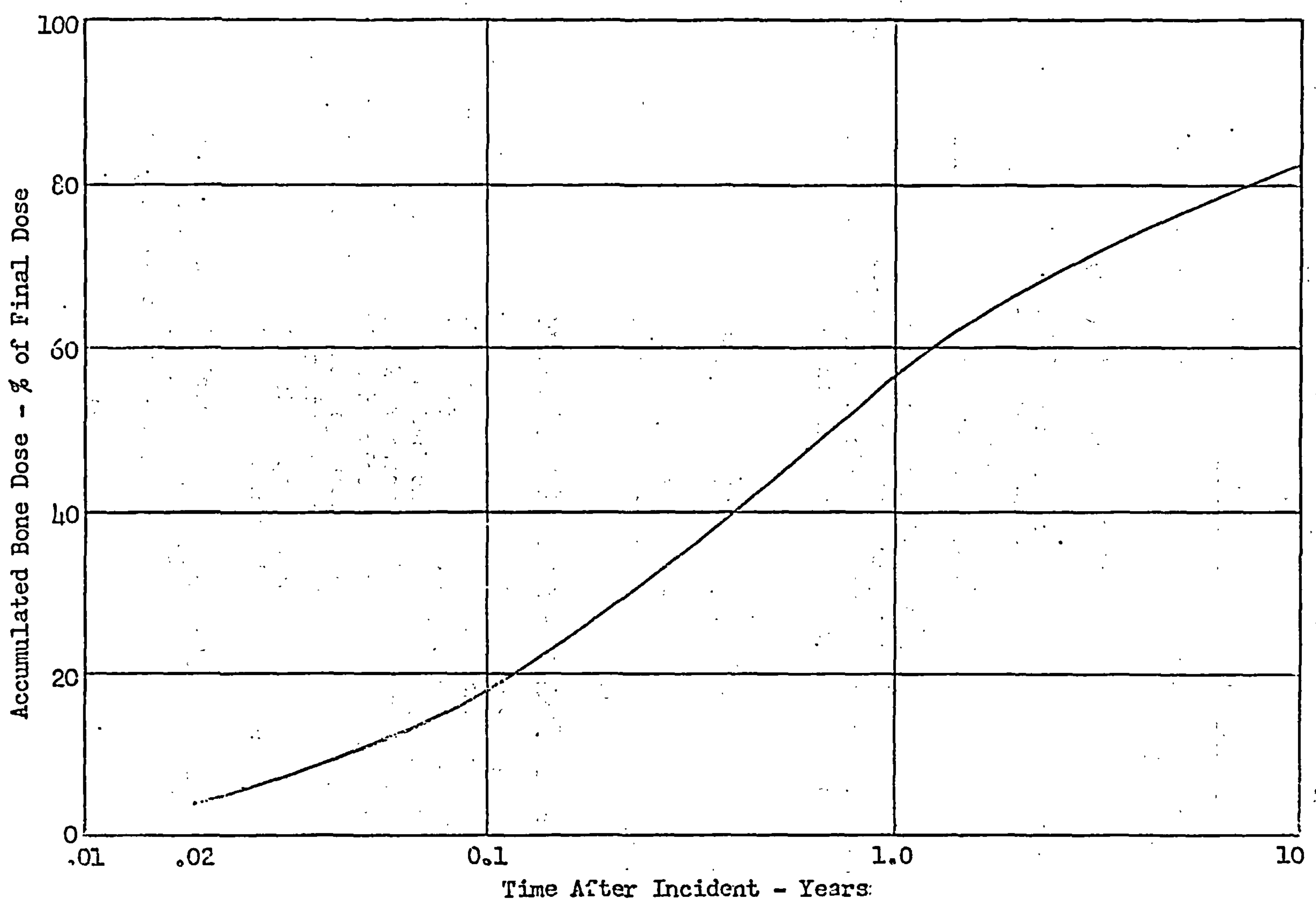
TOTAL INTEGRATED INHALATION DOSE TO BONE INVERSION CONDITIONS
100% GROUND LEVEL RELEASE AFTER REACTOR
OPERATION AT 4 MW FOR 231 DAYS



TOTAL INTEGRATED INHALATION DOSE LAPSE CONDITIONS
100% GROUND LEVEL RELEASE AFTER REACTOR
OPERATION AT 4 MW FOR 231 DAYS



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



ACCUMULATED BONE DOSE AS A FUNCTION OF TIME where Q_β is the total β-source strength in Mev/sec, and h is the height in meters of the cloud. Page 95 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. Page 96 gives the external γ-dose as determined by Holland's method (Ref. 12).

5. γ-Dose from Ground Deposition (See page 97)

The γ-dose from activity deposited on the ground (see Ref. 12) 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 (curies per square meter)
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

