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PRELIMINARY HAZARDS EVALUATION REPORT
CURTISS-WRIGHT RESEARCH REACTOR

AUGUST 24, 1956

trans. by 1tr. 10/24/56.

D-823

CURTISS WRIGHT

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

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PRELIMINARY HAZARDS EVALUATION REPORT

CURTISS-WRIGHT RESEARCH REACTOR

Prepared by

M. J. Abbate
P. R. Liller
C. J. Roberts
Physics Division
Nuclear Power Department

Approved by:

B. J. Mezger
Department Manager

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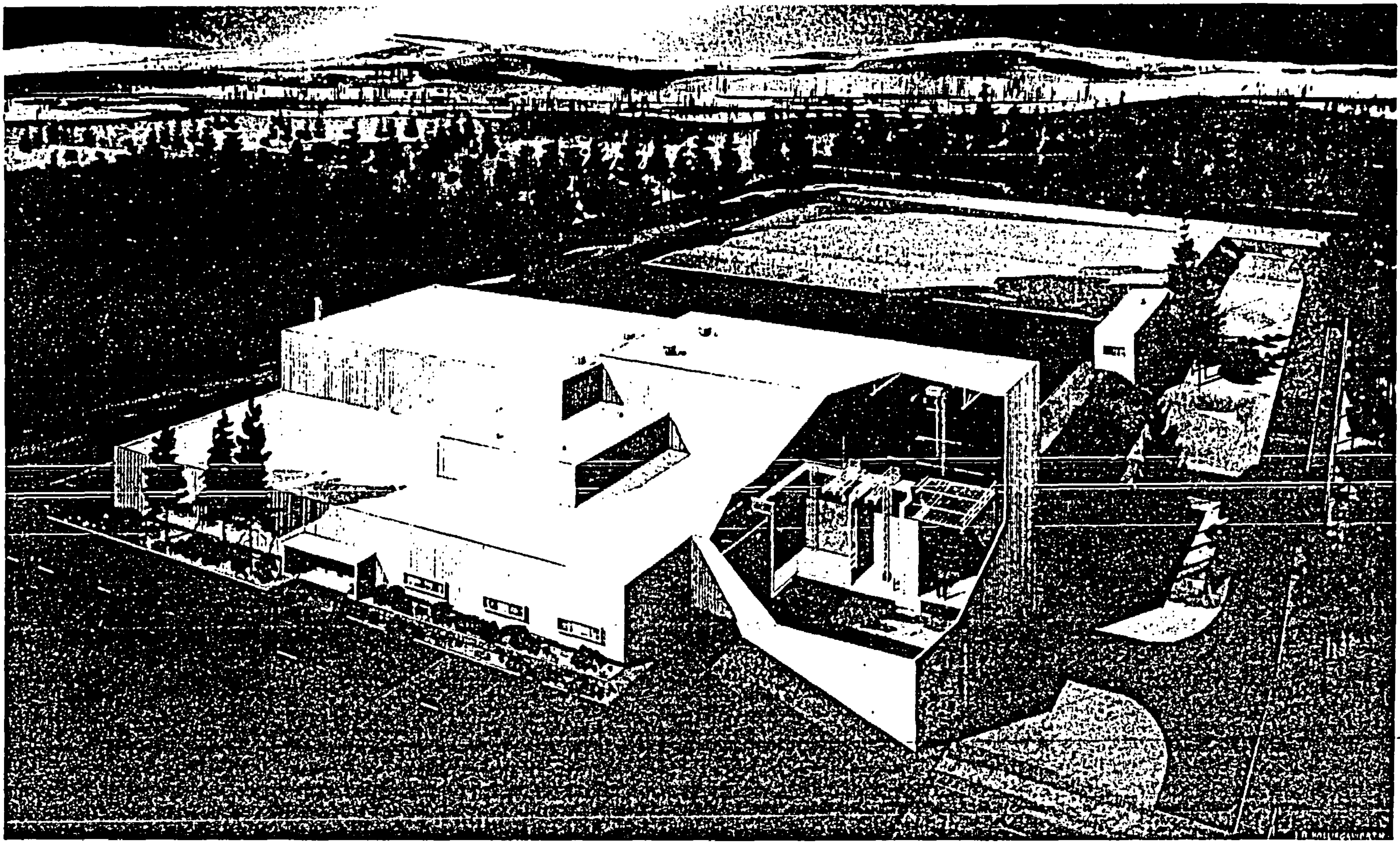


THE RALPH M. PARSONS COMPANY
ENGINEERS • CONSTRUCTORS
LOS ANGELES

FIG. I THE RADIOACTIVE MATERIALS LABORATORY

INTRODUCTION

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

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

The reactor will be 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 nuclear research facilities at this site will be the Radioactive Materials Laboratory shown in Figure 1, which will house the reactor under discussion. The Ralph M. Parsons Company, Los Angeles, is 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, on a very limited scale, training purposes.

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

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

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

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

I. THE REACTOR, ITS SUPPORTING FACILITIES AND SITE

A. Reactor

Some of the more important characteristics of the reactor are tabulated below.

TABLE 1

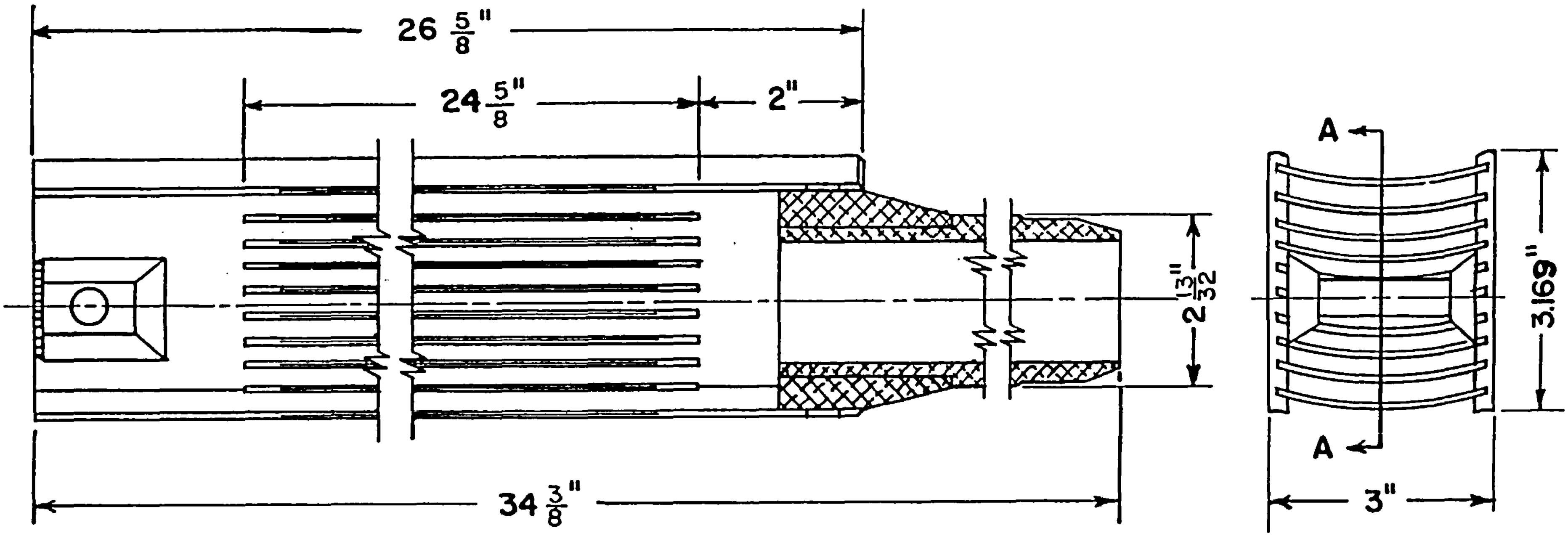
Characteristics of the Curtiss-Wright Research Reactor

Type	swimming pool (modified BSR-type)
Core	heterogeneous - uranium, aluminum, water
Al/H_2O volume ratio	0.31
Moderator	light water
Reflector	light water or beryllium oxide
Coolant	light water, free convection flow at 0 to 100 kw, forced circulation at 100 to 1000 kw
Biological shield	light water, normal and high density concrete
Critical mass	2.7 to 3.6 kg U^{235} depending on configuration (water moderated)
Power level	up to 1000 kw
Average thermal flux	8×10^{12} n/cm²/sec at 1000 kw

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

1. Fuel Elements. The reactor core will be made up of 18 or more modified MTR-type fuel elements containing a maximum of ten fuel bearing plates per element. Each plate is essentially a sandwich of aluminum-uranium alloy between two layers of aluminum cladding. The uranium will be enriched in the isotope 235 to $\sim 90\%$. The thickness of the aluminum cladding will be sufficient to contain all fission fragments under normal circumstances. In over-all dimensions, plates will be approximately 0.060 in. thick, 3 in. wide, and 24 in. long.

   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



SECTION A A OF FUEL ELEMENT ASSEMBLY

FUEL ELEMENT

FIG. 2 SKETCH OF PROPOSED STANDARD FUEL ASSEMBLY FOR CURTISS-WRIGHT RESEARCH REACTOR an element to about 36 in. This permits the end of the element to be inserted into a hole in the grid plate which supports the entire array. A rough sketch of a fuel element is shown in Figure 2.

A full element of ten fuel plates will contain about 170 gm of U-235. A number of partially loaded elements will also be on hand. These will be similar to the 170 gm variety except that a number of fuel bearing plates will be replaced by dummy plates containing only aluminum. Four elements will have three plates missing, and in their place will be a guide slot to accept the shim safety or control rods.

The proposed inventory of fuel elements is as follows:

TABLE 2

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

Type	Number to Be Procured	Gm U-235 Per Element	Total Gm U-235	Number of Fuel Bearing Plates Per Element
Full	20	170	3400	10
Partial				
80%	2	136	272	8
60%	2	102	204	6
40%	2	68	136	4
20%	2	34	68	2
Special (to receive control and safety rods)	4	119	476	7
Totals	32		4556	

Previous experience with this type of reactor core (Reference 1) places the minimum cold clean critical mass at 2.75 - 2.85 kg U-235, or, in other words, something between a 4 x 4 and a 4 x 5 configuration of full and partial fuel elements.

In many cases requirements for available reactivity to override xenon poisoning as well as experimental needs will push the critical mass to 3.4 - 3.6 kg.

Considerations of neutron economy may at times dictate the use of a BeO reflector. The BeO will be sealed in aluminum cans externally identical to the fuel elements. These can then be placed as desired in the grid plate.

CURTISS-WRIGHT RESEARCH REACTOR

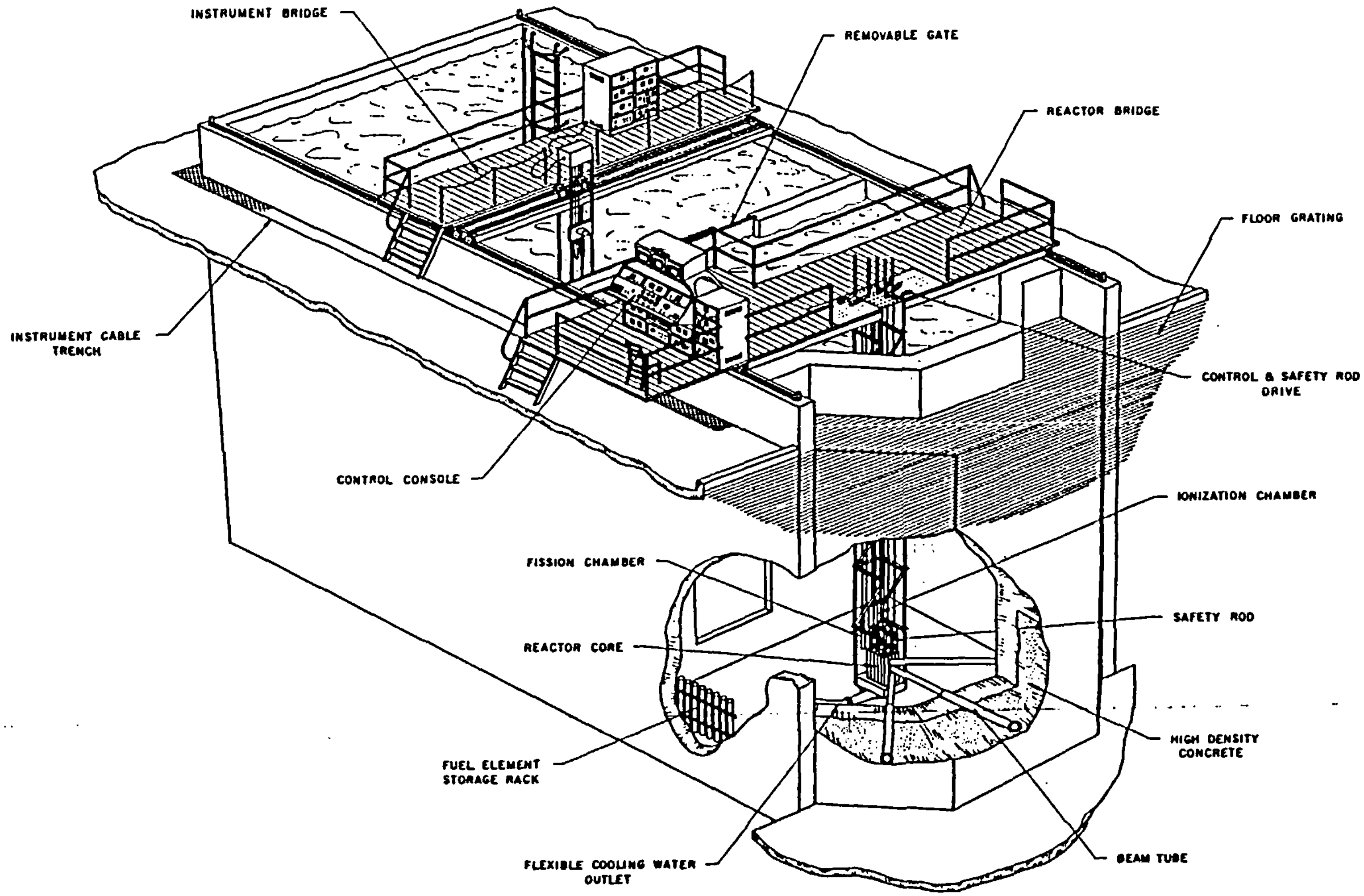


FIG. 3 DETAILED CUTAWAY OF POOL AND REACTOR

2. Supporting Structure. The fuel elements will be supported by a grid plate which is attached to a rigid aluminum tower suspended from the underside of the reactor bridge. The general arrangement is shown in Figure 3. The aluminum grid plate has 54 holes in a 9 x 6 array. Into these holes the mating part of the fuel elements will fit. A grid plate from a similar reactor is shown unloaded and partially loaded in Figure 4. Provision will be made in the grid plate for mounting an antimony-beryllium neutron source for start-up operations. Additional holes are provided in the grid plate to allow water to circulate around as well as through the fuel elements.

   The grid plate will be supported 4 ft above the pool floor by the aluminum tower. Neither plate nor tower will be capable of motion relative to the reactor bridge. The bridge will be designed to support a static load of 4000 lbs at its center, and will be cantilevered out over the operating floor at one end to support the control console and reactor operators. The bridge will be free to roll along rails mounted on top of the walls surrounding the pool so that the core may be positioned 4 ft above the pool floor at any point along the pool center line. The only constraint will be supplied by stops on the rails which will prevent the core being moved closer than 4 ft from the pool wall at either end.

3. Control Rods and Drives. The significant characteristics of the control and safety rods and drives are summarized below:

Type	Material	No. Used	Worth-reactivity H_2O Reflector	BeO	Rate of Drive in./min
Control	Stainless steel	1	0.6%	1.2%	24
Safety-shim	Boron carbide	3	2.6% ea	3.8% ea	6

   The three safety-shim rods are magnetically coupled to their drives. 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.

4. Control Console. The control console will be located on the cantilevered portion of the reactor bridge facing the reactor structure so the reactor operator can observe all work being performed. The console consists of two units. The main unit is approximately 62 in. long and 46 in. high with three miniature type recorders centered 7 in. above it in such a manner that the operator may see through the resultant 7 in. slot when seated, or may see over the recorders when standing. The prime consideration in the design of the main unit is centralization of the shim-safety rod switches,

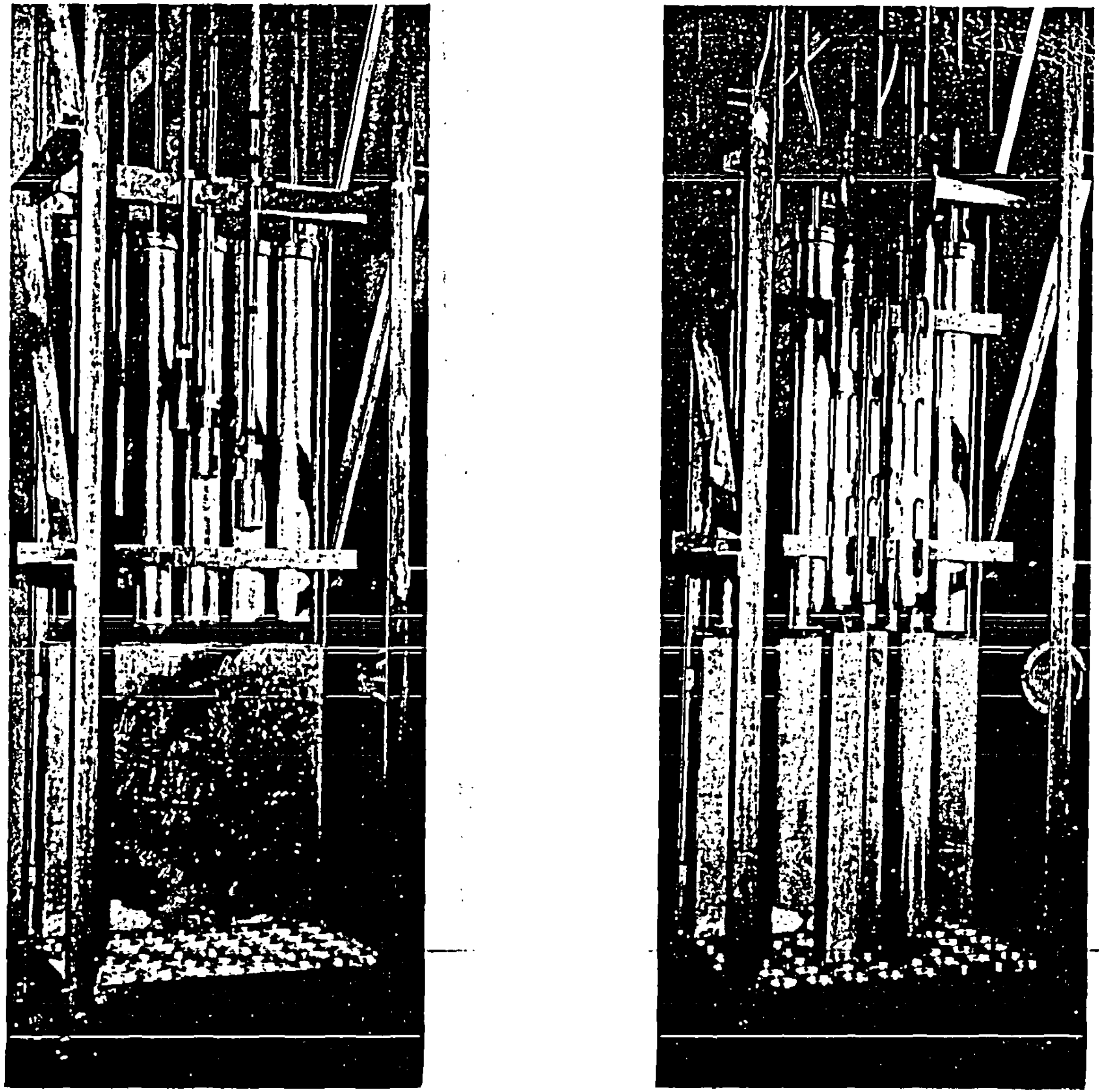


FIG. 4 VIEW OF GRID AND ACTIVE LATICE PARTIALLY LOADED. FROM THE BSR.

control rod switch, scram button, rod position indicators, count rate meter, flux level and period indicators so that entire control of the reactor will be 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 compensated ion chambers and associated equipment. The auxiliary unit is approximately 22 in. wide, and 46 in. high and will be mounted at the right of the main unit. It will be equipped with a radiation monitor, decade scaler with associated timer, and a test panel for checking the control console circuits and equipment.

5. Reactor Pool. The pool will measure 20 ft in width by 41 ft in overall length including the three-sided extension at one end for the beam tubes (see Figure 3). The water depth of 26 ft will insure a minimum of 19 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 experiments utilizing the beam holes. The reactor may be moved between the two sections through a vertically sliding, watertight aluminum gate, 5 ft x 22 ft long, located at the center of the concrete bulkhead. The larger pool's volume will be about 93,600 gal, while the smaller will contain 53,800 gal.

Pool walls will be ordinary reinforced concrete 12 in. thick, except at the beam hole end where the thickness is increased to 5 ft, 6 in. In order to further attenuate the radiation from the reactor core before it reaches the beam room, the outer 4 ft of concrete will contain ferrophosphorus aggregate, with a minimum density of 280 lb/cu ft. This high density insert will reach 9 ft 6 in. above the floor level in the beam room. The concrete sides and floor of the pool will receive several coats of the best available protective paint in order to prevent excessive leaching of minerals from the concrete into the demineralized water.

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 beam room. A large overhead door at one end of the bay will allow trailer-trucks to drive under the crane for removal of shipping casks and other heavy objects.

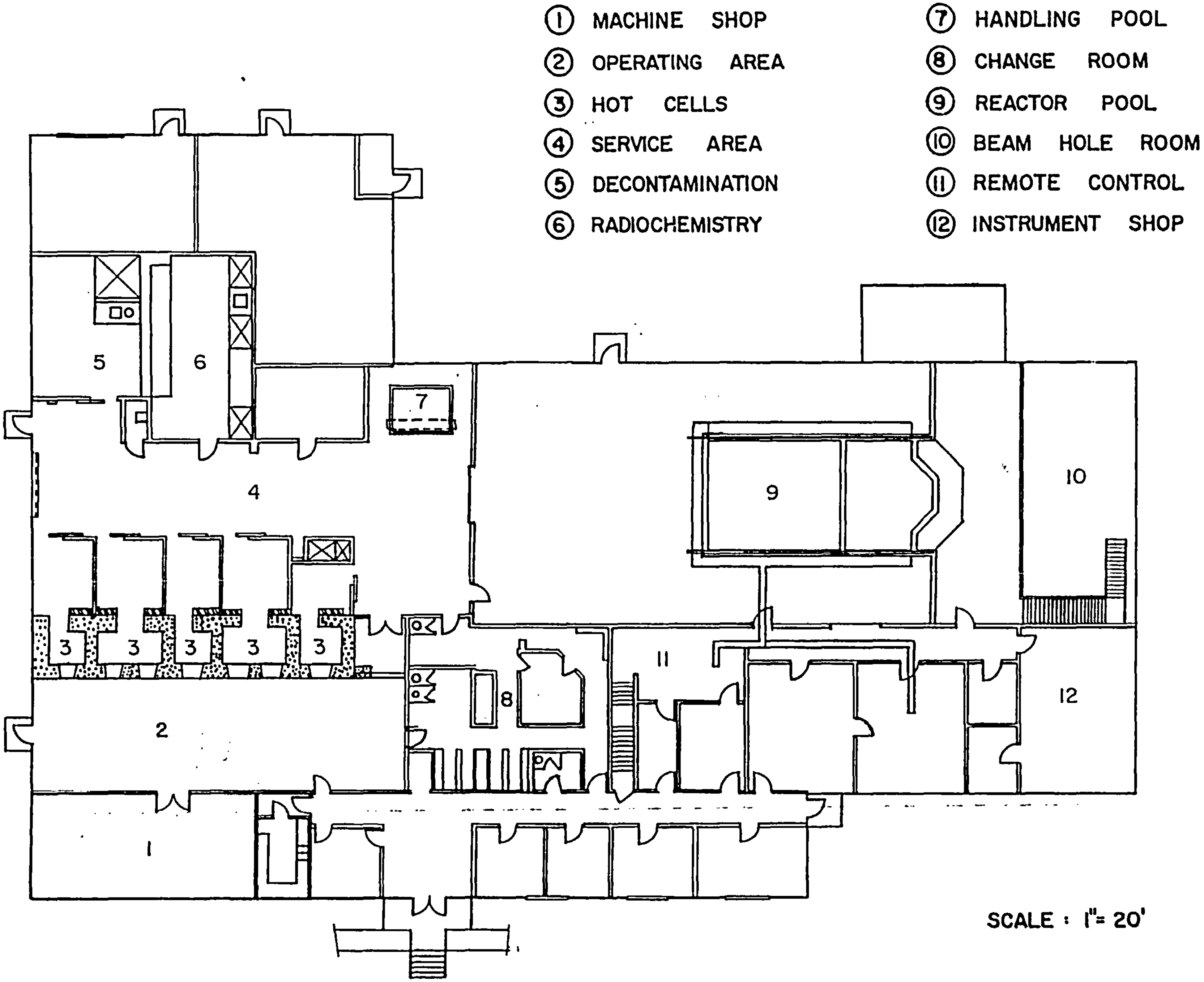


FIG. 5 FLOOR PLAN OF REACTOR AND HOT LABORATORY BUILDING

Heat and ventilation for the reactor bay is supplied by a single overhead space heater. This oil fired unit provides a minimum of 20% outside air and at least six air changes per hour. The exhaust from this unit will be continuously monitored for radioactivity. If the activity level exceeds a predetermined value, the blower will automatically be shut off, thus preventing the release of dangerous quantities of activity to the environment.

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

Opening from one side of the pool area is an additional 2800 sq ft of office and laboratory space. Adjoining the pool area at one end, but normally sealed off from it, is the service area of the hot lab. The relationship of the reactor bay to the rest of the building is shown in Figure 5. The change room is so situated that traffic to and from the reactor area will normally by-pass it completely. However, in the event of an accident causing gross contamination of the reactor area, this area can easily be sealed off so that the only access is by way of the change room.

2. Water Treatment and Cooling Systems

a. Demineralizer. In order to prevent excessive corrosion of the fuel elements and the buildup of significant levels of dissolved radioactive material the pool water will be demineralized prior to each filling. This will be accomplished by passing the water through a high capacity, regenerative type ion exchange bed before it enters the pool. The initial water treatment facility will be located in the mechanical equipment room at the rear of the hot lab service area. From here a 2 in. line runs to each section of the pool by way of a valve located in a valve pit at one side of the pool. Filling time will be less than 24 hours.

Once the pool has been filled, water quality will be maintained by recirculation at a rate of 15 gpm through a cartridge-type mixed bed ion exchanger. This system, as well as part of the cooling system, will be located in a relatively isolated equipment room at the side of the beam room. Water will leave the pool via a 1.5 in. line having its origin in a 2 ft deep sump in the pool floor. It then passes through the 15 gpm circulating pump and disposable de-ionizer. Return to the pool is via a 6 in. line shared in common with the cooling loop.

This line penetrates the pool wall 12 ft below the normal water level. A p^H of 6.5 and a specific resistance of 2×10^5 ohm-cm will be maintained by the recirculation system.

This system will also provide a means of decontaminating the pool water if it should become active. If this occurs, the reactor will be shut down and water passed through the cartridge resin bed until tolerable levels of activity are reached. The water can then be dumped in the usual fashion. The pump and ion exchange bed will be installed so that shielding can be erected, if necessary, and provision will be made to handle contaminated cartridges safely. Two disposable deionizers will be provided so that one is always operating while the other is undergoing cartridge replacement.

b. <u>Cooling System</u>. 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 the 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.

Water will be pumped through the core at 700 gpm and into an aluminum plenum chamber attached to the underside of the grid plate. Grid plate holes not containing fuel elements or moderator blocks 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 the hose in a short time. The flexible coupling will allow the core to be positioned as desired along its single axis of motion. The hose connects to a 6 in. pipe which rises to 18 ft above the pool floor before penetrating the wall.

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

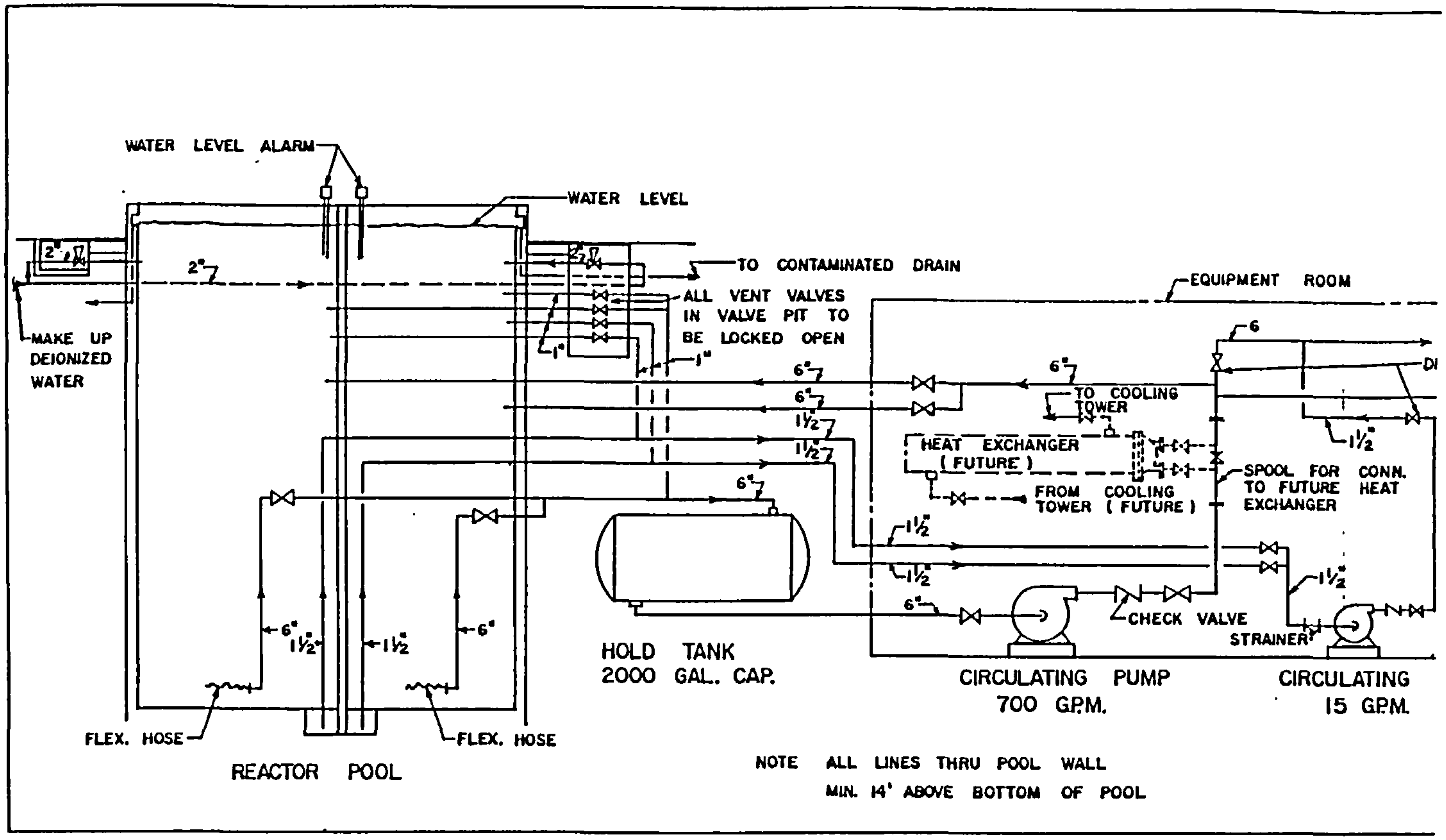


FIG. 6
REACTOR P

A large flapper valve on the bottom of the plenum will be counterweighted so that it will remain closed due to the pressure differential created by the flow of water through the fuel elements. When the coolant flow is reduced, this pressure differential decreases and the counter weight can then open the flapper valve, allowing free convective cooling. The opening of this valve and the drop in coolant flow rate will each initiate a reactor scram signal. Either of these two safety devices would prevent the reactor core from overheating in the event of a coolant stoppage.

Before emptying, samples of pool water will be taken and the radioactivity carefully measured. After assuring that the activity level is below one tenth the accepted MPL's, the water will be dumped. To accomplish this, the anti-siphoning vents must be unlocked and closed, and the dump valves unlocked and opened.

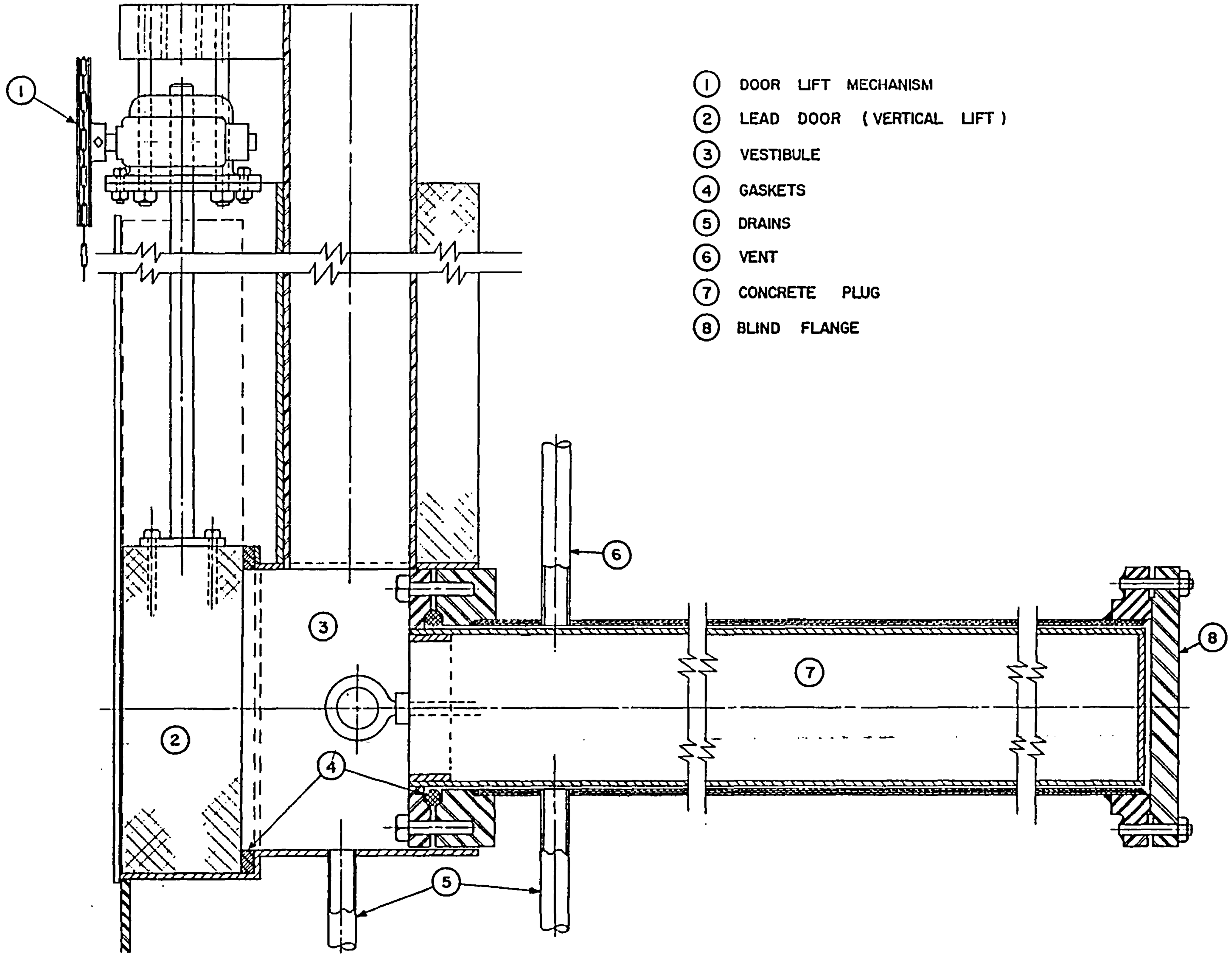
A schematic layout of the piping for these systems is shown in Figure 6.