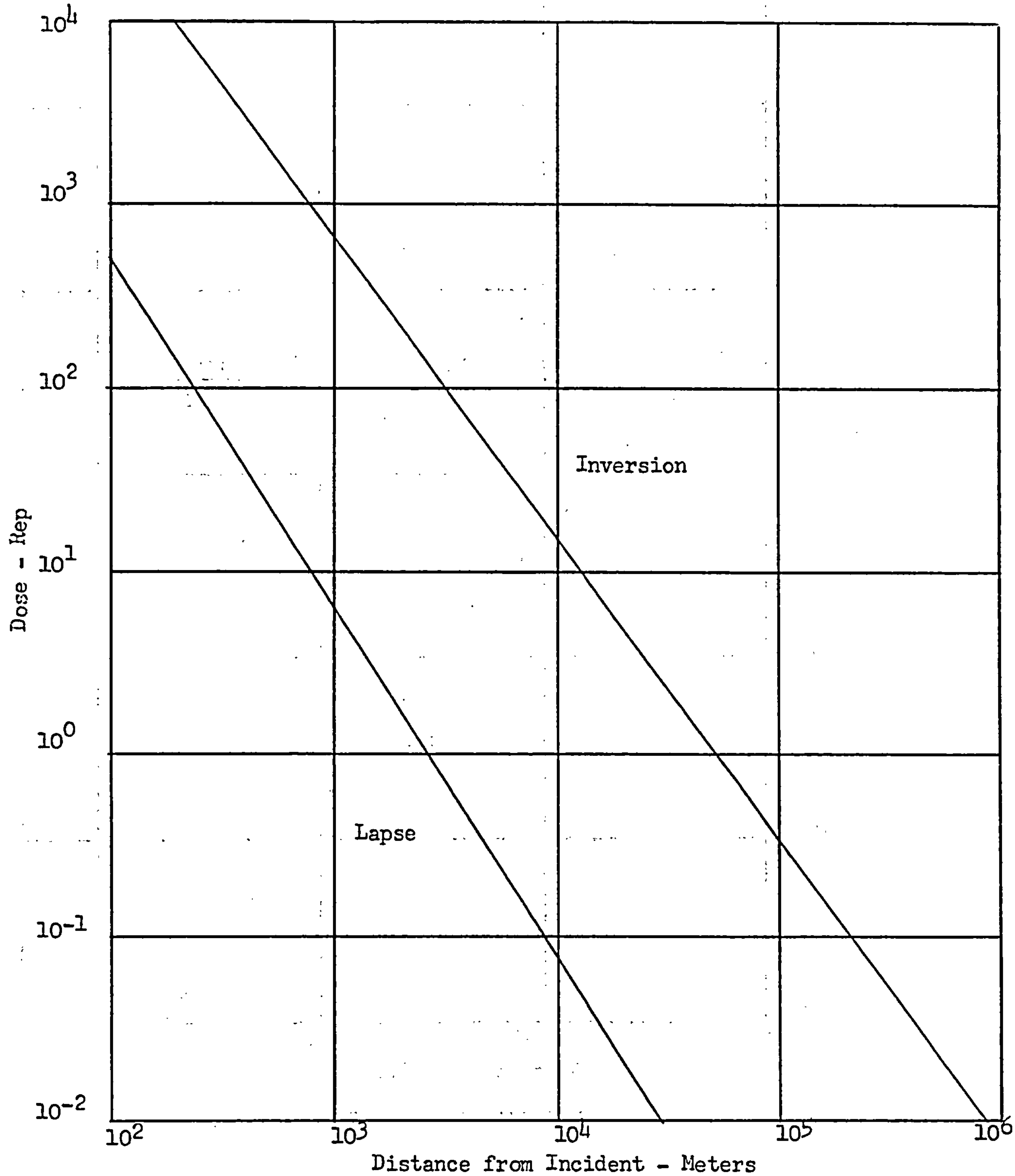
$$W_o = \frac{nQ}{2e\pi\ \frac{1}{2}C_y x^{(2-n/2)}} \qquad c/m^2$$

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.

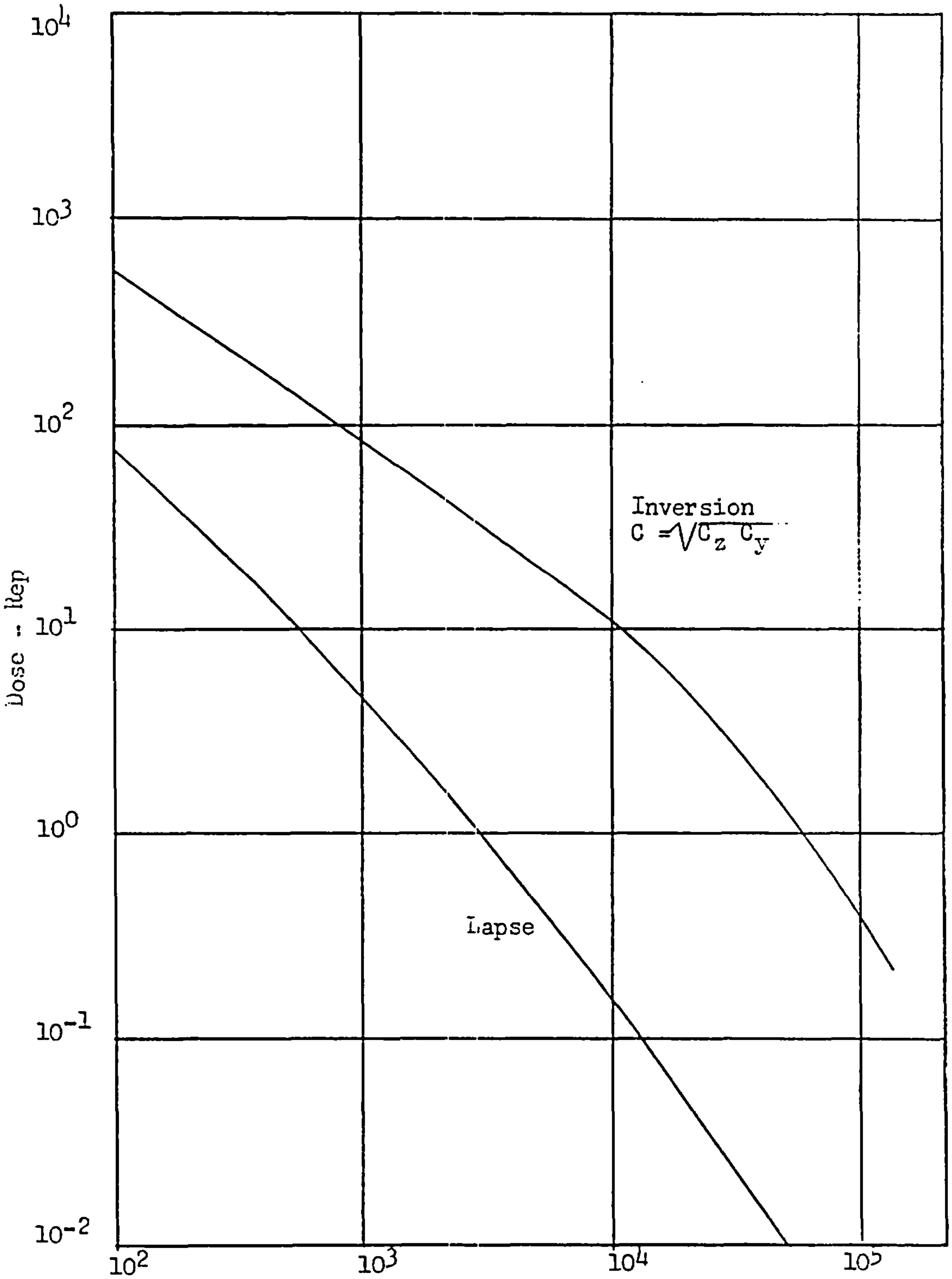
6. Modifying Factors for a Credible Accident

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. 5) gives "no indication that any large fraction of the fission products left the vicinity of the reactor." Workers at Oak Ridge (Ref. 13) 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%.

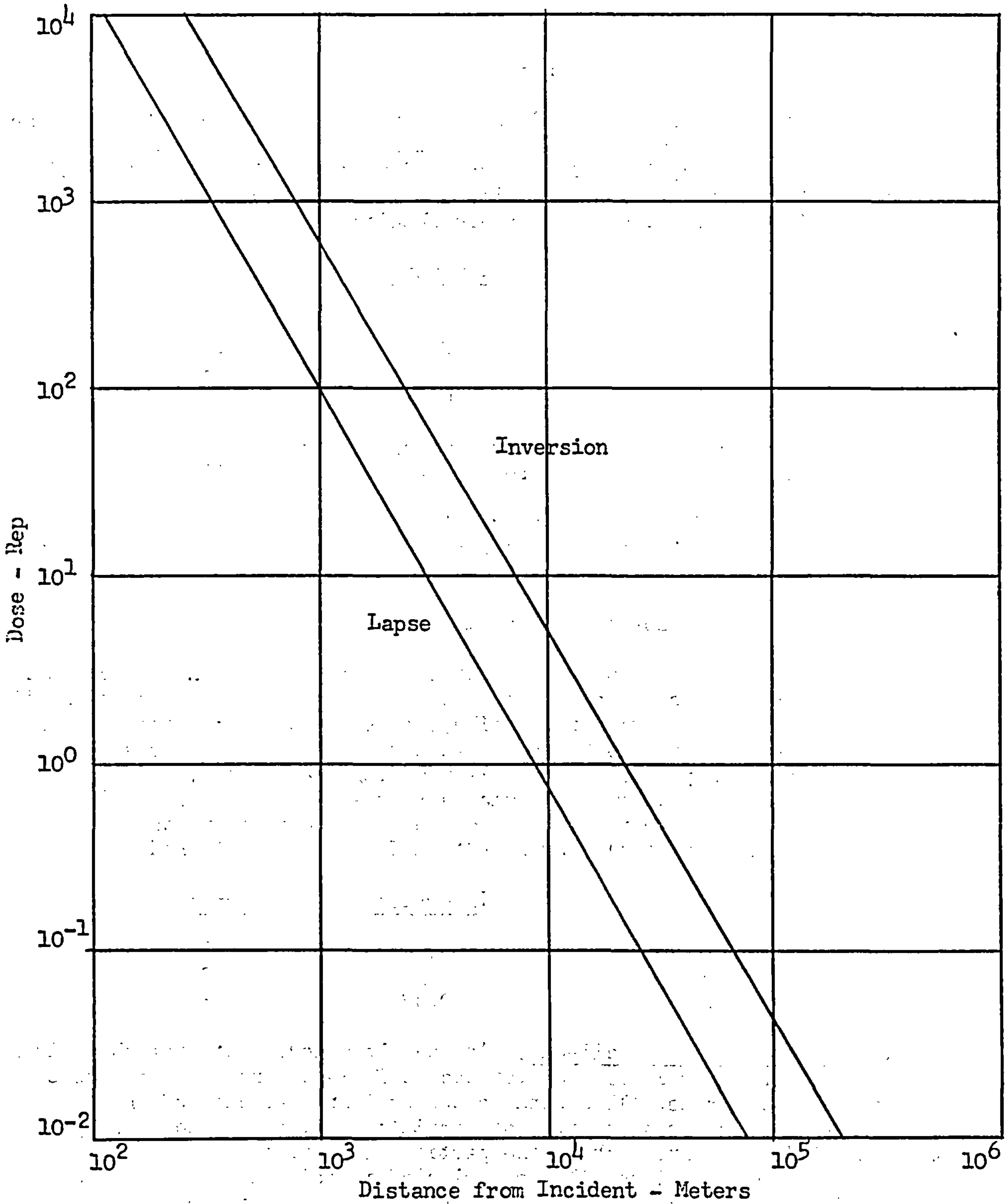
b. Rise of the Cloud. If an appreciable amount of material is released it would very likely be at an elevated temperature and hence would rise. This rise would have a marked effect on the dose, particularly at



EXTERNAL BETA DOSE FROM CLOUD



EXTERNAL GAMMA DOSE FROM CLOUD



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

short distances. The reduction factor is

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

where h is the cloud rise in meters.

According to Sutton (Ref. 12) 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 are used, the following reduction factors are obtained.

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 depletion 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

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

where y, the half-width, is the distance in meters from the axis to the point where the concentration is p percent. Page 100 gives the half width where p is 10%.