3. Beam Tubes. As shown in Figure 3, one end of the pool will be penetrated by three beam tubes. These tubes originate a fraction of an inch from the reactor face and terminate at the far side of the dense concrete shield which forms one wall of the beam room.

The projections of the ends of the beam tubes into the reactor would fall one above the other along the vertical center line of the reactor face. This configuration gives relatively high flux in all three tubes while minimizing their worth in terms of reactivity. This point is discussed further in Section III. A flux of 3-5 x 10^8 n/sec/cm^2 is expected at the exit aperture of the beam tube at 1000 kw (Reference 1).

The construction of the beam hole with the concrete plug in place is sketched in Figure 7.

To use the beam hole the pool would be drained, the concrete plug and sealing flange at the pool face removed, and the aluminum beam tube inserted. The watertight seal between tube and pool wall will be similar to that shown between the plug and the pool wall. As an additional safety feature the outer watertight door covering the opening of the beam tube into the beam room will normally be kept closed. Necessary connecting lines will be fed into the tube via the 6 in. conduit which originates above the pool water level. The drain lines from the beam holes will discharge into the contaminated waste system.



① DOOR LIFT MECHANISM
② LEAD DOOR (VERTICAL LIFT)
③ VESTIBULE
④ GASKETS
⑤ DRAINS
⑥ VENT
⑦ CONCRETE PLUG
⑧ BLIND FLANGE

FIG. 7 BEAM TUBE SECTION

4. Radioactive Waste Handling Facilities. As indicated above, in the event of an accident the pool water will be decontaminated by recirculation through the cartridge type deionizer.

Any low level liquid wastes other than pool water will be handled by the facility which serves the hot lab portion of the building. This facility will consist of two drain systems, one for suspect waste and one for low level contaminated waste. Each drain system empties to one of two 3000 gal retention tanks buried outside the building. After sampling, the contents of these tanks are dumped if below MPL's, or pumped to the waste disposal building. This is a small structure some 50 ft from the main laboratory and houses the pumps and an evaporator for the system. Here any contaminated aqueous waste is evaporated and the distillate is discharged to the sewer after monitoring. Drains in the reactor area will be connected to the suspect half of the system.

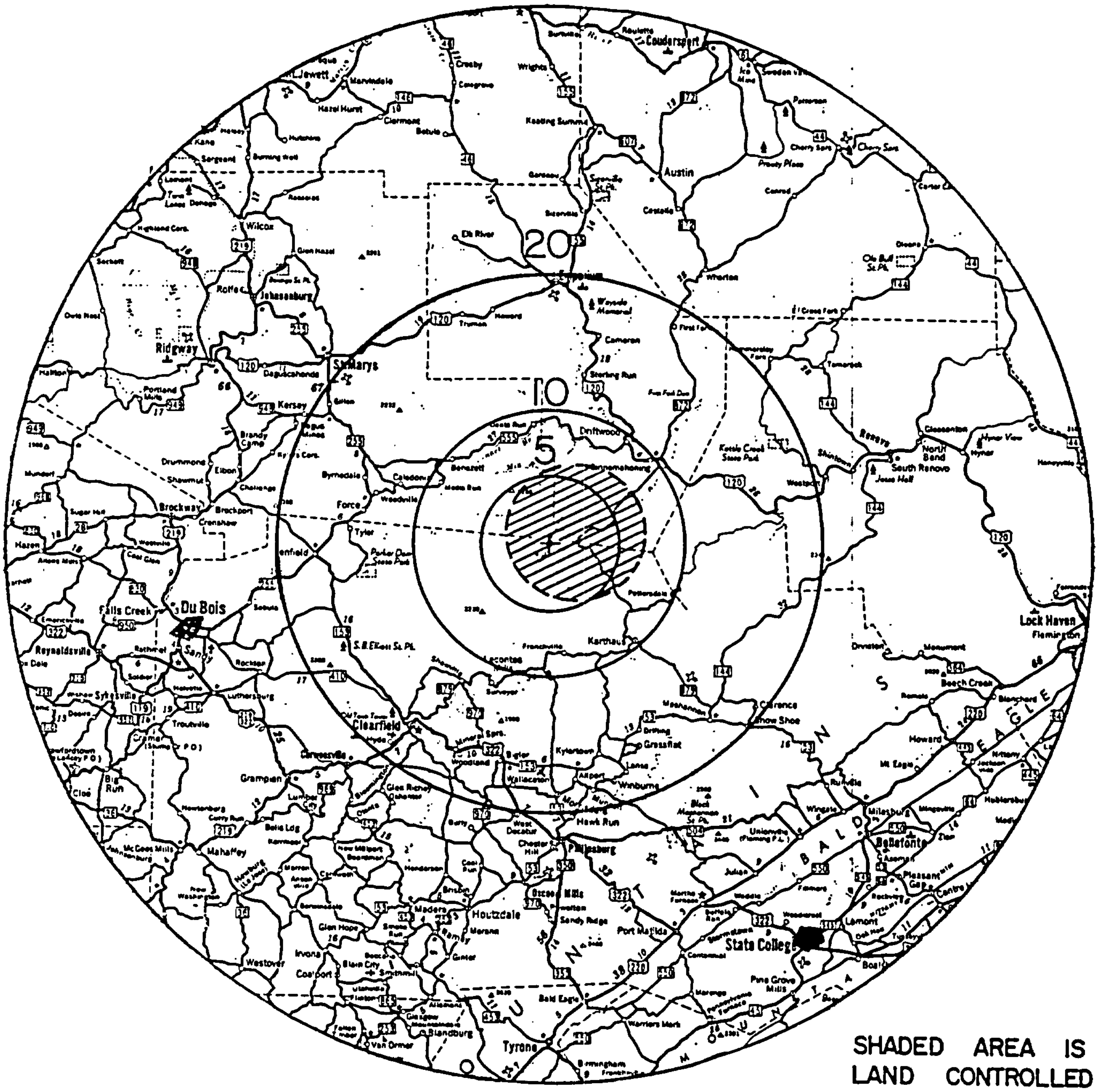
Solid wastes will be segregated as produced, according to activity level, and the containers stored at the waste disposal building until removed from the site. This will be the fate of contaminated ion exchange resin and residue from the evaporator. Special laundry facilities will be available in the waste disposal building to handle any clothing which becomes contaminated in the course of reactor operations.

C. Site

The Radioactive Materials Laboratory with its research reactor will be located on an 80 sq mi 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 diameter, lies in North Central Pennsylvania (see Figure 8). It encompasses portions of Elk, Cameron, and Clearfield Counties.

As Figure 9 indicates, the reactor itself will be located in the southwestern quadrant of the circle, a minimum of 3 mi 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.

40 MILES FROM REACTOR



SHADED AREA IS LAND CONTROLLED BY CURTISS-WRIGHT

FIG. 8 LOCATION OF REACTOR RELATIVE TO ENVIRONS

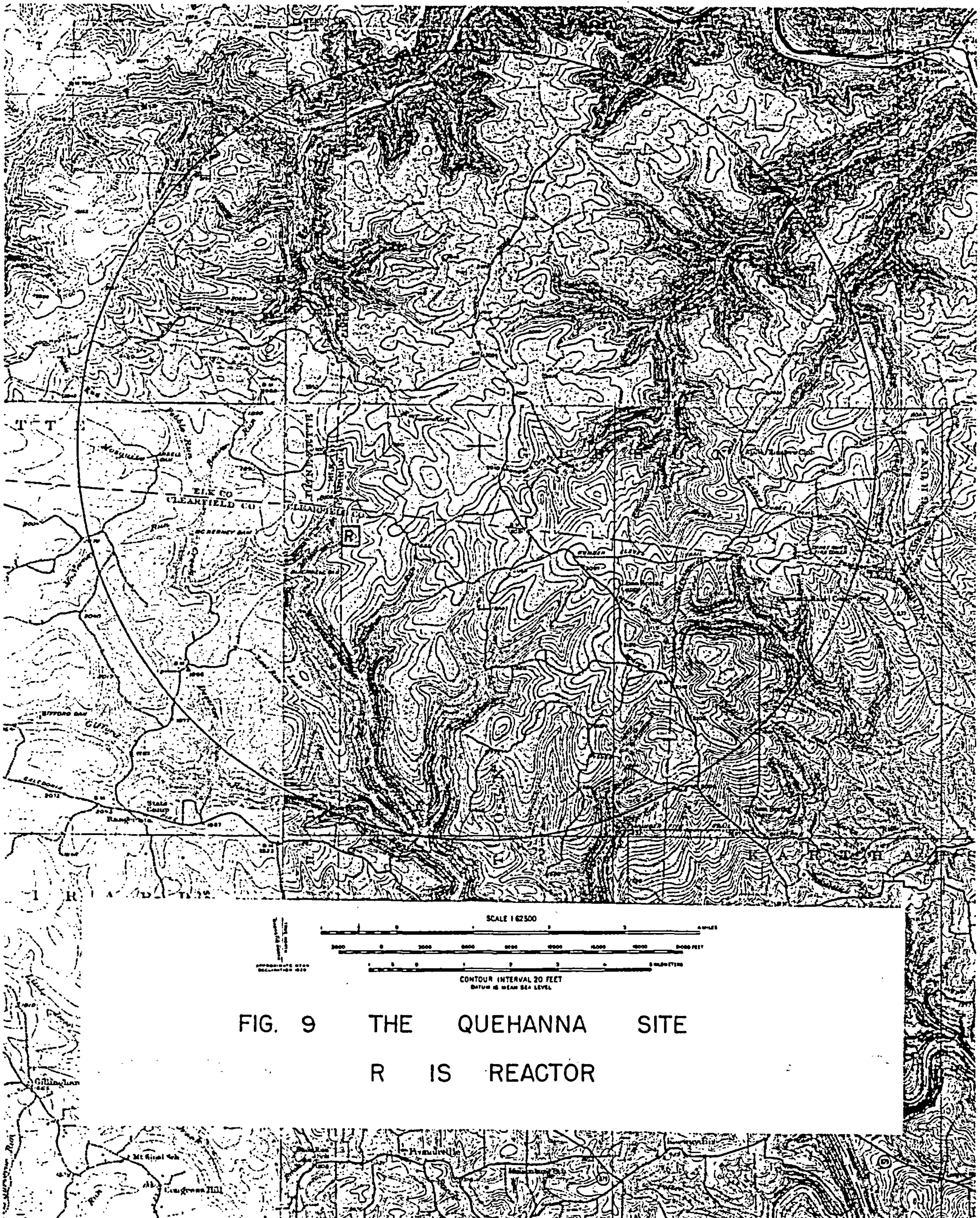


FIG. 9 THE QUEHANNA SITE

R IS REACTOR

TABLE 3

Towns Nearest Reactor Site*

	Population		Population
Bald Hill	250	Kylertown	500
Benezette	200	Lanse	350
Bloomington	200	Lescontes Mills	600
Blue Ball	900	Morrisdale	655
Brockport	415	Moshannon	500
Byrnedale	500	Munson	450
Caledonia	300	Oak Grove	300
Chester Hill	954	Oshanter	300
Clarence	530	Orviston	400
Clearfield	9,357	Penfield	600
Crenshaw	550	Philipsburg	3,988
Curwensville	3,332	Quail	600
Driftwood	289	Renovo	3,751
Drockton	250	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	Weedsville	600
Karthaus	575	Westport	201
Keewagdon	200	Woodland	650
		Winburne	700

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*From Rand McNally Commercial Atlas and Marketing Guide, Eighty-sixth Edition, 1955.

The reactor will be 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 as Appendix I. It appears that in the area of the reactor there is a layer of coarse-grained sandstone up to 200 ft or so in thickness. The underlying shale formation is probably impervious to fluid flow and is not fractured by folding. The penetration of surface water is therefore quite limited and lateral flow would soon return ground water to the streams in the area. The drainage of effluent from the reactor site into local ground waters is considered further in Section IV-B.

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

The countryside surrounding the site is largely uninhabited. The closest towns of any appreciable size will be 10 mi from the reactor. Table 3 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/sq mi.

II. INSURING SAFE OPERATION OF THE REACTOR

A. Reactor Control and Safety Systems

Three safety-shim rods and one control rod will be used in controlling the reactor. Each of the three safety rods, made of boron carbide, will provide a $\Delta k/k = 2.6\%$ for a light water reflector and a $\Delta k/k = 3.8\%$ for a BeO reflector. The safety rods will be magnetically coupled to their respective drive mechanisms allowing the rods to fall freely, during a scram, with an acceleration approaching that of gravity. The maximum rate of withdrawal of the safety rods will be 6 in. per minute. At their most effective position, about 50% withdrawn, the speed would correspond to a rate of change of reactivity for any one rod of about 0.021% per second for a water reflector and 0.031% per second for a BeO reflector. The safety rods may be withdrawn individually, or any combination of the three rods may be withdrawn simultaneously as desired.

Since the migration length for light water is so small (6.4 cm) the reactivity effect introduced by a given safety rod in the anticipated lattice configurations should be nearly independent of the position of the other safety rods. This relative independence has been indicated by rather inconclusive experiments performed in other reactors of similar construction. Thus, the maximum rate of change of reactivity for all three rods being moved simultaneously is roughly equal to three times the value for one rod: 0.063% per second for a water reflector and 0.093% per second for a BeO reflector.

One stainless steel control rod will be used in the control system. This rod will provide a $\Delta k/k = 0.7\%$ for a light water reflector and a $\Delta k/k = 1.2\%$ for a BeO reflector. The control rod will be mechanically coupled to its drive mechanism which will have a maximum withdrawal rate of 24 in. per minute. In its most effective position, the maximum rate of change of reactivity of the control rod will be 0.016% per second for a water reflector and 0.026% per second for a BeO reflector.

The position of all rods will be continuously indicated to within ± 0.05 in. by a synchro system transmitting position data to the reactor control console. Micro-switches, mounted on the drive mechanisms, will be actuated when the rods are in certain positions. This information will be indicated on a bank of lights on the console. The positions indicated by these lights will include upper and lower limits for all rods, shim range for safety rods, and most effective control/range for the control rod.

A Leeds and Northrup type PAT 60 Servo Amplifier System will be used to automatically control the power level at the desired operating value. This system will be inoperative until the power level is greater than 90% of the desired operating value.

Four gamma compensated ionization chambers and a U-235 lined fission counter will serve as the flux level detecting devices. A block diagram of these units and their associated circuits is shown in Figure 10.

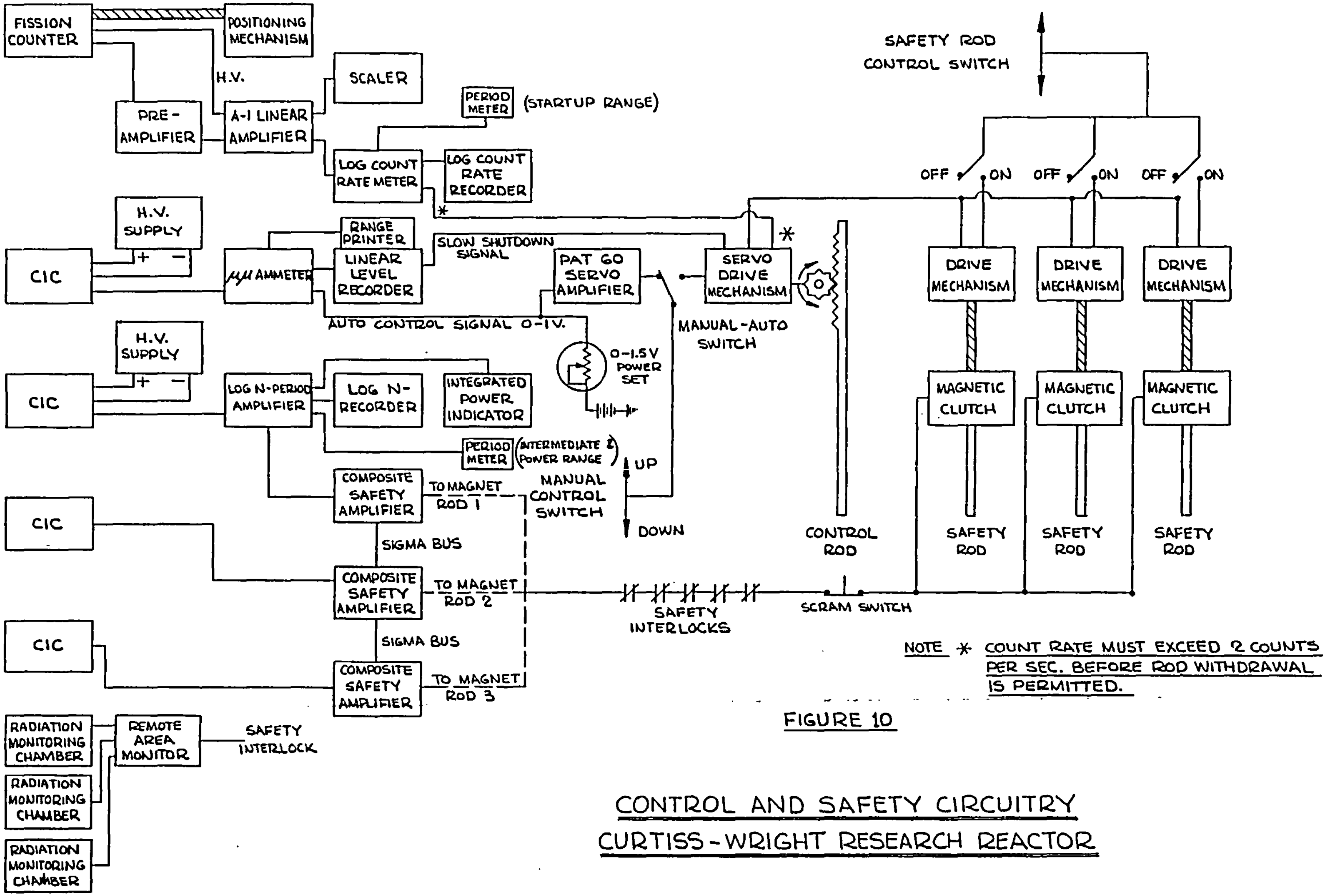
Two types of automatic shutdown, "slow shutdown" and "scram", provide protection against nuclear incidents. The slow shutdown action drives all control and safety rods to their lower limit at full speed. A milliammeter relay, actuated by the micromicroammeter, will initiate the signal for this action whenever the flux level exceeds 120% of the desired operating value. The slow shutdown feature is designed to be the first line of defense against a relatively slow rate of increase in flux level above the desired operating value. Additional interlocks in the safety circuit, as described in Table 4, will also initiate the slow shutdown action.

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. The scram will be reserved for only the most serious situations, namely, flux level exceeding 140% of desired operating value and reactor period less than 5 sec. A hazardous condition warranting a scram will be sensed by any one of three ionization chambers -- two sensing linear level and connected directly to the safety amplifiers, and the remaining one connected to the Log N system which furnishes a period signal to the safety amplifiers.

Portable air monitoring equipment will also be set up periodically in the reactor bay and beam room to insure that airborne activity from some unsuspected source does not build up.

In addition to the gamma radiation monitors which will be interlocked with the reactor safety system, there will be detectors in the beam hole room, equipment room, and reactor bay which will indicate radiation levels on a central control station.

A beta sensitive detector will constantly monitor the activity in the recirculation loop through the ion exchanger, and another similar unit will record the build up of activity on the resin



NOTE * COUNT RATE MUST EXCEED 2 COUNTS PER SEC. BEFORE ROD WITHDRAWAL IS PERMITTED.

FIGURE 10

CONTROL AND SAFETY CIRCUITRY
CURTISS-WRIGHT RESEARCH REACTOR

TABLE 4 -- Reactor Safety System

Note: The occurrence of any of the situations listed below will be indicated by an annunciator with the exception of those marked with an asterisk.

Situation	Detector	Unit Initiating Action	Resulting Action
140% neutron flux	Chamber	Safety amplifier	Scram
5 sec period	Chamber	Safety amplifier	Scram
120% neutron flux	Chamber	Sensitrol relay	Slow shutdown
Coolant water flow interrupted	Switch	Plenum flapper	Slow shutdown
Coolant water flow rate low	Pressure Switch	Flow monitor	Slow shutdown
Pool water level down 18 in.	Switch	Float level indicator	Slow shutdown
High radiation level	Ionization chamber	Radiation monitors	Slow shutdown
Coolant activity excessive	Scintillation counter	Coolant activity monitor	Slow shutdown
Reactor bridge moves during reactor operation	Switch	- - -	Slow shutdown
Safety rods withdrawn less than 60% of full travel*	Switches	Servo drive mechanism	Prevents withdrawal of control rod
Startup count rate less than 2 counts/sec*	Switch	Log count rate recorder	Prevents withdrawal of safety rods
Operator detects any dangerous situation	Operator	Scram button	Scram
Heat exchanger effluent high temperature	Thermocouple	- - -	None
Safety amplifier indication	Safety amplifier	- - -	None
Low level period - 30 sec	Fission counter	- - -	None
Safety circuit bypassed	Relay	- - -	None

itself. A meter will record the flow rate through this loop. The performance of the ion exchanger will be monitored by continually recording the specific resistance of the pool water. This instrumentation should be sufficient to insure that the cartridges are renewed as necessary and, after some operating experience has been gained, to allow the time of renewal to be predicted accurately.

The coolant loop will be monitored by temperature sensing elements following the pool outlet and the heat exchanger. The secondary coolant loop between exchanger and cooling tower will also be monitored for flow rate and temperature. An air monitoring device mounted in the exhaust stream of the reactor bay ventilating system will automatically shut off the system and prevent spread of airborne activity to the environment.

B. Administrative Controls

The likelihood of an accident involving a nuclear reactor can be greatly reduced by the clear definition of responsibility for the various phases of operation. The prime responsibility for the safe operation of the reactor will, therefore, be assigned to one man, the head of research reactor operations. Directly responsible to him will be the shift supervisors and the operators. The reactor operations group will be one section of the Nuclear Operations Division.

In order to prevent any compromise of safety, the responsibility for experimental work will be completely divorced from considerations of operating procedures. The Chief of the Physics Division will be charged with supervision of the experimental work alone and will only be able to make requests regarding reactor operation. As a further check on the research program, all proposed potentially hazardous experiments will be referred to a Curtiss-Wright Reactor Safeguards Committee for review.

The qualifications and duties of the individuals and committee mentioned above will be as follows:

1. Head of Reactor Operations. The head of reactor operations shall be an individual possessing a good theoretical knowledge of reactor operations and extensive practical experience. His responsibilities shall be as follows:

   Supervise bringing the reactor to criticality following any change in fuel loading or any significant change in nearby experimental equipment or specimens. In such cases, the approach to criticality will be by means of a critical experiment. The only exception will be when the change in loading has produced a configuration previously logged. In this case the critical experiment can be omitted.

Supervise the experimental program involving the reactor itself; i.e., calibration of rods, determination of temperature coefficient, determination of flux distributions and power calibration, etc.

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

Supervise all operating personnel.

Schedule and coordinate experimental programs.

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

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

Issue regular monthly reports on reactor operation to the Curtiss-Wright Reactor Safeguards Committee, Chief of Physics Division, Chief of Nuclear Operators Division, and other interested parties.

Issue special reports and recommendations for corrective measures to Curtiss-Wright Reactor Safeguards Committee following any significant malfunction, violation, or accident.

Appoint a competent alternate when not available for more than one day.

2. <u>Chief of the Physics Division</u>. The Chief of the Physics Division shall be an individual possessing a good theoretical knowledge of reactor operations and some practical experience. Moreover, he shall have knowledge of the over-all effort of the Corporation, its goals, and promising lines of attack. His responsibilities shall be as follows:

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. His approval must be obtained before any experiment affecting reactor operation in any way may be performed.

Advise the originator of a request for reactor time concerning the reasons for disapproval.

Notify the head of reactor operations concerning requests which have been approved and relative priority of programs.

Pass along all approved requests for reactor time to the Curtiss-Wright Reactor Safeguards Committee, unless the proposed experiment clearly introduces no potential hazard whatsoever. If the sole potential hazard lies in the production of radioactive material, the approved request may bypass the committee and go directly to Health Physics for source control.

Appoint a competent alternate when not available for an extended period.

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

Review all requests for reactor time which are forwarded by the Physics Division Chief following his approval. This review shall encompass only matters concerning health and safety, and shall not touch upon the technical feasibility or advisability.

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

Review any special reports issued by the head of reactor operations following any significant malfunctions, violation, or accident, and recommend the corrective action to be taken.

Keep informed concerning reactor operations by studying the monthly reports issued by the head of reactor operations.

4. Reactor Supervisor. A supervisor shall be experienced in the operation of the reactor, be thoroughly familiar with all its components and their function, have a good understanding of reactor theory as applied to this particular reactor and hold an AEC operator's license. The duties of a supervisor shall be as follows:

Accept responsibility for the safe operation of the reactor at all times during his shift except when relieved by the head of reactor operations.

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

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 <u>any</u> movement of <u>any</u> object in that portion of the pool in which the reactor is operating at the time.

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

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

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

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

5. <u>Reactor Operator</u>. A reactor operator shall be a person holding a valid AEC operator's license. His duties shall include:

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

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

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

Keeping all such records and logs as shall be required.

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

C. <u>Operational Controls</u>

The complete understanding of and compliance with a well conceived set of operating 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.

1. General.

No experimental work involving the reactor will be carried out under duress of a time limit.

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 supervisor is present at the console with him.

Loading or unloading of the active lattice, or movement of the reactor bridge may only be done under the direct supervision of the director of reactor operations. This will be enforced by keeping the bridge and fuel element handling tools normally locked in place with a key in his possession.

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 supervision of the director of reactor operations.

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

Once a configuration has been loaded and made critical, the reactor may be shut down and started up again under direction of a supervisor, provided an operator is also present in the reactor bay.

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

A licensed operator must be present at the console at all times when the reactor is operating.

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

No one but the director of reactor operations may authorize the bypassing of a safety interlock.

2. Critical Experiment. As specified above, whenever the active lattice is to be loaded in a previously untried configuration, or a large instrument or sample is to be placed adjacent to the core, the procedure of loading by increments and determining the sub-critical multiplication curve will be followed under the supervision of the director of reactor operations.

The experimental procedure may be outlined briefly as follows. The initial loading shall not be more than 50% of the calculated critical mass or a configuration previously shown to be subcritical. Fuel will be added in increments of one fuel element. During loading, two safety rods will remain three-quarters withdrawn at all times. After each addition of fuel, the count rate will be determined with the remaining two safety rods fully inserted and three-quarters withdrawn. The subcritical multiplication curve will be plotted as data are obtained to keep a continuous check on the approach to criticality.

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. Before start-up the operator will go through a checklist which will give definite assurance of proper functioning of all systems. When the checklist has been completed, it will be attached to the appropriate page in the log book. The checklist will assure that no safety interlocks are defeated.

4. Start-Up (Cold, Clean). After announcing intention to start up reactor, the operator will raise safety rods at least 60% of their full travel, and preferably 75% if reactivity requirements permit. While withdrawing rods he will watch count rate and period meters closely to detect the approach to criticality. (If the reactor appears to be going critical with the control rod still inserted, he will shut down at once and notify the director of reactor operations.)

The operator will raise control rod to one-half its full travel (thus putting it in its most effective range), while watching count rate and period meters to detect approach to criticality.

Safety rods will be raised intermittently until reactor goes critical and power level begins to increase with desired period (never less than 20 sec). Count rate recorder will be turned off when it goes off scale.

When desired power is reached, the operator will insert safety rods until power stabilizes, and will energize the servo-system.

Fission chamber will be withdrawn until register operates about once a second. Power level will be announced over PA system, which may be switched, if desired, so register may be heard throughout building.

Date, time, experiment, and run number will be noted on Log N chart.

The data specified in the detailed operating manual will be recorded hourly.

5. Start Up (High Residual Power Level). Under conditions of high residual power level start-up should only be attempted with the utmost caution. Between increments of rod withdrawal, adequate time must be allowed so that equilibrium conditions can be observed.

6. Shutdown. Normal shutdown will be accomplished by driving the rods down, not by dropping them (scram).

Before leaving the console, the operator will turn the key switch to the "off" position and return it to the director of reactor operations. He will also check to see that the reactor bridge and fuel element handling tool are properly locked in place.

D. Shielding from External Radiation Hazards

The points of closest approach normally accessible to operating personnel are the reactor bridge and the beam hole room. A minimum of 19 ft of water will cover the core, and therefore represents the minimum shield for an individual standing at the center of the reactor bridge. According to data from the BSR (Reference 2), the dose rate due to the direct penetration of gamma rays from the core at 1000 kw will be about 7 mr/hr at the water surface. The neutron flux will be entirely negligible.

Continual recirculation of the pool water through a mixed-bed ion exchanger will maintain the concentration of dissolved substances far below that which could be a problem from the standpoint of external radiation hazard. The water in the recirculation loop will be continually monitored and activity buildup would be detected long before it was an external hazard. The continual measurements of water conductivity will serve as an additional check on purity, and hence induced activity.

The 7-sec N^{16} activity induced by fast neutron irradiation of the water itself 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 the 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 from the core, significant quantities of N^{16} rise to 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 down through the core from top to bottom and out of the pool to a hold-up tank. The details of the cooling system have been described previously. If the flow of cooling water drops significantly, a flapper valve on the plenum beneath the reactor will open, and the reactor will be automatically shut down. If this device should fail and N^{16} begins to rise to the surface in large quantity, the gamma sensitive monitron on the reactor bridge would shut down the reactor. The 2000 gal hold-up tank will provide sufficient time for essentially all of the N^{16} to decay. Since it is buried it presents no external radiation hazard.

Because of the 3 min transient time between the core and the equipment room, there will be no gamma activity in this room due to the N^{16}. A small amount of activity will accumulate on the ion exchangers through which the pool water is continuously circulated at 15 gpm. However, it is not anticipated that this will normally become excessive and, in any event, it will be limited since replacement of the cartridges will be necessary because of exhaustion of the ion exchange capability of the resin. Nevertheless, the background in the equipment room will be continually checked by a remote monitron, and also by daily monitoring by a health physicist.

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

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

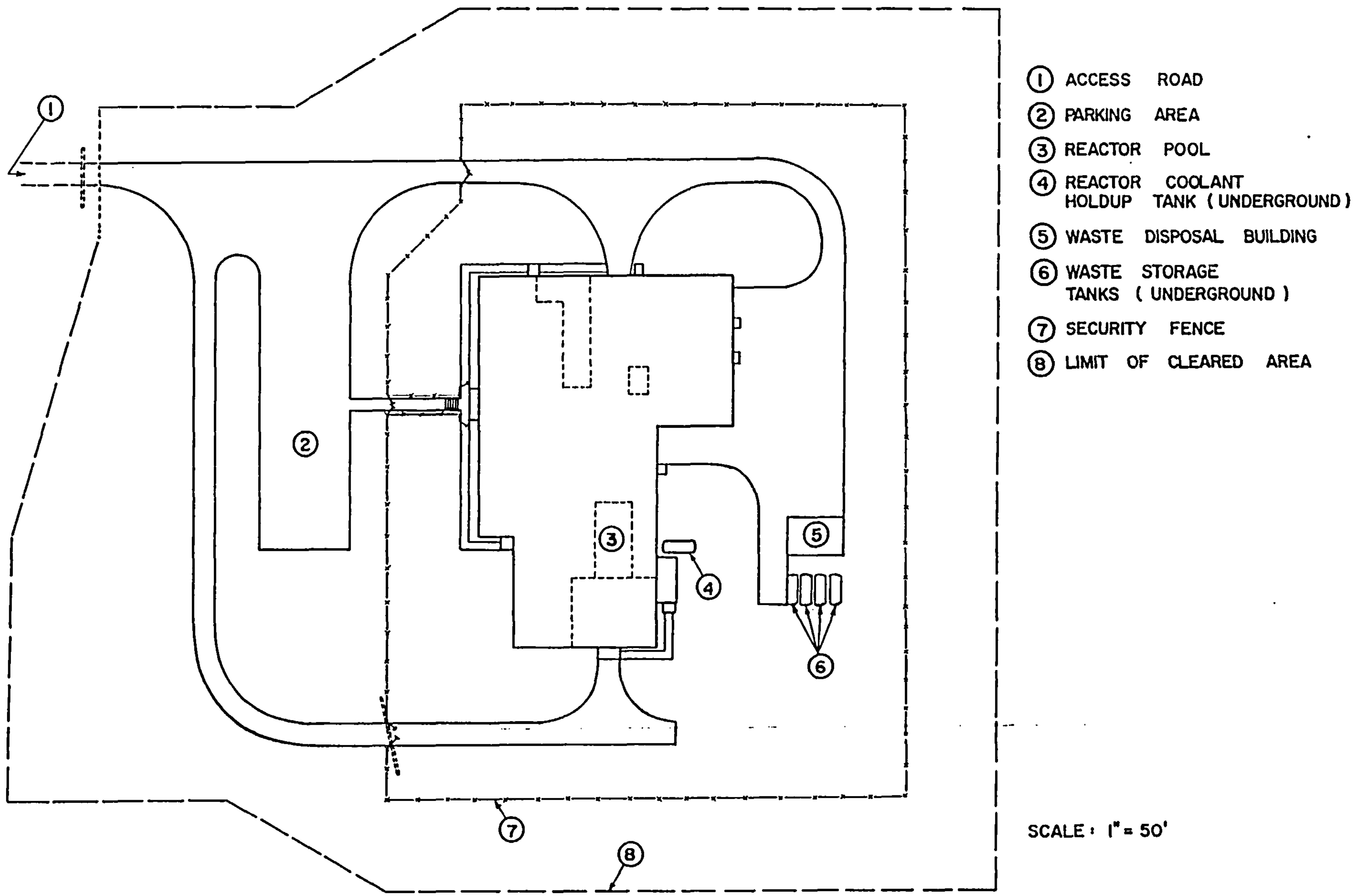


FIG. 11 REACTOR BUILDING SITE PLAN

addition, radiation levels will be continually monitored by several detectors strategically located in the beam room.

E. 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 5 mi to reach the reactor building.

The Radioactive Materials Laboratory, which houses the reactor, and the Waste Disposal Building are enclosed by a chain link fence topped by three strands of barbed wire reaching to the height of 9 ft. The layout of the buildings, fences, and approaches are shown in Figure 11. 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 employees or properly authorized visitors holding appropriate security clearances to enter the building unescorted. All other individuals must be under constant surveillance by a cleared escort. Entrance to the building by another door will normally 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 every hour 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 will consist of overhead water sprinklers in the office areas, charge room, and equipment rooms. All other areas will be liberally supplied with hand and cart type CO_2 extinguishers. In addition, all areas will be supplied with automatic rate-of-rise detectors connected to annunciators in the building itself and at the communications center which serves all facilities at the site.

In the event of a serious fire which cannot be handled by equipment available at the laboratory, the mobile fire fighting equipment will be called in. Two fire trucks and full time firemen will be located about five miles from the building and will be available 24 hours per day.

Additional 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 has, as a major duty, the prevention and management of forest fires. Mobile equipment and a well trained staff will be available at all times. In addition, the State will supply man power and equipment as needed. A meteorological program at the site will indicate the likelihood of a fire condition and will aid in planning emergency measures. The reactor building will be surrounded by a cleared area several hundred feet wide which will serve as a fire break. With the excellent water supply available to hose down the building, and the extremely fire resistant exterior construction, it is probable that a severe fire in the vicinity would do essentially no damage.

F. Fuel Management

Upon receipt from the manufacturer, new fuel elements will be stored in a safe configuration in the classified document vault of the main Research Division laboratory. When the reactor is ready to begin operations they will be transferred as needed to the reactor pool for loading into the core. Tampering or possible theft will be prevented by the security measures restricting entrance to the building and by keeping the handling tools locked up when not actually in use. The best deterent against theft once a few kilowatt-hours are logged will be the residual activity of the fuel elements themselves.