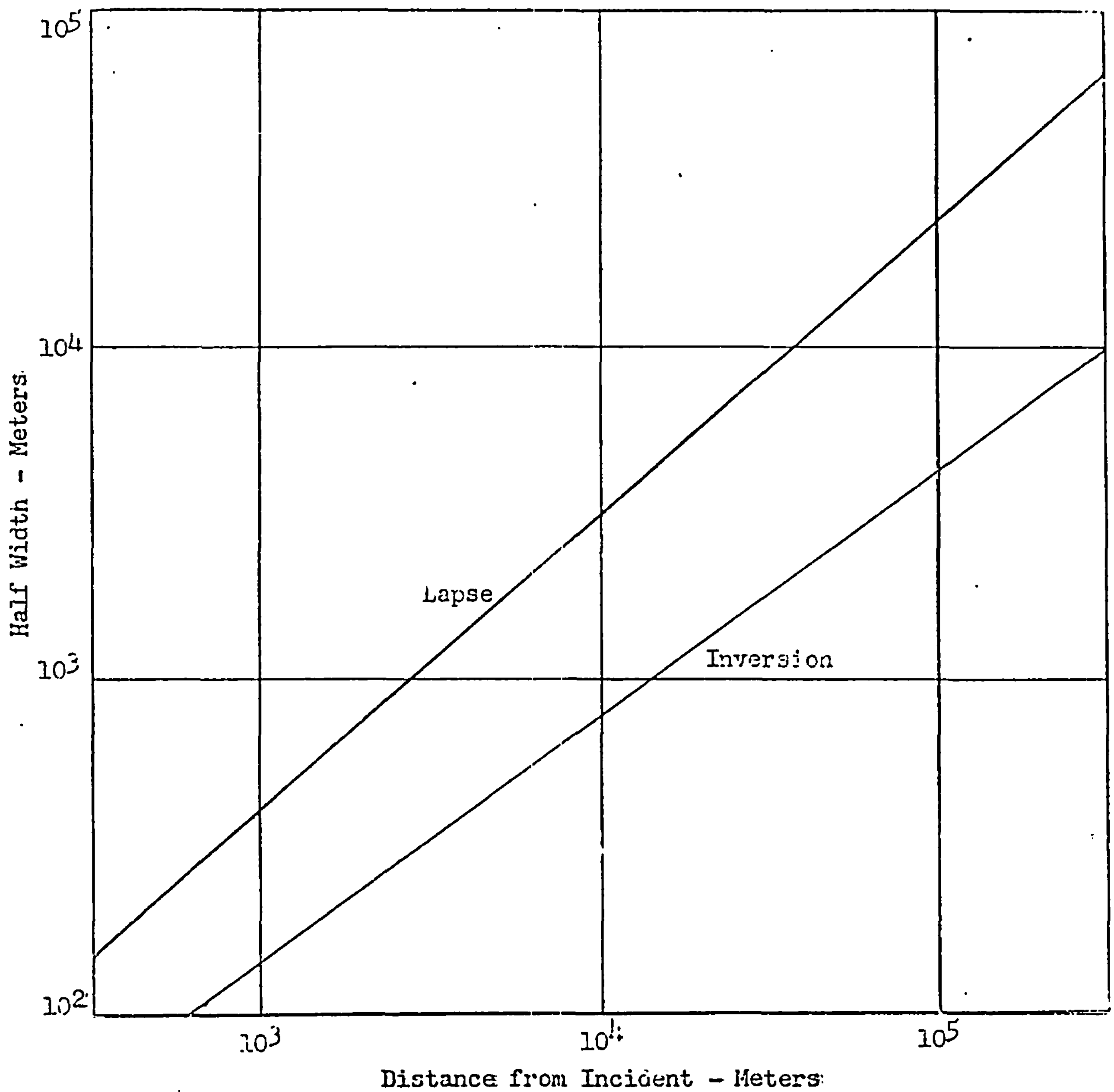
8. Maximum Credible Accident

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 then 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



CLOUD HALF-WIDTH

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 dose	0.60 rep
External beta dose	0.04 rep
External gamma dose	0.04 rep
12 hour fall-out dose	0.40 rep
12 hour rain-out dose	3.20 rep

This obviously represents no serious hazard.

<u>Location 2.</u> 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:

	Inversion (rep)	Lapse (rep)
Inhalation	49.2	.24
External beta	2.0	.01
External gamma	1.2	.03
12 hour fall-out	0.8	.12

If depletion of the cloud, cloud rise, a lower fission product inventory, and 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. 14) has suggested as a maximum permissible 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 2.0 reps and declines steadily in each succeeding week. An initial cloud rise of less than 65 meters would reduce the dose to the limit. The maximum case is about twice the 25 rep dose sometimes suggested as a maximum acceptable emergency dose; a dose at which no injury is expected.

There is over two hours in which to evacuate this area if there is a possibility of the cloud traveling in this direction.

The doses at locations 3 and 4 likewise represent no significant hazard being approximately 18 and 10 rep respectively for the inversion case with no modifying factors considered. Doses at these locations under lapse conditions 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-instantaneous 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 event which 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 hr after the incident.

The exposure of the gastrointestinal tract, if all of the activity remains there, will be of the order of 0.24 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 2.0 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.

REFERENCES

1. Geisler, G. S., Roberts, W. J., and Sjoborg, W. F., "The Temperature Effect of the Reflector on a Water Moderated, Water Reflected Reactor," Paper Presented at the Winter Meeting of the American Nuclear Society, December 10, 1958, Curtiss-Wright Corporation, Research Division.

2. Johnson, E. B., et al, Reactivity Measurements with the Bulk Shielding Reactor: Control Rod Calibrations; Beam Hole Coefficients; Pastial Reflector Coefficients, Oak Ridge National Laboratory, September 16, 1955, (ORNL-1871).

3. Personal Communication from George Geisler, Pennsylvania State University.

4. "Beam Tube Void Effects on Reactivity and Flux Measurements," Curtiss-Wright Research Reactor Technical Note #6.

5. Dietrich, J. R., Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled Water-Moderated Reactor, Borax-1 Experiments, Argonne National Laboratory, 1954, (AECD-3668).

6. Nyer, W. E., et al, Experimental Investigations of Reactor Transients, Phillips Petroleum Co., Idaho Operations Office, April 20, 1956, (IDO-16285).

7. McCullough, C. R., ed., Safety Aspects of Nuclear Reactors, The Geneva Series on the Peaceful Uses of Atomic Energy, New York, D. Van Nostrand, Inc., 1957.

8. Forbes, S. C., et al, Instability in the Spert-1 Reactor Preliminary Report, Phillips Petroleum Co., Idaho Operations Office, October 10, 1956, (IDO-16309).

9. Breazeale, W. M., A Low Cost Experimental Neutron Chain Reactor, Oak Ridge National Laboratory, April 15, 1952, (ORNL-1105, Part 1).

10. Edlund, M. C., and Noderer. L. C., Analysis of Borax Experiments, 1955, Reissued 1957, CF-57-7-92.

11. USAEC, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, March 1957, WASH-740.

12. Meteorology and Atomic Energy, prepared by the U. S. Department of Commerce, Weather Bureau, for the U. S. Atomic Energy Commission, 1955.

13. Corngold, N., Summary Hazards Report on Enriched Fuel Element Loading for the BNL Research Reactor, Brookhaven National Laboratory, May 24, 1956, (BNL-3146).

14. Morgan, K. Z., et al, Maximum Permissible Concentration of Radioisotopes in Air and Water for Short Period Exposure, Proceedings of the International Conference on Peaceful Uses of Atomic Energy, 1955, P/19.

15. National Bureau of Standards, Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water, Handbook 52, 1953.

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 (Appendix I, Ref. 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 (Appendix I, Ref. 1, 2, and 3) reports that an anticline and 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 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 ground-water 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 five 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 Pennsylvanian Drilling Company of Pittsburgh, Pennsylvania, were abandoned because of low production. Wells No. 3 and 4 drilled by Kohl Brothers Drilling Company of Harrisburg, Pennsylvania at about the center of the site at engine test stands No. 1 and No. 2 were successful wells and are being used at these locations.

Well No. 5 drilled by E. W. Webber's Sons Drilling Company of Luthersburg, Pennsylvania, is located at the Research Reactor and Radioactive Materials Laboratory. The drillers logs of the No. 1 and No. 2 wells are given in Table A-2.

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 drillers of the particular wells were contacted and the following information obtained.