According to the operational time-table as it is now envisioned, the reactor should go critical about the middle of 1957, reach a power level of 1000 kw early in 1958, and begin routine operation for extended periods at 100 kw early in 1959. It is anticipated that burnup of 5-10% can be achieved. The actual limit which is set will be determined by experimental requirements and the rate of corrosion of the fuel plates. If this time-table works out, the first replacement of fuel elements will have to occur sometime around mid-1959.

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

Burned out elements will be transferred to semi-permanent storage in the pool racks or to a 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. They will then be loaded under water into a suitable shipping container and sent to a reprocessing plant.

G. Emergency Procedures

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

A communications center is to be set up at the Quehanna site some 5 mi from the reactor. This center will act as a command post for directing emergency operations. It will be in communication with the reactor building by telephone (two lines) and, if these lines should be destroyed, by radio. Any incident in the laboratory will be announced over the building loudspeakers or by coded warning bells. Following such an announcement, the communications center would be notified of the incident by telephone or radio. The communications center would then notify health physics, plant protection, and other organizations which would act quickly according to predetermined plans.

Every attempt will be made to ascertain quickly the extent of any release of activity to the environment. Vehicles equipped with detection equipment will be available at the reactor site and at the communications center. Under stable atmospheric conditions in which significant off-site exposures could occur, there would be ample time to warn people in the cloud's trajectory, which almost always would 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 at 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 which will be continuously 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 quickly be taken through state health authorities. Since all streams are now chemically contaminated, all water for drinking purposes would necessarily have to be treated. Therefore, intake of water from the radioactivity contaminated stream could easily be prevented for the short time necessary.

H. Health Physics

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

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

The chief health physics officer reports directly to the manager of the Nuclear Power Department and has authority to shut down any operation which has not satisfied the standards of radiation protection established by the Curtiss-Wright health physics group. These standards will be at least as conservative as those published by the AEC.

III. ACCIDENTS INVOLVING THE REACTOR

A. Reactor 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. Perhaps the first question to be answered is, therefore, how much reactivity would ever be available for rapid addition. Table 5 lists the reactivity requirements for extended operation at 100 and 1000 kw.

TABLE 5
Estimated Reactivity Requirements

Source	Reactivity Required at 100 kw	Reactivity Required at 1000 kw
Negative temperature coefficient	.0006	.001
Equilibrium poisons (Xe, Sm, etc.)	.018	.040
Xe override	.000	.006*
Burnup (100 days)	.0005	.005*
Adequate rate of change of power level	.003	.003
Addition of smallest increment of reactivity available	.003	.003
Totals	.025	.053

*only one indicated in total

As indicated by Table 5 no allowance has been made for experimental reactivity requirements. This is because any experiment requiring more than 0.1% $\Delta k/k$ will be set up with the core unloaded. The reactor will then be brought to the desired power by means of a critical experiment. Therefore, there need be no built-in-reactivity to compensate for the introduction of large experiments.

The maximum reactivity requirements for prolonged operation at 100 kw are 2.5%. During the first day of operation this would be reduced to less than 1.4%. As noted below this is a very modest requirement in terms of the damage which would be expected to result from the sudden introduction of this amount of reactivity into the core.

The maximum reactivity requirements for continual operation at 1000 kw are significantly greater, amounting to 5.3% with the stipulation that provision will not be made simultaneously for Xe override and burnup. Although the figure is high in terms of the magnitude of the excursion it represents, it is felt that the risk involved is well within acceptable limits. It will probably be very seldom that extended operation at 1000 kw is necessary, so that a loading with 5.3% reactivity will be rare. Moreover, within less than 12 hours the xenon buildup will reduce the reactivity to under 4.0%. Although experience has shown that an excursion of this magnitude would destroy the reactor, the resulting incident could hardly be termed catastrophic. After 500 hours of operation the available reactivity would be less than 1.6%, a value which has been shown to cause only negligible damage to a reactor of this type.

2. Results to Be Expected from Excursions of Various Magnitudes. Having considered the maximum reactivity available for introduction into the core at any time, the next problem is to estimate the effects of such an event. The results of the 1953-1954 Borax experiments (Reference 3) and the more recent Spert tests (Reference 4) appear to be quite conclusive in this respect. These experiments, carried out with solid fuel water moderated reactors, indicate that almost immediately after being made supercritical this type reactor will become subcritical through expulsion of the moderator due to steam formation.

The fuel elements used in the Borax experiments were essentially the same as those proposed for the Curtiss-Wright Research Reactor. Repeated experiments were performed in which up to 2% k_{eff} was introduced in stepwise fashion without significant damage to the reactor core or components. This was done with the water moderator at saturation temperature. It was found that the severity of the excursion was considerably greater with the moderator subcooled, i.e., below saturation temperature. Therefore experiments were limited to periods of greater than 0.013 sec for the subcooled case.

The Borax tests produced relatively little data for the subcooled case of interest to designers of swimming pool reactors. Luckow and Widdoes (Reference 5) have used an empirical method of extrapolating the available Borax data to subcooled conditions. Their correlative method predicts the melting of the hottest fuel plate upon the introduction of 1.8% reactivity at 80°F. Comparison of the predicted and measured temperature excursions made at 120°F moderator temperature during the 1954 Borax tests show that predicted values average within 20% of the measured temperatures.

This subject will be treated in considerably more detail when the results of recent Spert tests become available. The expanded discussion will be included in the final hazards evaluation report.

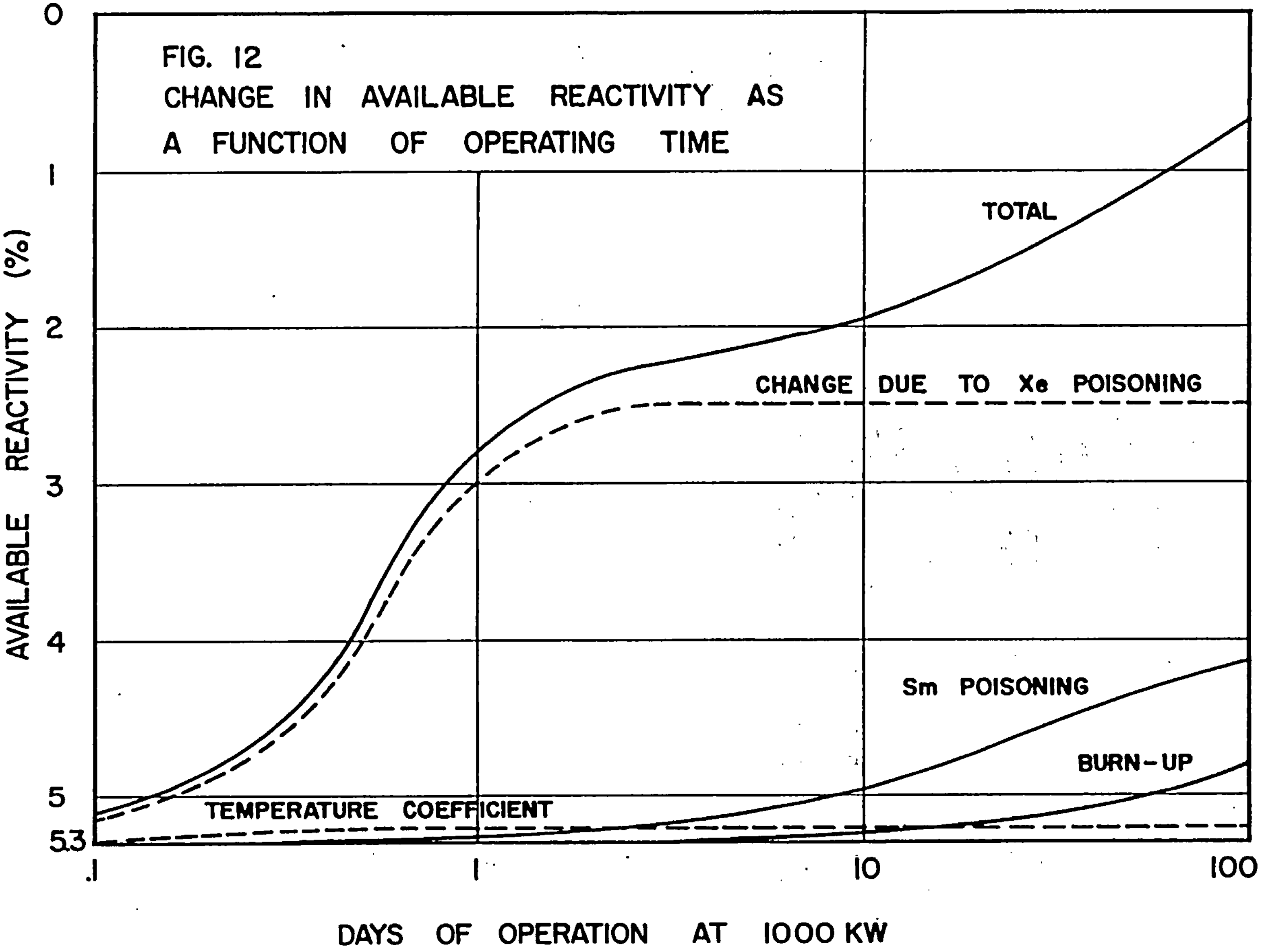


FIG. 12
CHANGE IN AVAILABLE REACTIVITY AS A FUNCTION OF OPERATING TIME

B. Chemical Reactions

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

Inasmuch as almost all the reports on this subject contain restricted security information they will not be discussed here. However, it may be said that in the Borax experiments (Reference 6), 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 by causing the cloud to rise, would help reduce the radiation dose at ground level.

IV. RESULTS TO BE EXPECTED FROM RELEASE OF RADIOACTIVE MATERIAL TO ENVIRONMENT

A. Radiation Hazards Due to Release of Radioactive Material to Atmosphere

The following is an attempt to consider the radiation dose to the population surrounding the reactor site in the event of a nuclear incident. Exposure would result from the inhalation of fission products in the air and from the external radiation from the passing cloud and from deposition on the ground. Without specifying the cause of the incident, the maximum possible accident will first be assumed, and then modifying factors which would be present in a credible accident will be considered.

If it were possible, the maximum accident would instaneously release all the fission products from a reactor that had operated long enough so that an equilibrium quantity of these fission products were present. These products would be formed either as a vapor or as fine particles, and the size of the cloud would initially be quite small. This cloud would not rise but would proceed downwind at ground level diffusing according to the meteorological conditions prevailing. It would be assumed that conditions are stable and, therefore, diffusion is poor. The observer at the point of interest would be standing at the center of the cloud and would remain there until the cloud passes.

In reality only a fraction of the activity would escape the core; it would initially have a large volume; it would rise because of its heat content and many of the released particles would not leave the immediate vicinity of the reactor.

1. Meteorological Parameters. In the calculations below, the diffusion of the cloud bearing the fission products is based on Sutton's equations. The meteorological parameters required to estimate diffusion are the wind speed μ, the diffusion coefficients C_x, C_y, C_z and the stability parameter n.

   Although a meteorological program is planned for the Quehanna site, there is no data presently available either for the site or for the surrounding area. Consequently, the values to be used are estimates based on data for other sites. D. H. Pack, of the U. S. Weather Bureau, (Reference 7), believes that they may be applied to Quehanna and agrees that they are clearly pessimistic. Because of the crudeness of the data, an isotropic diffusion C, will be used in place of C_x, C_y, C_z. Three conditions will be treated:

I. Deep inversion conditions that may exist at night time.

II. Daytime inversion conditions.

III. Average daytime conditions.

The following values of μ, C^2 and n will be used:

	I	II	III
μ	1	3	5
n	0.50	0.33	0.25
C^2	0.004	0.01	0.033

It need hardly be pointed out that the use of Sutton's equations and the above parameters are only crude approximations. Most meteorologic research has been carried out on level grassland or uniform terrain and even then the diffusion equations have yet to be validated for large distances from the origin. The Quehanna site is neither level nor uniform and the topography will be crucial in determining the micrometeorology. With low wind speed, for example, channeling in the valley will determine wind direction. Diffusion in the valley will be difficult to predict but certainly will not be great. Smith and Singer (Reference 8) have pointed out that during an inversion a cloud originating in the valley would be confined there and that this site offers the unique possibility of separating, meteorologically, two or more nuclear operations by locating them in different valleys. One is faced with the alternative of locating a potential emitter on the crests of hills where released activity will be dispersed with low concentration or locating in a valley where activity is confined with high concentration. The latter situation makes use of the valley for other purposes impossible.

2. Maximum Activity Available.

a. Mixed Fission Products. The β and γ energy is expressed by

$$F.P._{\gamma} = 2\times10^{17}\,t^{-0.2} \quad , \quad F.P._{\beta} = 2.2\times10^{17}\ \frac{\text{Mev/sec}}{\text{Mw}}$$

where t is the time in seconds after shutdown and the reactor is considered to have operated long enough for equilibrium to be established. Taking the maximum operating power as 1 Mw and assuming an average β-energy of 0.4 Mev and an average γ energy of 0.7 Mev, then maximum activity becomes:

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

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

b. Specific Isotopes.

$$Q \; \underline{\text{curies}} = \frac{\gamma \text{ atoms/fis} \times 3.12(10^{16}) \; \frac{\text{fis/sec}}{\text{Mw}}}{\frac{3.7(10^{10})}{\lambda} \text{ atoms/curie}}$$

γ = fission yield of isotope of interest
λ = decay constant of isotope of interest
t_o = operating time of reactor

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

Assuming a maximum fuel burnup of 10% and that the core contains 3.6 kilo of uranium-235, then

$$\frac{360 \text{ gm(burnup)}}{235 \text{ gm/mole}} \; 6.023(10^{23}) \frac{\text{atoms}}{\text{mole}} = 9.24(10^{23}) \text{ atoms U-235 fissioned}$$

If continuous operation at 1 Mw is considered, the time required for 10% burnup is

$$3.12(10^{16}) \frac{\text{fis/sec}}{\text{Mw}} \times t_o \text{ sec} = 9.24(10^{23}) \text{ fissions}$$

$$t_o = 2.96(10^7) \text{ sec}$$

$$= 342 \text{ days}$$

An operating time of one year is used for the following calculations of activity of the significant isotope to be considered for inhalation dose.

Isotope	γ	$Q_{1\text{ yr}}$	λ day^{-1}
Sr-90 + Y-90	.053(2)*	2460	7.6(10^{-5})
Sr-89	.046	3.82(10^4)	1.3(10^{-2})
I-131	.028	2.35(10^4)	8.64(10^{-2})
Y-91	.054	4.54(10^4)	1.21(10^{-2})
Ba-140 + La-140	.061(2)*	10.24(10^4)	5.4(10^{-2})
Ce-144 + Pr-144	.053(2)*	5.33(10^4)	2.5(10^{-3})

*multiplied by 2 to consider both isotopes

3. Inhalation Dose. An observer in the path of the cloud will take radioactive material into his body through inhalation. Generally the material will concentrate in a particular portion of the body. The inhalation dose from a particular isotope to the cirtical organ can be expressed as

$$D = \frac{1.16(10^4)\,Q}{C^2 \mu\, \chi^{2-n}} \; \frac{Ef}{M\lambda} \left[1 - \varepsilon^{-\lambda t}\right]$$

where D = dose in rep to the critical organ
M = mass in gm of the critical organ
f = fraction of isotope reaching the critical organ
λ = the effective decay constant of the isotope in the critical organ (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)
μ = wind speed m/sec
m = stability parameter
C = diffusion

The derivation of this equation is shown in Appendix III.

Figure 13 gives the inhalation dose for the six most important isotopes, and the total dose for the five bone seekers as a function of distance, for weather conditions I. Figures 14 and 15 show the dose-distance relation for weather conditions II and III respectively.

In calculating the total dose resulting from internal radiation the total bone dose will be neglected and only the thyroid dose is considered. This is justified since the whole body dose is of interest, and doses to different parts of the body are not additive. Only the most damaging exposure will be considered. The total dose (i.e., the dose delivered from time 0 to ∞) for the bone seekers is about 50% greater than the I-131 thyroid dose. However, the half life of I-131 is much shorter and consequently the biological damage is greater.

Figure 16 illustrates the point with data from the Radiological Health Handbook (Reference 9). It is seen that the lower the dose rate, the less the damage for a given total dose; e.g., if a given dose is spread over 16 days instead of one day, the damage is only half as great. If we calculate the internal dose for the first day after the incident for Case I at 10^4 meters, it is seen that the I-131 dose is 106 rep as compared to about 13 rep for the bone seekers.

Figure 17 shows the distribution of the thyroid dose with time expressed as percent of the total dose delivered at time $= \infty$. The effective dose of I-131 is about one-half of its total dose.

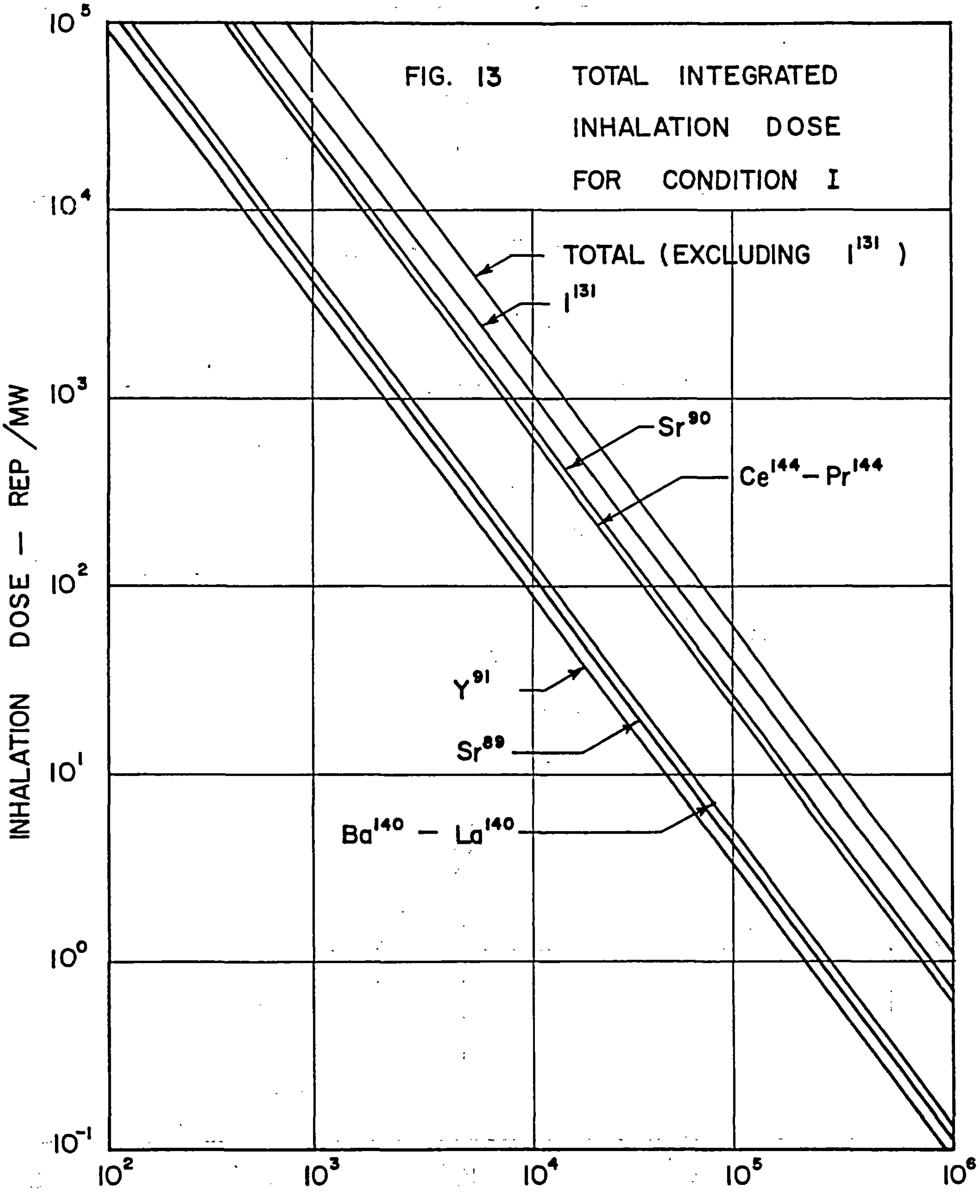


FIG. 13 TOTAL INTEGRATED INHALATION DOSE FOR CONDITION I
TOTAL (EXCLUDING I^{131})
I^{131}
Sr^{90}
$Ce^{144}-Pr^{144}$
Y^{91}
Sr^{89}
$Ba^{140}-La^{140}$
INHALATION DOSE - REP /MW
10^{5}
10^{4}
10^{3}
10^{2}
10^{1}
10^{0}
10^{-1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}
DISTANCE FROM INCIDENT - METERS

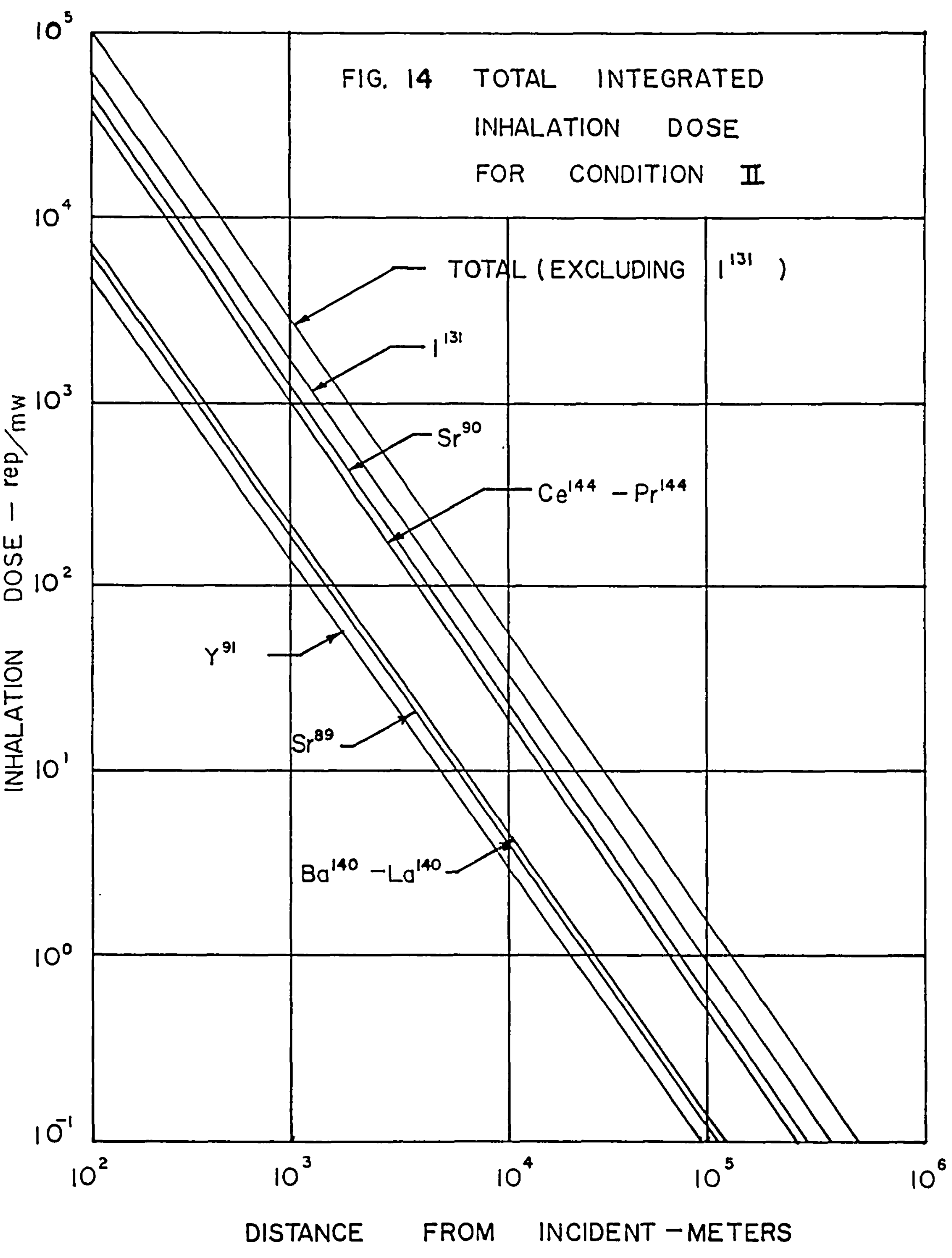


FIG. 14 TOTAL INTEGRATED INHALATION DOSE FOR CONDITION II

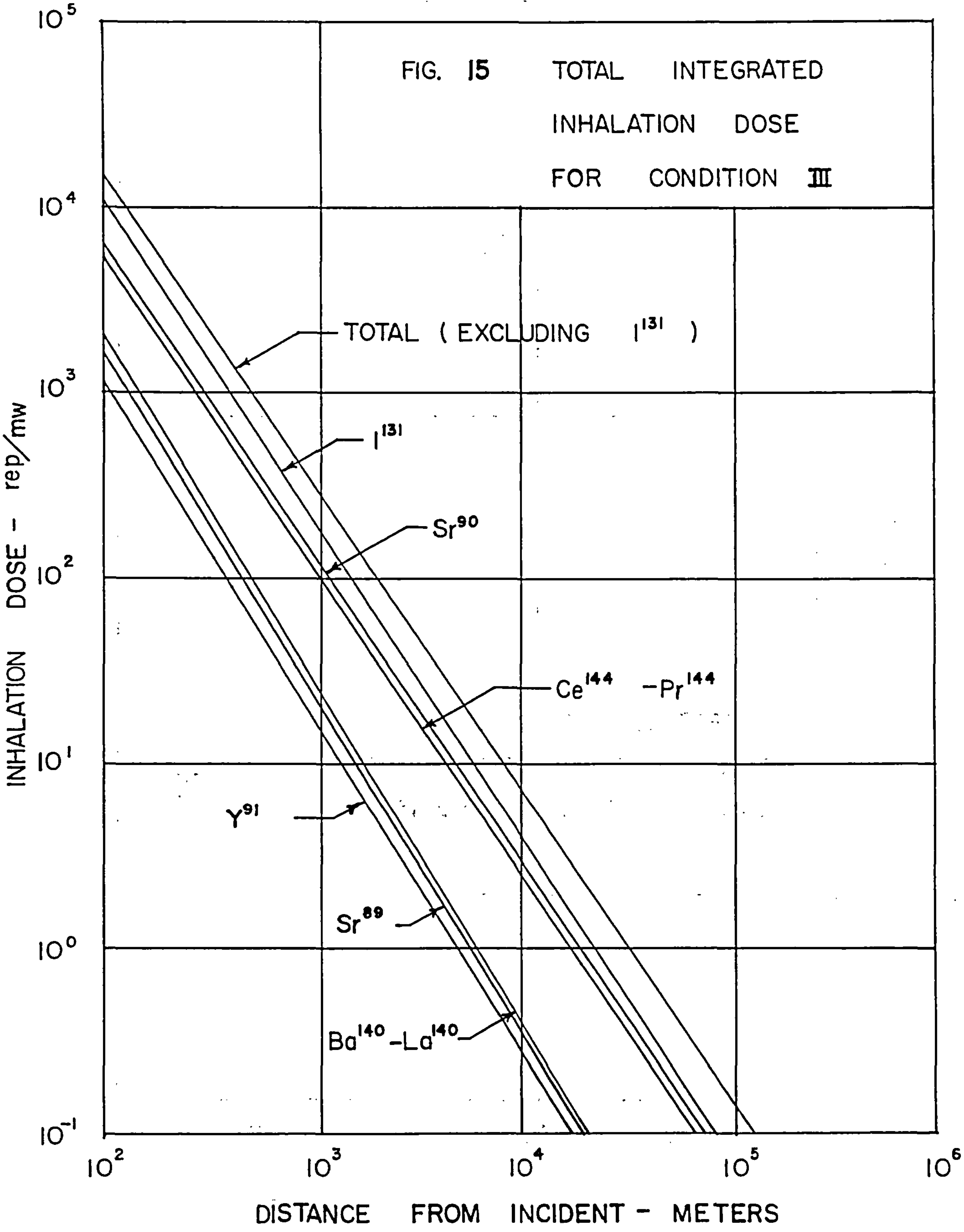


FIG. 15 TOTAL INTEGRATED INHALATION DOSE FOR CONDITION III

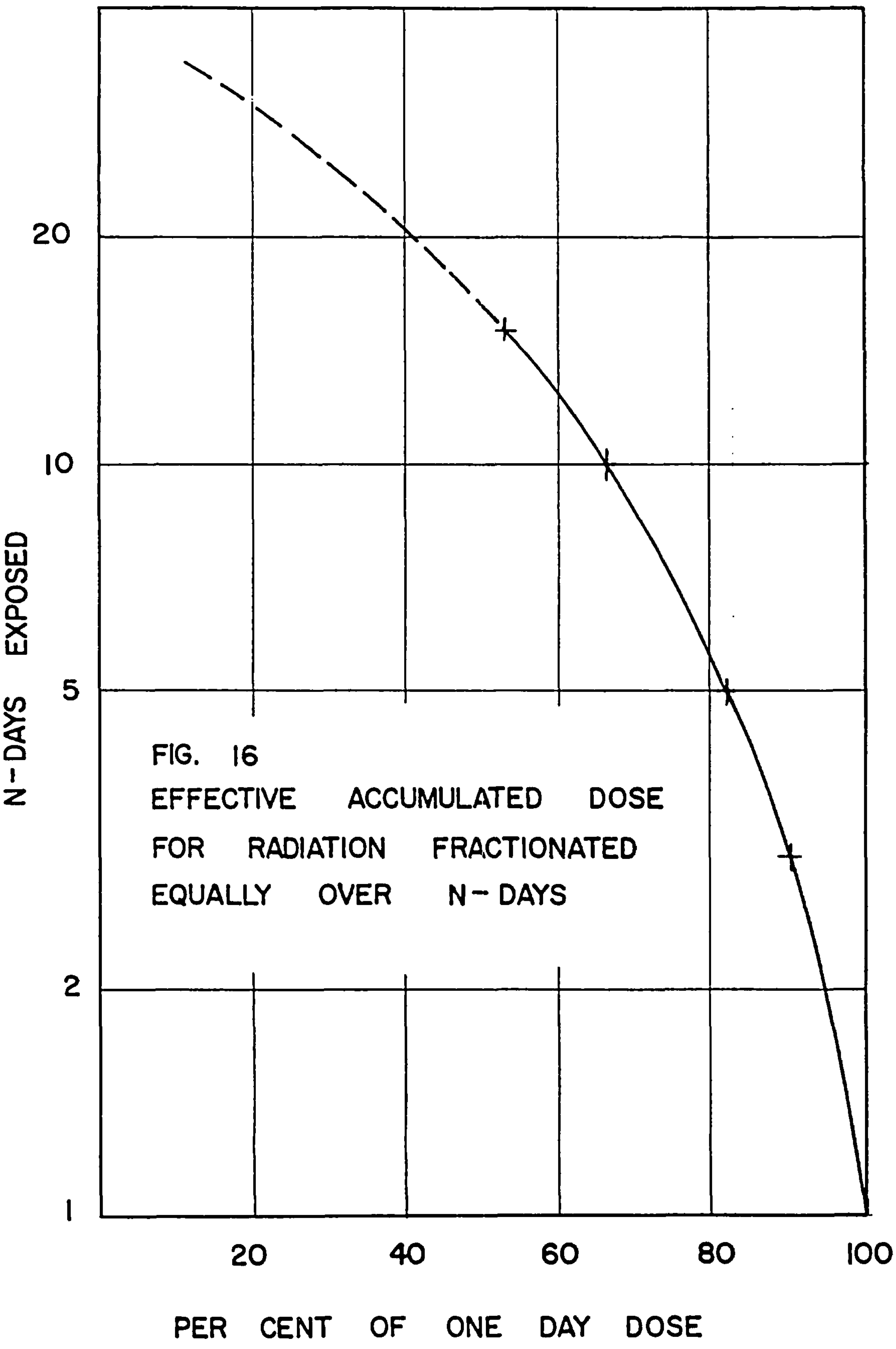


FIG. 16
EFFECTIVE ACCUMULATED DOSE FOR RADIATION FRACTIONATED EQUALLY OVER N-DAYS

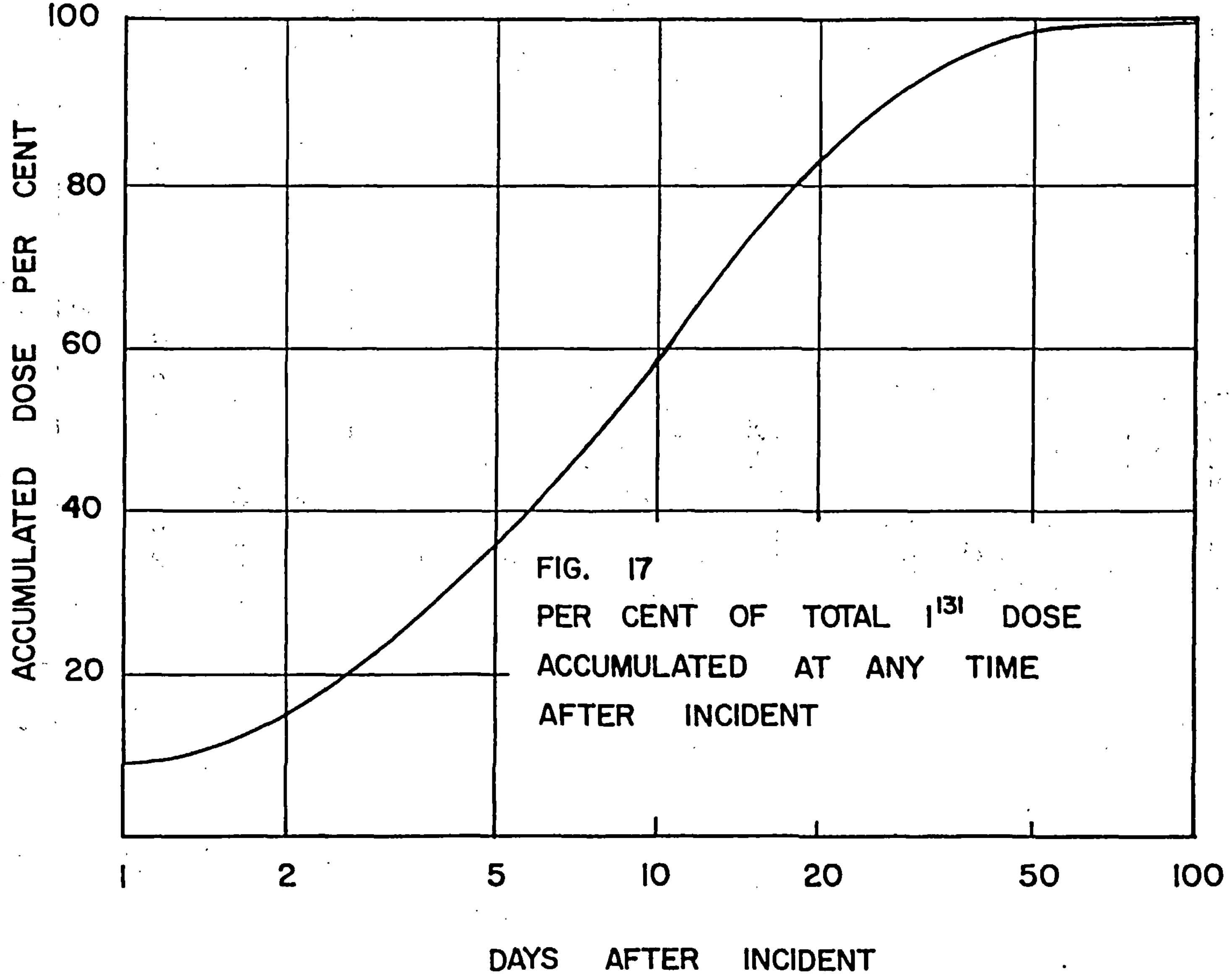


FIG. 17
PER CENT OF TOTAL I^{131} DOSE ACCUMULATED AT ANY TIME AFTER INCIDENT

4. Modifying Factors for a Credible Accident

a. Activity Released. Almost certainly not all of the fission products would be released to the atmosphere and probably 10% would be an upper limit to the activity that gets beyond the immediate vicinity of the reactor. The Borax experiments (Reference 6) indicate that very little activity leaves the reactor area.