	Well No. 3	Well No. 4	Well No. 5
Diameter, in.	6	6	8
Depth, ft	250	241	400
Length of casing, ft	15	15	20
Water level (static), ft	75	72	213 (above pump)
Pump setting, ft	200	180	369 (to intake)
Pump, hp	3 Fairbanks Morse	3 Fairbanks Morse	20
Capacity, gpm	52	140	120
Pumping water level, ft	205	-----	160 (above pump)

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 five 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.

TABLE A-2

Drillers Logs of Test Well 1 and 2

Test Well No. 1 Thickness (in feet)	Test Well No. 1 Depth (in feet)	Strata	Tentative Geologic Correlation	Test Well No. 2 Thickness (in feet)	Test Well No. 2 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 and 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 rock
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

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 (Appendix I, Ref. 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	I	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 (Appendix I, Ref. 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.

3. Reservoirs

There are two water reservoirs located on the Quehanna site. Reservoir No. 1 is located 2 miles Northeast of the main entrance with a capacity of one million gallons, fed by a stream with a 400 gpm flow. This reservoir furnishes both potable and fire supply water to all buildings, near the main entrance.

Recreation Reservoir No. 2 is located approximately 9 miles from the main entrance and 4 miles north of the Research Reactor Building.

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 seismographs 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 of 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 movable 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 (Appendix I, Ref. 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

REFERENCES

1. Cathcart, S. H., 1934, Geologic Structure in the Plateau Region of Northern Pennsylvania and Its Relation to the Occurrence of Gas in the Oriskany Sand; Pennsylvania Topographic and Geologic Survey, Bull. 108.

2. Cathcart, S. H., 1939, The Possibility of Finding Gas in Cameron County, Pennsylvania; Pennsylvania Topographic and Geologic Survey, Bull. 109.

3. Cathcart, S. H., 1934, The Possible Occurrence of Gas in the Oriskany Sand of Elk County, Pennsylvania; Pennsylvania Topographic and Geologic Survey, Bull. 110.

4. Lohman, S. W., 1938, Ground Water in South-Central Pennsylvania; Pennsylvania Topographic and Geologic Survey, Bull. W 5.

5. Lohman, S. W., 1939, Ground Water in North-Central Pennsylvania; Pennsylvania Topographic and Geologic Survey, Bull. W 6.

6. Mangan, J. W., Letter to Richard Greene, Curtiss-Wright Corp. Research Division dated June 8, 1955, giving streamflow data near the Quehanna site.

7. Industrial Utility of Water in Pennsylvania. Chemical Character of Surface Water 1944 to 1946. Pennsylvania Department of Commerce, State Planning Board Pub. 17, 1947.

8. Heck, N. H., 1947, Earthquake History of the United States, Part I. Continental United States (exclusive of California and Western Nevada) and Alaska: U. S. Coast and Geodetic Survey Ser. 609, revised.

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

1. 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.

2. 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.

3. 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 Atlantic 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.

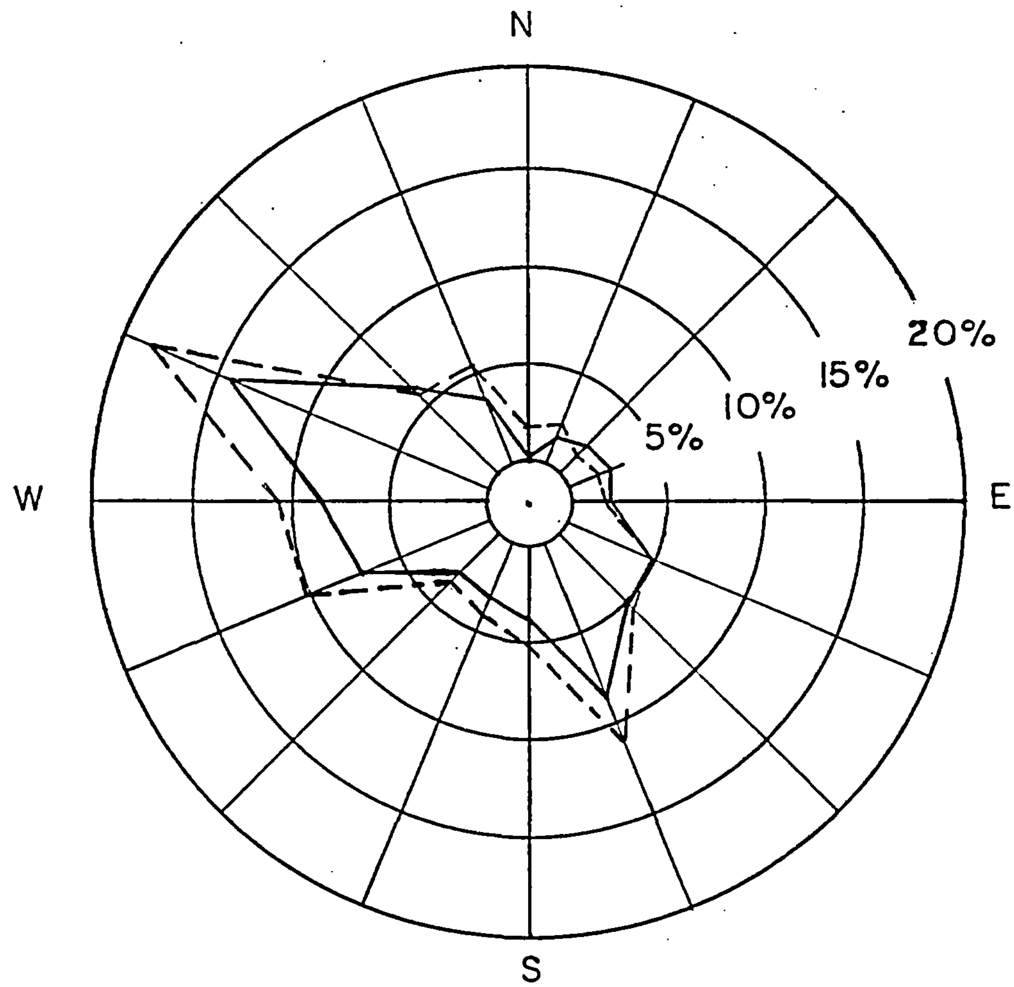
1. Surface Wind Direction

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. Page 119 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 east 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

ANNUAL FREQUENCY AND WIND DIRECTION

... ...t wind speeds are generally observed with westerly winds. The ...imum wind speed observed during this period of record was 50 mph ... an instantaneous peak gust reaching 60 mph. It should be noted, ...ever, that the wind observations were taken from an instrument ... ...ipped with a recorder. Thus, higher velocities may have occurred ... being observed. It is estimated that rare wind gusts which ... reach as high as 80 to 90 mph are not improbable.