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

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

where h is the cloud rise in meters.

How high does the cloud rise? To vaporize all of the aluminum of the fuel elements exclusive of end structure (about 8.3 (10^4) gm in our case) would require about 2 (10^8) calories.

According to Sutton (Reference 10) this would give a total rise of about 400 meters. The cloud does not attain this height instaneously, so that for the shorter distances from the incident one cannot use the full rise for reduction calculations. Probably by 10^4 meters it will have reached its maximum height. Meteorological conditions would determine the actual rise and rate of rise.

Calculating the cloud rise required for a reduction factor of 10^{10} at distances of 10^2 and 10^3 meters, one obtains the following:

Cloud Rise Required in Meters

x	Case I	II	III
10^2 meters	47	57	82
10^3 meters	82	180	390

Probably there will be no significant doses at 10^2 and 10^3 meters with a 400 meters final rise.

The reduction factors due to cloud rise of 400 meters at distances 10^4 and 10^5 meters are:

x	Case I	II	III
10^4 meters	0	.041	0.61
10^5 meters	0.28	0.91	0.99

c. Volume Source. At large distances from the incident the point source treatment is valid. However, at short distances the fact that the source has some volume has a significant effect. As an approximation it will be assumed that the cloud directly after the incident has a volume of 64,000/cu ft. (The reactor bay is 40 x 60 x 120 ft.)

The volume source can be treated as if it were a point source located upwind a distance χ_0 such that at the actual origin it would have attained the actual initial cloud size. For χ_0 Holland (Reference 10) gives

$$\chi_0 = \left[\frac{2Q/\chi(0)}{\pi^{3/2}C^3}\right]^{2/3(2-n)} \qquad \chi(0) = \frac{Q}{\text{Volume of Cloud}}$$

$$\chi_0 = \left[\frac{657}{C^3}\right]^{2/3(2-n)}$$

This gives a virtual source origin χ_0 equal to 725,213 and 82 meters for Case I, II and III respectively.

The dose reduction factors considering the volume source are obtained by using $\chi_0 + \chi$ in place of χ in the dose formula. These factors are as follows:

χ	Case I	II	III
10^2	0.043	0.15	0.36
10^3	0.44	0.73	0.89
10^4	0.89	0.93	0.98
10^5	1.0	1.0	1.0

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

e. 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 decay before release.
2. In the continuous source equation of Sutton, the wind direction is considered as unvarying. This is most unrealistic if the release extends over a few hours. If the cloud sweeps uniformly through an angle θ during the time of release, the area

swept will be uniformly irradiated (neglecting the edges). The resulting reduction in the dose is a function of the distance from the incident and the degree of shift of wind direction. The estimated leakage time for the Curtiss-Wright reactor building intact is estimated to be about 30 hours. In this time, extremely conservative values of θ would be 20° for Case I, 25° for Case II and 30° for Case III. The following reduction factors for a continuous source are obtained.

χ	Case I	II	III
10^2	0.10	0.18	0.34
10^3	0.06	0.13	0.27
10^4	0.03	0.08	0.20
10^5	0.02	0.06	0.15

f. Width of Cloud. Estimating the hazard to an area requires some knowledge of the area covered by a cloud. A cloud width is defined as the crosswind distance where the concentration is p percent of the axial concentration. From Sutton's equations:

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

The value of y for a 10% ($y_{1/10}$) and a 1% ($y_{1/100}$) reduction is given:

χ	Case I		II		III	
	$y_{1/10}$	$y_{1/100}$	$y_{1/10}$	$y_{1/100}$	$y_{1/10}$	$y_{1/100}$
10^2	3.04	4.3	7	9.9	16	22.6
10^3	17.1	24.2	48	68	120	170
10^4	96.0	136	304	430	910	1290
10^5	539	762	2160	3050	6900	9760

Figure 18 illustrates the half width for $y_{1/10}$

5. External Dose from Passing Cloud.

a. External β-dose. The external β-dose (see Reference 10) is given as

$$D_\beta = \frac{0.64\, Q_\beta \exp\left[-\frac{h^2}{C^2\chi^{2-m}}\right]}{\pi\mu C^2\chi^{2-m}\; 6.8\times10^{10}} \quad \text{rep}$$

Figure 19 gives this dose for the three weather conditions for an instantaneous ground source.

The reduction factors given for the inhalation dose are applicable here. The β-dose ignores all shielding afforded by clothing or shelters.

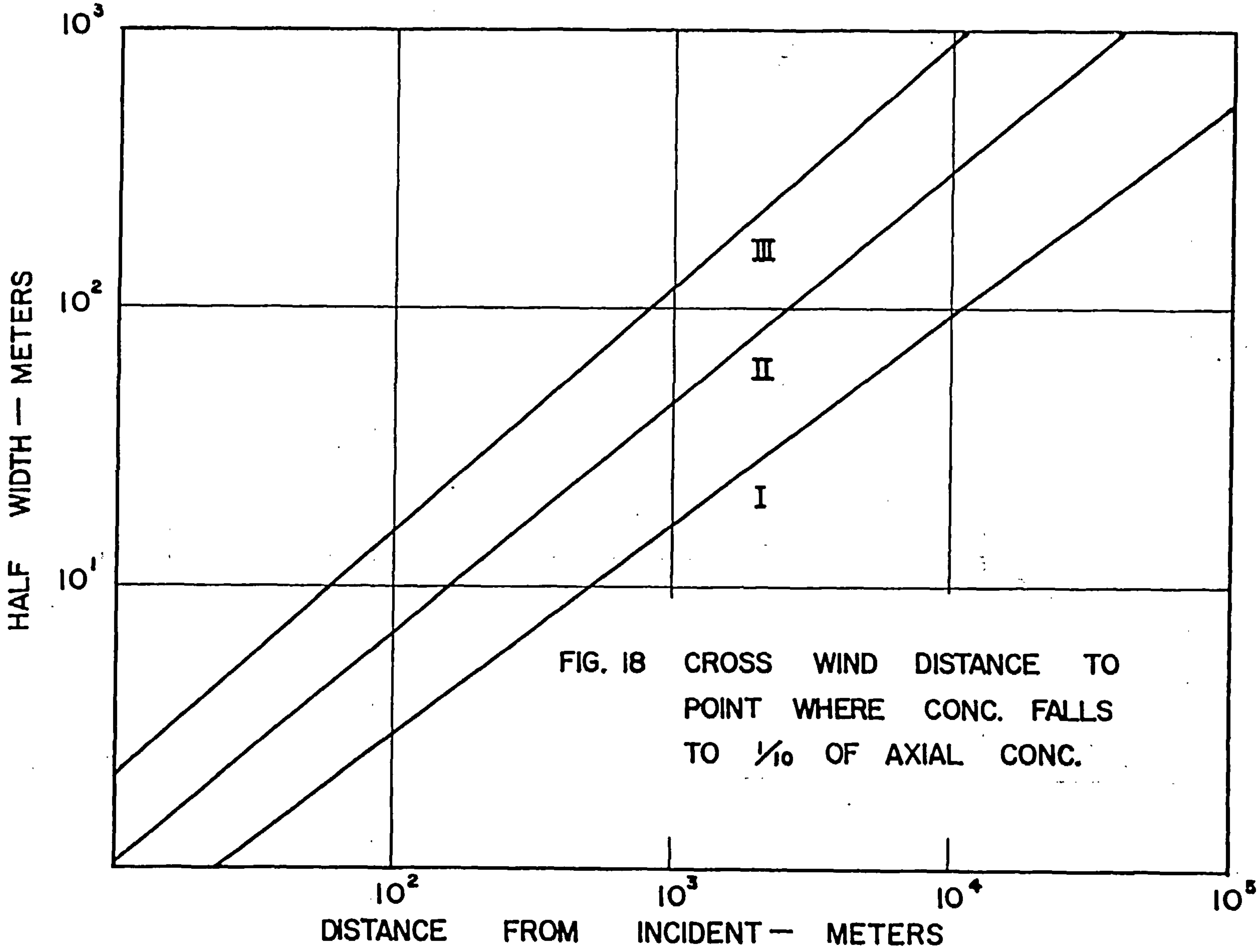


FIG. 18 CROSS WIND DISTANCE TO POINT WHERE CONC. FALLS TO 1/10 OF AXIAL CONC.

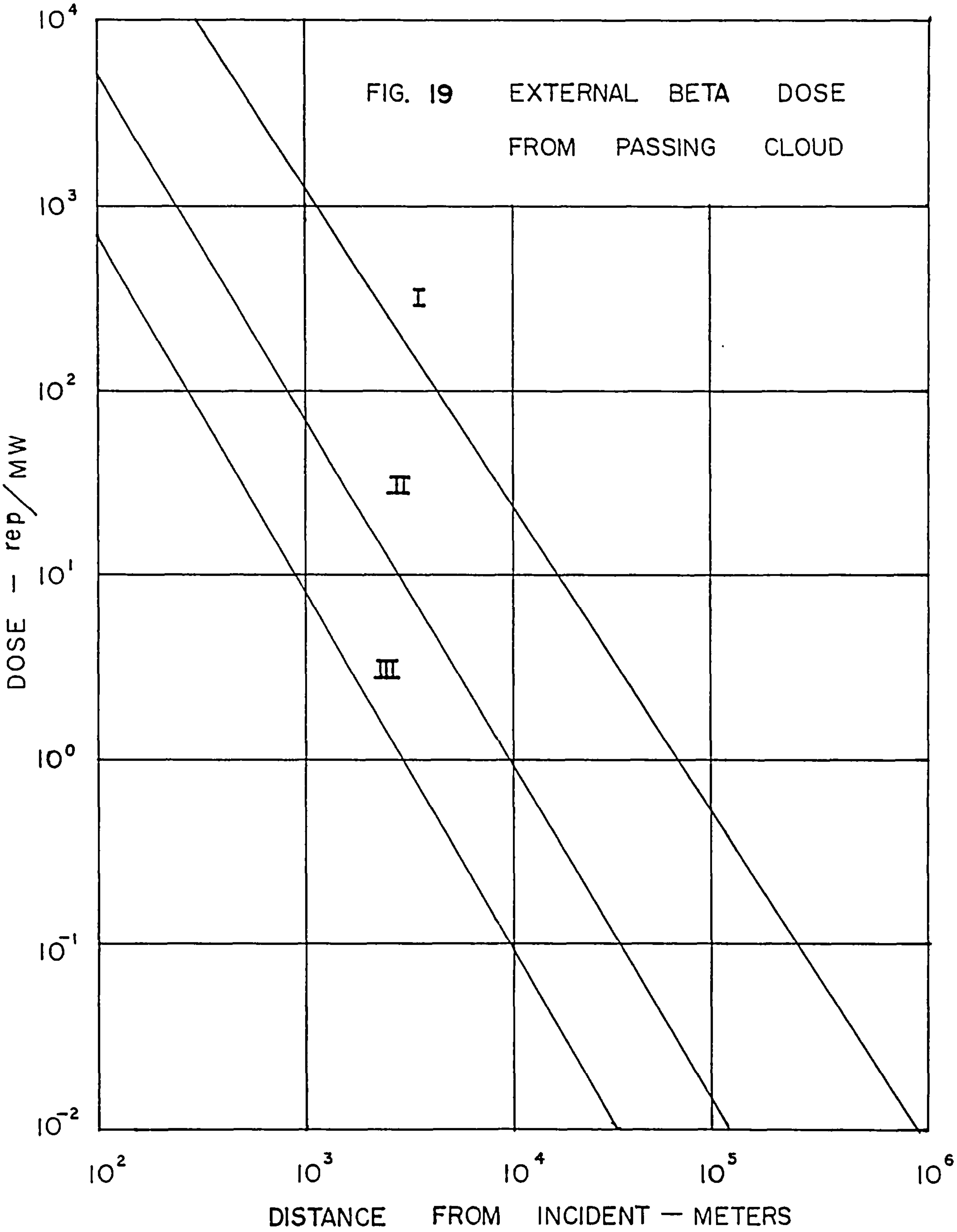


FIG. 19 EXTERNAL BETA DOSE FROM PASSING CLOUD
I
II
III
10⁴
10³
10²
10¹
10⁰
10⁻¹
10⁻²
DOSE - rep/MW
10²
10³
10⁴
10⁵
10⁶
DISTANCE FROM INCIDENT — METERS

b. External γ-dose. Figure 20 gives the external γ-dose determined by Holland's method (Reference 10). The reduction factors given previously are not applicable here.

6. γ-Dose from Ground Deposition. The γ-dose from activity deposited on the surface (see Reference 10), may be expressed as

$$D_\gamma = 3(10^{-3})\,\omega_0\left(\frac{1}{0.79}\right)\left[t_2^{0.79}-\left(\frac{x}{\mu}\right)^{0.79}\right]$$

where ω_0 = surface deposition c/m^2
x/μ = the time to reach a point x from origin
t_2 = the time after the incident over which the dose is integrated.

a. Dry Fall-Out. As previously indicated, to specify fall-out deposition a knowledge of particle size, density and shape is required. This is unknown, but a maximum deposition, ω_0, can be calculated for any particular location. This deposition is given as

$$\omega_0 = \frac{n\,Q}{2\epsilon\pi^{1/2}C\,x^{(2-n/2)}}\ \text{curie}/m^2$$

Figure 21 gives the γ-dose for the first day due to dry fall-out. It is valid for considering only one point at a time; it does not represent a maximum dose distribution for one situation.

b. Rain-Out. The expression for maximum rain-out deposition is

$$\omega_0 = \frac{Q}{\epsilon\pi^{1/2}C\,x^{(2-n/2)}}$$

It is seen that the rain-out γ-dose is $\frac{2}{n}$ times as great as the dry fall-out dose. This is equal to 4, 6 and 8 times as great for Case I, II and III respectfively.

7. Theoretical Incident. Several specific examples of incidents and their effects at a number of sensitive locations will be considered in this section. On the whole the Quehanna site is reasonably isolated. Four locations will be considered for the theoretical incident.

Location 1: the shortest distance from the reactor to the site boundary. This is about 3 mi SW.
Location 2: the shortest distance to a population center on-site. This would be a distance of 5 mi ESE to the Main Area where the Research Plastics and Administration Buildings are located. It is expected that normally several hundred people will work here.
Location 3: the shortest distance to an off-site population center. This is a distance of 10 mi to each of four towns, Karthaus SE, Driftwood NNE, Sinnemahoning NE and Benezette NW. Karthaus with 575 people is the largest town.
Location 4: along Mosquito Creek for inversion situations.