... 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 P...nsylvania an analysis of 35 years of record shows that during this ...iod five tornadoes occurred in the area covered by the five ...unties surrounding the Quehanna site. Because of the usually short ... covered by any single tornado and the small width of this path, ... would be impractical to assign probabilities of a tornado striking ... particular building or installation. However cognizance should be ... of the fact that this phenomenon can occur at this site.

... data on winds occurring with precipitation was included in order ... one might consider the effect of washout of possibly airborne ...taminants. The wind frequency during periods of low visibility was ...luded as a method of estimating the wind direction during periods ... atmospheric stability. It is of considerable interest to note that ... 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 ... periods of atmospheric stability as represented by the low visibility ...dition.

.. 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 ≤ 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 2½ in. per month at minimum periods to around 4½ 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 unusually 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 valleys 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 Phenomena
(Average Number of Days per Year)

Thunderstorms	Visibility $\leq 1/4$ mi	Snow ≥ 1 in.	Temperature ≤ 32	Temperature ≤ 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-southwest, 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 = Curies 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})

$$\text{TID} = \frac{2Q}{\pi C^2 \quad ux^{2-n}}$$ from Suttons equation. (Ref. 12)

The integrated dose from time of inhalation to time t is

$$\text{Integrated dose} = \int_0^t Ke^{-\lambda t}\,dt \qquad K = 5.5\,(10^7)\cdot\frac{E}{M}\,(\text{TID})\,Bf$$

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

The integrated dose in rep is therefore

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

Using the values of E, f, M and λ from Handbook 52 (Ref. 15)

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

A. Reactor Start-Up Checkout Procedure (CWR Procedure 103)

		Flooded	Plugs	Doors
1. Make sure bridge is locked in position and motion switches are set. Check condition of beam tube, plugs, and doors.	1.	______	______	______
	2.	______	______	______
	3.	______	______	______

2. Turn on lights which illuminate core. Make sure that source is in proper position, and that loading and samples are as indicated by the reactor log book. Refer to CWR procedure 102. ______

3. Check fission product delay neutron monitor. Refer to CWR procedure 204. ______

4. Check ion exchange fission product monitor. Refer to CWR procedure 203. ______

5. Set zero on micro-microammeter. Make sure that this instrument is on the most sensitive range, compatible with residual reactor power. ______

6. Check compensation of linear and Log N C.I.C. Chambers. Refer to CWR procedure 201. ______

7. Read and record CIC power supply voltage.
   1. + ______ - ______
   2. + ______ - ______

8. Record reading on power integrator. Refer to CWR procedure 207. ______

9. Check Jordan Rams system. Refer to CWR procedure 205. ______

10. Energize recorders. Manually standardize recorders after recorders' servo systems are operating. Check chart supply. ______

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

12. Close scaler "count" switch. Record gain setting on amplifier, and PHS setting on both A-1 and scaler.
    - A-1 Gain ______
    - A-1 PHS ______
    - Scaler PHS ______

13. Energize electromagnets by turning magnet key switch. Observe magnet current meters on safety amplifiers to make sure current is flowing. Make sure magnet currents and "bus protect" adjustments are correct; if so, read and record magnet currents. Refer to CWR procedure 202. ____________

#1____________ #2____________ #3____________ #4____________

14. Test #2 (power level) safety system by withdrawing all shim-safety rods about three (3) inches and inserting "Scram" on #2 safety amplifier panel. All rods should drop. ____________

15. Test #3 and #4 (power level) safety systems in same manner. ____________

16. Withdraw fission chamber and make sure that count rate drops. Reinsert fission chamber. ____________

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

18. Calibrate Log N - Period Amplifier;
   (a) Turn selector switch (located on chassis of instrument) to ground position. Log N meter should read .0001 (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 "Operate." ____________

   (f) Raise shim safety rods to 3 inches. ____________

   (g) Turn selector switch to "Lo Calibrate;" all rods should drop. ____________

   (h) Turn selector switch to "Operate." ____________

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

Period Slow Shutdown ________ Shim Range ________
Radiation high level ________ Coolant Hi-Activity ________
Count Rate ________

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

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

22. 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, refer to CWR procedure 301.
1. Make sure primary and secondary cooling system is ready for operation. Check valves to make sure water is flowing through core, etc. ________

2. Close plenum door. ________

3. Start pump. ________

4. Make sure proper core effluent temperature bulbs are connected. ________

5. Read and record water temperature and flow rate after flow has stabilized. Refer to CWR procedure 305.

Core Effluent Temp. ________ HX Effluent Temp. ________
Core Diff. Temp. ________ Flow ________

23. Remarks (list any temporary changes during checkout, etc.) ________

____

____

____

24. Make sure that all annunciator indications and the reactor control and safety system in general are normal and ready for start-up. ________

Operator ________ Date ________
Reactor Engineer ________ Time ________

B. Reactor Shutdown Checkout Procedure (CWR Procedure 108)

1. Make sure rods are at fully insert position. ____________
2. Turn off recorders. ____________
3. Stamp and record data on Log N recorder chart paper. Refer to CWR procedure 102. ____________
4. Turn scaler count switch to "Off" position. ____________
5. Set scaler selector switch to most sensitive position. ____________
6. Insert fission chamber, compatible with residual power. ____________
7. Set micro-microammeter range selector switch to most sensitive range, compatible with residual power. ____________
8. Turn magnet power key "Off," remove key and give to reactor engineer. ____________
9. Turn Log N selector switch to "Ground." ____________
10. Do not "clear" period fast scram, safety amplifier indication, and 140% neutron "flux" annunciations. ____________
11. Record reading on power integrator. Refer to CWR procedure 207. ____________
12. Read and record the following: Refer to CWR procedure 305.