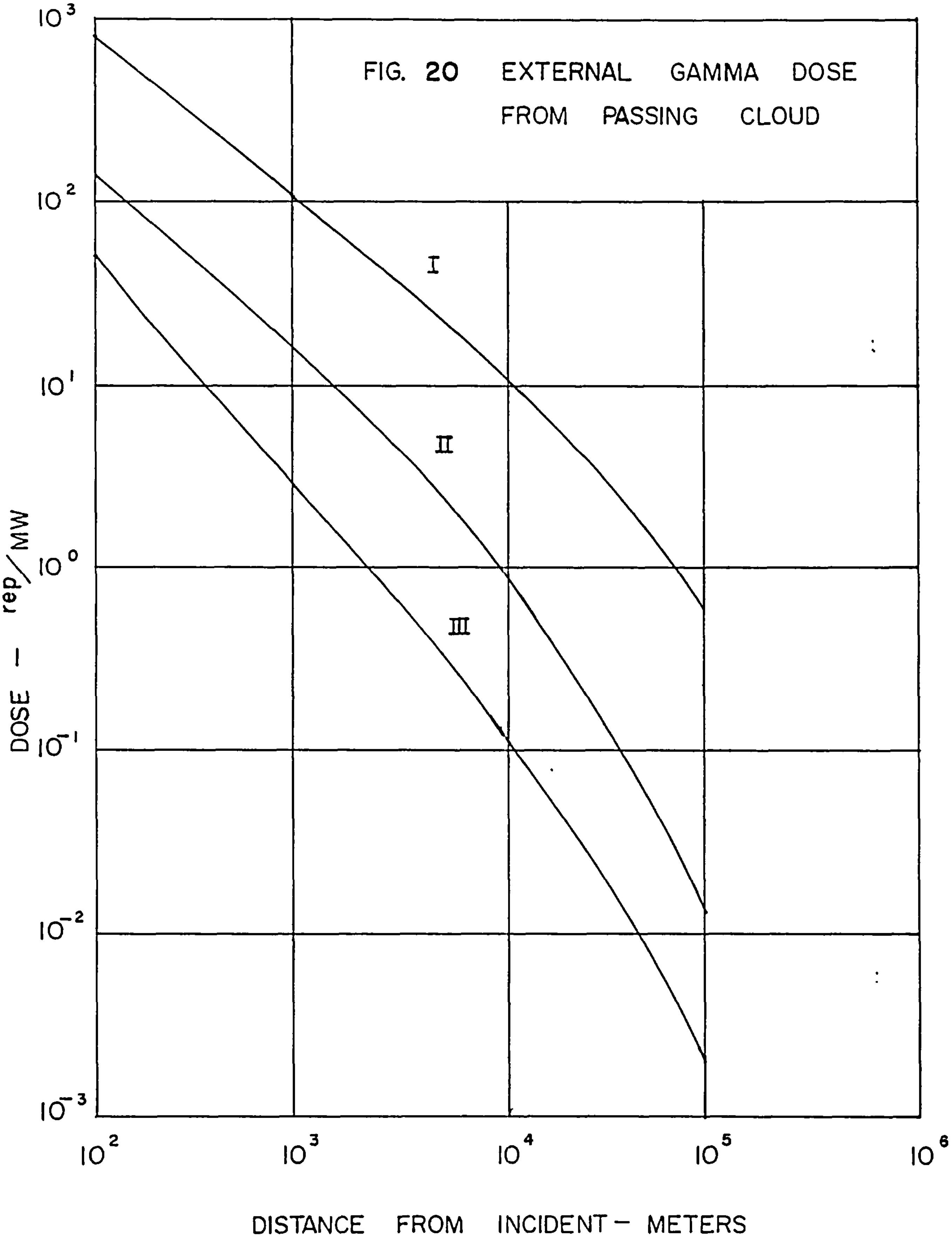


FIG. 20 EXTERNAL GAMMA DOSE FROM PASSING CLOUD

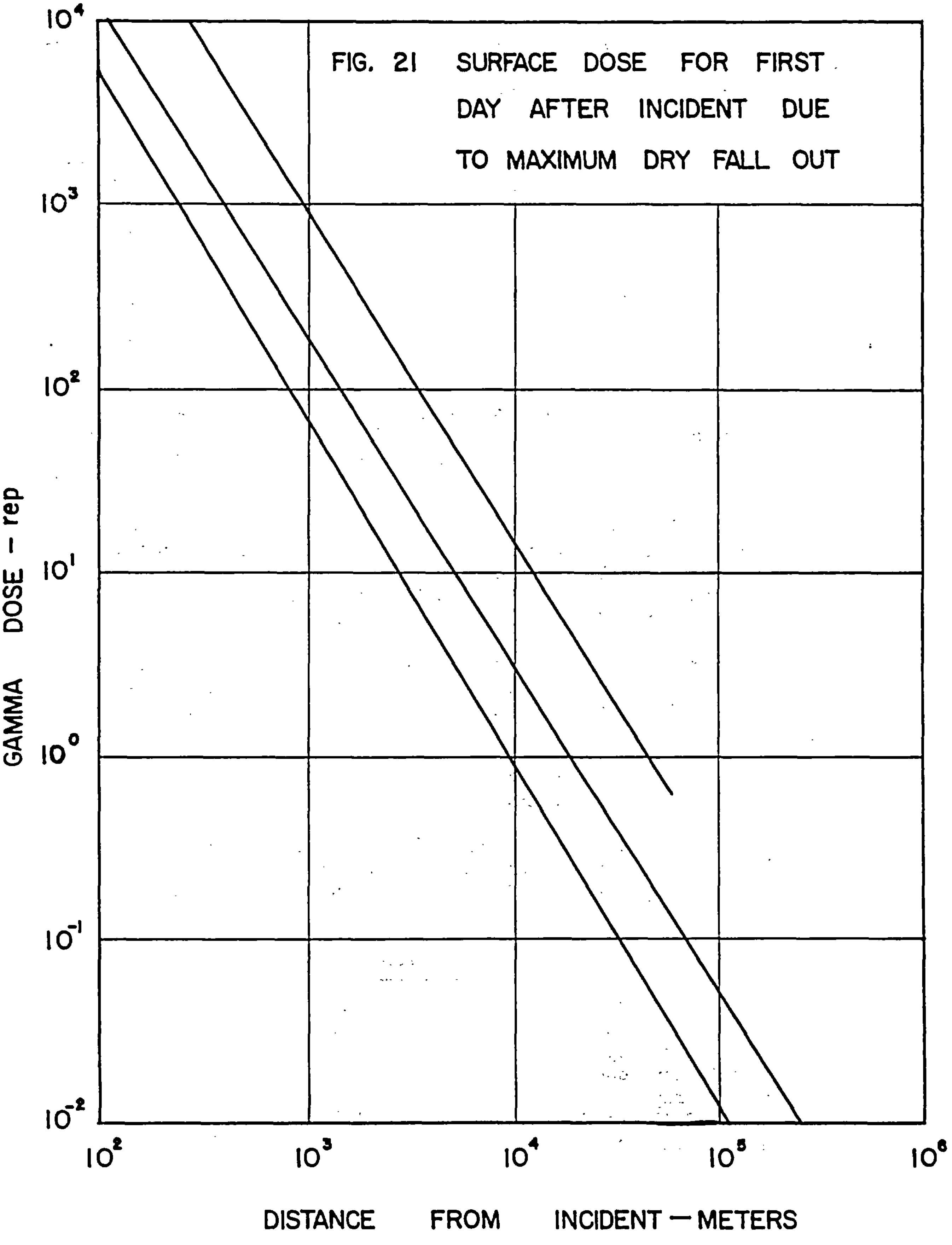


FIG. 21 SURFACE DOSE FOR FIRST DAY AFTER INCIDENT DUE TO MAXIMUM DRY FALL OUT

a. Location 1: Site boundary (3 mi). Non-Curtiss-Wright employees may be just beyond the site boundary either day or night. Considering first a 100% ground release, the doses are the following:

	Case I	II	III
Cloud width, ($2\ y_{1/10}$), meters	100	320	880
External β-dose, rep	100	4	0.4
External γ-dose, rep	25	2.2	0.4
Total inhalation dose (thyroid), rep	4,000	130	13
Maximum dry fall-out one-day dose, rep	70	15	4
Maximum rain-out one-day dose, rep	280	90	32

It may be safely said that with the reactor located in the valley, meteorological Cases I and II cannot apply to this location due to channeling. Rather, the cloud would travel generally south along Meeker Run and Mosquito Creek.

Generally the inhalation dose is the most important contributor to the total dose. If the cloud were to rise as would be likely for a 100% release, then the inhalation dose for Case III would be reduced to about 2 rep. If one considers 10% of the fission products released as a continuous source, the dose would reduce to about 0.3 rep.

The maximum rain-out can produce a significant dose which would not be reduced by cloud rise. However, even in the case of 100% release the emergency warning system would make the full one-day dose of 32 rep unlikely.

It is not too probable that a serious dose would be received at the boundary location considered.

b. Location 2: Main Area on-site (5 mi). This area is populated during daylight hours so that only Case II and III need be considered. Again considering a 100% instantaneous ground release, the doses are the following:

	Case II	III
Cloud width, ($2\ y_{1/10}$), meters	500	1440
External β-dose, rep	1.2	0.12
External γ-dose, rep	1.0	0.13
Total inhalation dose (thyroid), rep	52	4.6
Maximum dry fall-out dose, rep	4.5	1.3
Maximum rain-out dose, rep	27	10.4

Again Case II which represents inversion conditions can be neglected as far as this direction is concerned and the dose for Case III which was not serious at 3 mi is less so at 5 mi.

c. <u>Location 3</u>: off-site population centers (10 mi). The arguments for the closer locations apply here and no serious situation exists with the possible exception of Karthaus which lies on Mosquito Creek. This is treated below.

d. <u>Location 4</u>: along Mosquito Creek. Clearly the valleys of the Mosquito Creek and the West Branch of the Susquehanna offer the greatest possibility of potential over-exposure. During inversions activity released at the reactor will probably be confined to the valley. If there is a great evolution of heat and the cloud is able to rise and penetrate the inversion layers, the situation is entirely different and the inversion might prevent over-exposures at the ground.

The critical points along the valley are the intersection with the site boundary (at least 5 mi), Karthaus, the first population center (at least 15 mi), and Renovo, the second population center(50 mi).

Since the population centers are reasonably far from the reactor, several factors tend to ameliorate the situation. First, with the inversion conditions envisioned the wind speed would not exceed a few miles per hour and there would be about 7 to 8 hours for warnings to Karthaus and a full day for Renovo. Second, during this time, there are reasonable chances for meteorological conditions to change and improve diffusion of the cloud. Or, it if possible for "valley-slope circulation" to reverse the flow during the afternoon hours and keep the cloud confined between the site and Karthaus.

Karthaus lies on the southern slope of the valley of Mosquito Creek, rising to about 150 ft above the creek on its uphill border, and about 500 meters in projected width. If diffusion was the same as for level terrain during night time inversion conditions, then the half-width of the cloud would be about 180 meters and the maximum dose, at the cloud center, or in the creek, would be 32 rep for a 10% release and the dose to most of the town would not be serious. If the cloud had diffused less than it would have over level terrain, then the dose would be higher (about 100 rep maximum for a half-width of 100 meters) but only a small fraction of the community would be affected and a short evacuation, south-west, along Route 879 would provide safety.

B. Radiation Hazards Due to Release of Radioactive Material to Ground Water

There are two conceivable, although highly improbable ways in which gross fission product activity could reach the ground waters of the Quehanna area. Release of activity into the atmosphere during or just prior to a heavy rainfall would result in most of the activity coming down as rain-out and running off into the streams of the area. An even less likely event would be a power excursion in which the fuel elements would melt, releasing fission products to the pool water, following which the water might escape through a cracked pool wall.

In either case it seems pessimistic to assume that as much as 10% of the fission product activity will escape from the core. If the fuel elements are taken to 10% burnup, the greatest concentration of fission products available will be that following one megawatt-year of operating time. Again I-131 will be considered the most dangerous isotope.

It will be assumed that 10% of the maximum available I-131, or about 2350 curies, will be released into the atmosphere. In the case of escape during a rainfall, it would wash out and regardless of the prevailing wind the runoff would find its way into the Mosquito Creek. Assuming that the rain-out of I-131 is accompanied by a total fall of 0.1 in. over the 35 sq mi drained by the Mosquito Creek, and that both rain-out and runoff are complete, about 6×10^7 gal of water with an average concentration of 0.01μc/ml would dump into the Mosquito.

In the event of a simultaneous release of activity into the pool water and rupture of the pool wall, it may be assumed that 1.4×10^5 gal with an average I-131 concentration of 4.3μc/ml (again assuming a 10% release of fission products) would reach Meeker Run (see Figure 9). This stream joins the Mosquito Creek which in turn joins the West Branch of the Susquehanna River at Karthaus. It is reasonably conservative to assume that diluting by a factor of 10 would occur between the time the pool water enters Meeker Run and the time it merges with the West Branch. If this river is flowing at its average rate of 1×10^6 gpm (Reference 11) the concentration to be expected at various points downstream from Karthaus for the two cases are shown below.

	Rain-Out μc/ml	Pool Rupture μc/ml
West Branch Susquehanna River, Karthaus	$5(10^{-3})$	$5(10^{-1})$
West Branch Susquehanna River, Williamsport	$1.4(10^{-3})$	$1.4(10^{-1})$
Susquehanna River, Harrisburg	$3.6(10^{-4})$	$3.6(10^{-2})$

The values above may be considered to be peak concentrations. Under the pessimistic conditions assumed these concentrations would exist at any point for a period of no more than several hours.

The integrated dose from drinking contaminated water is given by the following equation:

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

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

Setting $t = \infty$ and $\frac{fE}{M\lambda}$ for I-131 equal to .0246 (Reference 12) and V for accidental consumption equal 18250 ml, the equation reduces to

$$D = 3.4 \times 10^8 A$$

The integrated doses for the locations above are then

	Rain-Out (rep)	Pool Rupture (rep)
Karthaus	1.7	170
Williamsburg	0.48	48
Harrisburg	0.12	12

The immersion dose is negligible compared to the dose from drinking the contaminated water.

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APPENDICES

APPENDIX I -- GEOLOGY AND HYDROLOGY OF THE QUEHANNA SITE

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

A. Geology

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

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

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

B. Hydrology

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

TABLE A-1

Generalized Section of Geology at Quehanna, Karthaus, Pennsylvania

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

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

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

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

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

TABLE A-2

Drillers Logs of Test Well 1 and 2

Test Well No. 1 Thickness (in feet)	Test Well No. 1 Depth (in feet)	Test Well No. 1 Strata	Tentative Geologic Correlation	Test Well No. 2 Thickness (in feet)	Test Well No. 2 Depth (in feet)	Test Well No. 2 Strata
15	15	clay		15	15	clay
65	80	sand hard		15	30	sand
12	92	shale	Probably lower Pottsville formation	25	55	shale
76	168	sand		25	80	sand
7	175	shale		20	100	slate and shale
10	185	red rock		40	140	sand
8	193	shale	Probably Mauch Chunk shale	40	180	slate
27	220	shale and sand				
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 formation			
15	525	sand				
23	548	slate and shells				
42	590	sand				
5	595	slate and shells				
22	617	sand-hard				
18	635	red rock				
10	645	slate and shells				
15	660	red rock				
40	700	slate and shells				

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

Well No. 1

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

Well No. 2

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

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

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

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

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

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

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

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

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

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

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

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

G. Earthquake Activity

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

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

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

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

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

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

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

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

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

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

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

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

TABLE A-3

List of Earthquakes with Epicenters in Pennsylvania

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

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

D. Quehanna Operations

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

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

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

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

In the case of overland movement of radioactive fluids, the time factor is short in moving a contaminant 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 contaminant 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 reasonable 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

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Appendix II — Meteorological Appraisal of the Quehanna, Pa. Site of The Curtiss-Wright Corporation

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

A. Introduction

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

B. Local Topography

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

C. Source of Data

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

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

D. Climatological Review

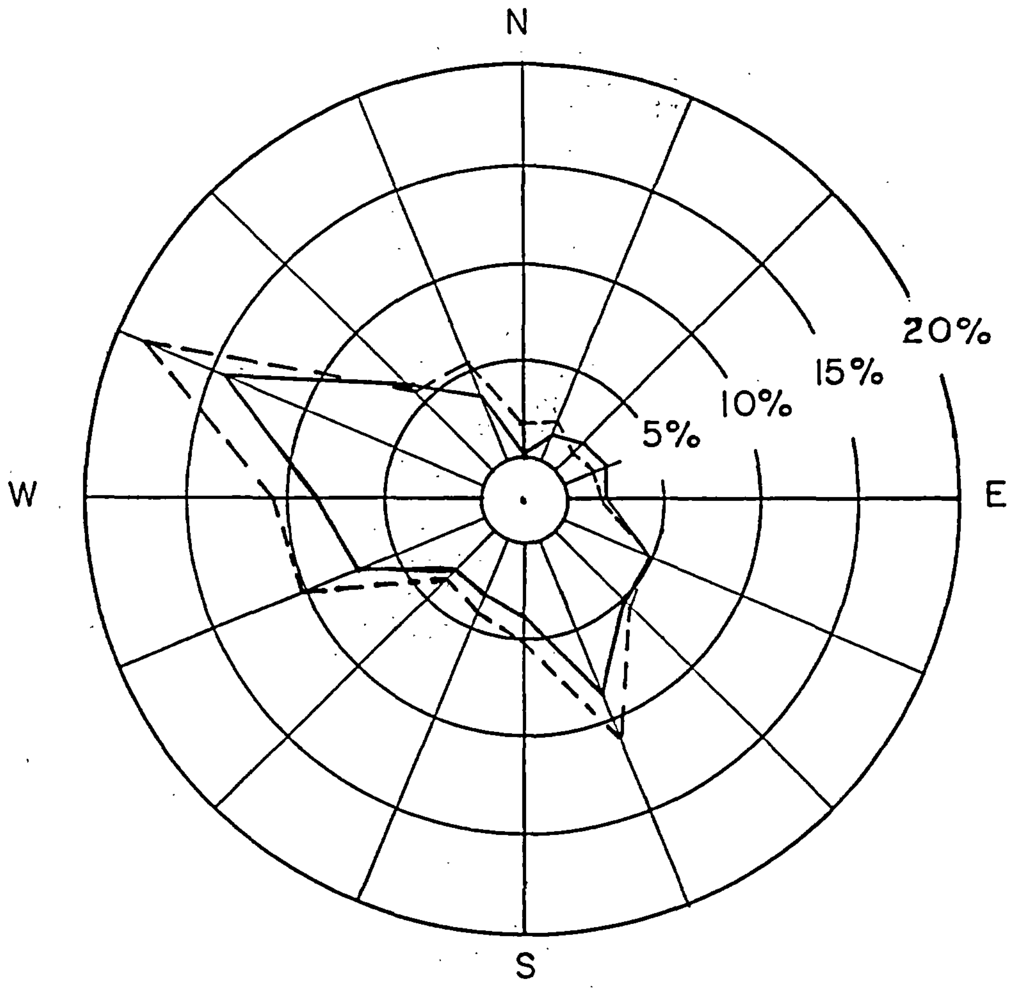
The general climate of Pennsylvania is a modified continental type with occasional intrusions of Atlantic maritime air from the east. The area has generally adequate rainfall without extreme variations from year to year. Temperatures have, in general, a continental range with hot summers to cold winters, 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. Figure A-1 shows the remarkably constant prevailing wind directions with various wind conditions somewhat more graphically than does the table. It can be easily seen that about 30 to 40% of the winds are from the west-southwest through west-northwest and generally speaking, 20% of the remaining winds are from southeast to south quadrant regardless of the weather conditions occurring at that time.

TABLE A-4

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

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



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

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

FIG. A1 ANNUAL FREQUENCY AND WIND DIRECTION

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

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

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

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

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

TABLE A-5

Annual Frequency of Wind Speeds (Percent)

Mph	Calm	1-3	4-12	13-24	25-31	32-46	> 47
Daylight	6.8	3.4	59.8	27.3	2.2	0.4	< 0.1
Night	23.0	7.9	51.2	16.3	1.3	0.3	< 0.1
During precipitation	5.1	3.4	52.4	34.4	3.8	0.9	< 0.1
Visibility ≤ 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 Rainfall 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 fom nearly every night there is nothing in the available records which could be interpreted as signifying that the Quehanna area experiences any unusual poor stability conditions. On the contrary, the fairly high elevation and the high average wind speed would indicate that good atmospheric dispersion would be obtained during the majority of the daylight hours.

5. General Weather Conditions. The average monthly temperature for the Quehanna area will range from about 65° in July to a low of nearly 22° in January, with an estimated annual temperature of around 44°. The area can expect generally cool nights with considerable temperature extremes between the hilltop areas and the valleys. The nighttime minimum temperatures in the 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 $\geq$ 1 in.	Temperature $\leq$ 32	Temperature $\leq$ 0
33	69	16	157	8

TABLE A-10

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

Winter	11%
Spring	7%
Summer	3%
Fall	6%
Average Annual	6.6%

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

E. Conclusions

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

APPENDIX III -- DERIVATION OF THE INHALATION DOSE EQUATION

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

D = Dose in rep/day

A = 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 = Effective decay constant (day^{-1})

$$\text{TID} = \frac{2Q}{\pi C^2 \mu \chi^{2-n}}$$ from Sutton's equation. (Reference 10)

The integrated dose from time of inhalation to time t is

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

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

The integrated dose in rep is therefore

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

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

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

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

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