    Core Effluent Temp.____________ HX Effluent Temp.____________

    Core Diff. Temp.____________ Flow____________

13. Turn off forced cooling pump. ____________
14. Check fission product delay neutron monitor. Refer to CWR procedure 204. ____________
15. Check ion exchange fission product monitor. Refer to CWR procedure 203. ____________

16. Make sure fuel handling tool and bridge handwheel are padlocked. __________

17. Make sure primary and secondary cooling system is secure for shut-down conditions. Refer to CWR procedure 302. __________

18. Perform necessary duties in connection with experiments and sample irradiation as directed by the reactor engineer. __________

19. Turn underwater lights off. __________

20. Remarks (list any temporary changes during checkout, etc.)__________

21. Make sure log book and all records are complete. Refer to CWR procedure 102. __________

Operator__________ Date__________

Reactor Engineer__________ Time__________

APPENDIX V -- GENERAL EMERGENCY PROCEDURES FOR THE RESEARCH REACTOR SITE

A. Purpose

Emergency conditions which may involve a serious hazard to personnel may arise in the reactor or hot lab areas, building 14, and require special action. This procedure defines the general action which will be required in such emergencies. In addition, each Supervisor is to prepare and put into effect such supplementary procedures as may be necessary to assure that those employees working under him are prepared to meet such emergencies in an organized manner. These supplementary procedures shall be approved by Health Physics prior to posting.

B. Definition of Emergency Conditions

Emergency conditions, for purpose of this procedure, shall be: unusual hazardous conditions which cannot be handled by the existing personnel in the immediate vicinity, and/or conditions not fully under control in which potential or actual hazards exist to the personnel in the immediate vicinity or any greater area.

The health hazard is liable to arise from two conditions which may appear independently or simultaneously: a high level of external radiation; or a release of radioactive contamination, particularly as fine dust in the atmosphere.

The first, a high level of external radiation, would effect only a small area except under extreme conditions. The most likely hazard arises from the second possibility of a spread of contamination. The chief hazard would arise from breathing in the radioactive dust or from getting radioactive matter in custs or wounds.

C. Announcement and Termination of Emergency

1. Announcement of Emergency

Any employee who finds any condition which he believes to be an emergency as defined above, shall report it immediately to his supervisor. The supervisor, if he agrees that an emergency exists, shall ensure that the siren is sounded and shall announce the location and nature of the emergency over the public address system.

If the supervisor is not immediately present and immediate action is deemed necessary, the person discovering the emergency shall take this action.

2. Termination of Emergency

When it has been decided that the state of emergency should be terminated this should be announced over the public address system. If personnel have been sent to building 7, they should be notified.

3. General Procedures Applicable to An Emergency Condition

1. The building 14 guard, upon hearing the announcement of an emergency condition, shall notify the Communication Center by radio in building 7 (8:00 a.m. to 4:30 p.m., Extensions 403-404 -- 5:30 p.m. to 8:00 a.m., Telephone 645). The guard will indicate that a "Condition Red" exists at building 14 and shall also indicate the wind direction and its velocity. The Security Supervisor or his delegate will act as co-ordinator at the Communication Center.

2. Upon receiving notice of an emergency condition in building 14, the guard at the Communication Center in building 7 shall notify the following in the order shown:

During the Day

Fire Department	Ext. 333
Security Supervisor	Ext. 403-404
General Offices (Transportation)	Ext. 401-400
Nurse in building 7	Ext. 511
Health Physics Section, Building 7	Ext. 347
Office of Manager of Nuclear Power Department	Ext. 686-687
Plant Engineer	Ext. 382-383-335

During holidays, week-ends, or between the hours of 4:30 p.m. and 8:00 a.m. daily, the following should be notified.

Fire Department	Ext. 333
Security Supervisor	Clearfield PO 5-3283
Research Reactor Division Chief	St. College AD 8-8272
General Offices (Transportation)	Ext. 400
Research Reactor Health Physics Supervisor	Philipsburg DI 2-3082

Manager of Nuclear Power Department	Quehanna Lodge Ext. 315, or Washington, D.C. HE 9-1512
Health Physics Section Head	Clearfield PO 5-9536
Plant Engineer	Clearfield PO 5-4787
Industrial Relations Manager	Clearfield PO 5-8220

Personnel as notified above will stop at the Communication Center in building 7 prior to proceeding to building 14.

3. The fire department, on receipt of notification, shall proceed to building 14 with all available firemen.

4. The nurse, on receipt of notification, shall proceed to the main lobby of building 7, carrying her emergency kit with her, and will board the ambulance to building 14 or the assembly point.

5. General Offices, on receipt of notification, shall immediately dispatch all available vehicles to building 7 to transport necessary personnel and equipment to the scene of the emergency. If plant transportation is not available, it will be necessary to use privately owned vehicles. In addition, it shall be the responsibility of the General Offices Supervisor to dispatch the radio-equipped vehicle to the assembly point as quickly as possible where it may be used for communication purposes.

6. The General Offices Supervisor and the Plant Engineer shall arrange to provide all available services under their control in order to best meet the emergency.

7. In addition to his duties delegated in Section VI, the Head of Health Physics will ensure that personnel, equipped with appropriate radiation monitoring instruments, proceed to the assembly point as defined in Section V where they will monitor all personnel arriving from the building 14 area.

8. The Chief of the Research Reactor Division or his delegate shall direct the handling of the emergency at building 14. In particular, he shall have collected immediately the following information:

   Data on the number of injured or trapped persons in the emergency area.

   The results of an activity survey for radiation and contamination hazards.

   Condition of all building services, steam, water, etc.

9. Telephones or radio at the emergency site are not to be used during the time of the emergency except by persons delegated by the Chief of the Research Reactor Division. It is essential that this regulation be obeyed strictly in order that the communication channels may be clear to facilitate handling of the emergency.

10. Supervisory personnel are responsible for the alerting of personnel under their jurisdiction or who are working in areas in or around the reactor building in which the emergency signals cannot be heard.

E. Special Procedures Applicable to An Emergency Condition

On the sounding of the siren and/or the announcement of an emergency condition over the P. A. System the following procedures shall become operative.

1. All personnel shall don their respirators and shall leave the building at once, after securing classified documents and taking elementary precautions to prevent fire and other similar damage. They shall proceed approximately 50 yards upwind of building 14 and wait instructions.

   Personnel not required to combat the emergency will be notified by the Research Reactor Division Chief or his delegate. They will take any available transportation and proceed to building 7.

   An assembly point, designated by the Communication Center coordinator, will be set up beyond the hazard area where outgoing vehicles will be held until such time as complete monitoring of the people leaving the area can be arranged. All incoming vehicles, except those necessary to combat the emergency, will be stopped and not allowed to proceed to the building 14 area.

2. The following personnel, equipped with respirators, shall remain in the vicinity of building 14 to combat the emergency unless otherwise instructed:

   The Research Reactor Division Chief.
   The Supervisor and all Operational Staff of the section where the emergency occurred.
   The Reactor Health Physicist and all Radiation Surveyors on duty.
   The building 14 guard.
   The maintenance Supervisor.
   Any other staff whose absence would create another hazard.

3. All personnel remaining at the scene of the emergency shall wear respirators previously issued until notified by Health Physics that this is no longer necessary. Personnel who for any reason have not a respirator or who find their respirator is not functioning properly, will obtain a replacement from Health Physics.

4. If an emergency exists at a time when a change of shift is scheduled, the new shift may not be allowed to proceed to the reactor building, but should remain available at building 7 until directed by the Communication Center coordinator to proceed to building 14.

5. It shall be the responsibility of the Chief of the Research Reactor Division, in consultation in Health Physics, to determine when the emergency no longer exists and to advise the Communication Center coordinator who shall arrange to terminate the emergency. Any special precautions regarding the existance of residual contamination shall be issued by Health Physics before personnel are allowed to return to area so affected.

F. <u>Procedures for Notifying Quehanna Personnel and Surrounding Communities of an Impending Radiation Hazard.</u>

Although very elaborate control instrumentation and safety interlocks have been installed in the reactor system, it is always possible, although most highly improbable, that an emergency could arise whereby a portion of the reactor core could become atomized. In such an occurrence a cloud of fission products could form which might drift across the countryside towards an area of habitation. Dispersion of the cloud with the atmosphere is dependent on weather conditions at the time of the accident.

If such a situation occurs it is the responsibility of the Head of Health Physics to collect the following data immediately and to ensure that the required action is carried out.

1. From information available such as direction of travel of the cloud, weather conditions and an estimate of the amount of fuel atomized, to decide if it will be necessary to advise any of the surrounding communities such as Karthaus, Driftwood, Sinnemahoning, etc., of an impending radiation hazard, and to ensure that all Quehanna personnel, working in an area other than the reactor area, remain or proceed indoors where all windows and doors will be tightly closed and all supply and exhaust ventilation fans shut down.

2. If from the data collected, a hazard to any community cannot be ruled out, he will advise the Industrial Relations Manager of the Research Division giving him all necessary information. The Industrial Relations Manager will in turn notify the community which might be affected, advising them if they should:

    a. Vacate the community and in which direction, or

    b. Proceed or remain indoors with all windows and doors closed until notified that the emergency is terminated.

3. The radiation monitoring trailer shall proceed to the community nearest to the passage of the cloud, take radiation readings and continuously sample the air to determine the exposure, if any, to the general populace.

4. Following dispersion of the cloud, the amount and location of fallout shall be determined and adequate action taken to prevent exposure of personnel to this fallout. Particular attention shall be paid to water supplies, grazing land, and any areas inhabited by humans.

G. Emergency Drills

Emergency drills will be held at least four times per year and at intervals of no longer than four months. The Supervisor of Security and Safety shall be responsible for the initiation of such drills, and supervisory staff are to ensure that all their personnel participate.

Prior notice of an emergency drill may or may not be given.

APPENDIX VI -- SUMMARY OF HEAT TRANSFER CONSIDERATIONS

A. Introduction

Of prime consideration in evaluating the power levels attainable with both 10-, and 19-plate elements was the surface temperature of the fuel element. Unless this temperature is kept below the saturation temperature of the coolant water at the core depth, boiling will occur at the fuel plate surface. Such boiling is undesirable from the point of view of reactor control.

B. Procedures

1. The heat transfer coefficient was calculated using classic methods applied to heating fluids inside tubes (channels).

2. The average difference between the fuel plate surface temperature and bulk of the coolant was determined based upon 95% of the power being transferred from the fuel elements to the coolant.

3. Because the neutron flux varies across the core, and heat flux can be assumed to vary with neutron flux, a maximum-to-average neutron (or heat) flux ratio of 1.8 was applied to the temperature difference previously obtained.

4. The coolant temperature rise from core inlet to the point of maximum-average surface temperature was determined. The maximum-average surface temperature was considered to be about two-thirds of the length of the fuel element, in the direction of flow. This value was added to the temperature difference previously obtained.

5. A hot channel factor of 1.67 (resulting from MTR test) was applied to the temperature difference to accommodate fuel element fabrication tolerances, operational dimensional change, etc. This factor is considered to be conservative.

6. Finally, a factor of 1.25 was applied to the temperature difference adjusted for hot channel effects, to take into account the fact that at clean, cold startup the maximum-to-average axial flux with the rods partially inserted is about 25% greater than with the rods withdrawn.

C. Summary

Based upon a maximum flow rate of 1200 gpm, a core inlet temperature of

90°F, and a saturation temperature for water of 240°F at the core depth, graphs of maximum fuel element temperature vs. power level were drawn for 10-, and 19-plate elements.

For 10-plate elements with safety rods partially inserted, the fuel plate temperature will equal the saturation temperature at 1.7 megawatts. For 19-plate elements, the comparable power level limit will be 4.0 megawatts.

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

CURTISS-WRIGH'. .ORPORATION
RESEARCH DIVISION
QUEHANNA, PENNSYLVANIA

SHIPPING NOTICE

OUR ORDER NO.	AMENDMENT NO.	DEPARTMENT	SHIPPING ORDER DATE 5/1/59	NO. CA 59-135

CUSTOMER'S ORDER	OUR ENG WORK ORDER	OUR SHOP WORK ORDER
	15476	

MOVE-ORDER TO: M. Bowles

CONTROL ACCT.	DETAIL ACCT.

CHARGE TO

PKG.	GROSS	TARE	NET	DIMENSIONS	CUBE
~~1~~	~~14#12OZ.~~				
		12 OZ.	14#	12X10X6"	.41 C/F
~~2~~	~~3½#~~	~~½#~~	~~4#~~	~~11½X2X10"~~	~~.13 C/F~~

CONSIGNEE SAME - UNLESS OTHERWISE INDICATED BELOW

U. S. Atomic Energy Commission
Division of Licensing and Regulations
1901 Constitution Avenue
Washington 25, D. C.
Attention: Mr. Lyall Johnson

SHIPPED FROM QUEHANNA, PA.

VIA Best Way PARCEL POST SPECIAL HANDLING | B/L NO. | DATE SHIPPED 5-5-59 | F.O.B. Destination

QUANTITY ORDERED	QUANTITY SHIPPED	PART NO.	DESCRIPTION	
15	15		Copies of Hazard Report #4062	[redacted]

[redacted]

*RETURN RECEIPT REQUESTED

FORM CO-22

MAIL TO CONSIGNEE, AFTER SHIPMENT

2