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VOLUME I

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1954

THE OAK RIDGE NATIONAL LABORATORY

RESEARCH REACTOR SAFEGUARD REPORT

VOLUME I

F. T. Binford
Editor

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ERRATA TO VOLUME I
ORNL 1794

December 1, 1954

Page 5 Line 1	Change "twenty-six feet eight inches" to "twenty-nine feet eight inches"
Page 5 Line 2	Change "twenty-one feet of water" to "twenty-four feet of water"
Page 8 Figure 3	Change "2nd floor" to "3rd floor"
Page 9 Figure 4	Change "2nd floor" to 3rd floor"
Page 11 Figure 6	Change "2nd floor to 3rd floor"
Page 11 Figure 6	Change "balcony" to "2nd floor"
Page 12 Line 14	Change "adjacent to the NE corner of the building" to "east of building 3010, the Bulk Shielding Facility"
Page 26 Line 10	Change "approximately two feet" to "approximately three to four feet"
Page 26 Line 15	After "3" overflow line" insert the words "terminating 3 - 4 feet above the reactor lattice"
Page 27 Line 2	Change to read " a ceiling either of poured barytes concrete two feet thick, or of ordinary concrete three feet thick"
Page 27 Line 6	After "facilities" add "and cooling loop demineralizer"
Page 32 Figure 13	Change "300 GPM" to "150 GPM"
Page 33 Line 21	Change "offices" to "office"
Page 35 Figure 15	Remove twin bays on east side
Page 37 Figure 17	Roll up door on North side should be moved from left to right hand side
Page 39 Line 7	After "located in the basement" add " and power exhausters on the roof"
Page 73 Line 1	Change "site of the river" to "site to the river"

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ERRATA TO VOLUME I (Cont')
ORNL-1794

Page 84	Figure 22	Multiply source strengths by 10^{-3}
Page 85	Figure 23	Multiply source strengths by 10^{-3}
Page 86	Line 19	Change "470 curies" to "470 μ curies"
Page 86	Line 24	Change "3.7 miles" to "4.7 miles"
Page 87	Line 4	Change "exceeded only at night" to "exceeded at night"
Page 87	Line 4	Change "under 0.2" to "under 36"
Page 87	Line 4	After word "miles" insert "and during the day under 6.5 miles"
Page 87	Line 18-19	Delete the sentence beginning "These distances ----"
Page 88	Figure 23	Ordinate is "time in minutes"
Page 89	Figure 24	Ordinate is "time in minutes"
Page 90	Figure 25	Reduce values of KAPL 1045 by factor 1/3 at each distance
Page 91	Table II-(2)	Increase all values of dose by a factor of 10^3 except MPE and the following entries in the second column. For the 10 mw dose corresponding to 2.2m increase by 10^2 For the 10 mw dose corresponding to 10m increase by 10^2 For the 10 mw dose corresponding to Max increase by 10
Page 92	Table II-(4)	Increase all values of doses except MPE and the 10 mw dose corresponding to 5m by a factor 10^3 For the 10 mw dose corresponding to 5m increase by factor 10^4
Page 93	Table III	10 micron column. Entries corresponding to hot cloud should read > 100
		1 micron column. Entry corresponding to hot cloud-day should be deleted
		Entries corresponding to cold cloud should read >100

ERRATA TO VOLUME I (Cont')
ORNL-1794

Page 94	Line 5	Change last word from "complete" to "maximum"
Page 95	Line 17	Change "8 miles" to "10 miles"
Page 95	Line 18	Change "8 miles" to "3 miles"

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THE OAK RIDGE NATIONAL LABORATORY RESEARCH REACTOR

SAFEGUARD REPORT

Research Reactor Design Group
F. T. Binford, Report Editor
T. E. Cole, Chairman, Steering Committee
~~J. P. Gill, Project Engineer~~

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ACKNOWLEDGMENT

This report was prepared through the cooperative efforts of the Oak Ridge National Laboratory Research Reactor design group. Other Laboratory organizations shared responsibility in their special fields.

In particular we wish to acknowledge the assistance of R. F. Meyers and D. L. Purdy of the U. S. Weather Bureau, and of T. H. J. Burnett of the ORNL Health Physics Division.

1. Abstract

The proposed ORNL Research Reactor is designed to serve as a general purpose research tool delivering a maximum thermal flux of 8×10^{13} n/cm^2-sec at the initial power level of five megawatts. Operation at power levels up to ten megawatts is proposed for such times as sufficient cooling capacity is available to handle the increased heat load.

The reactor will use MTR type fuel elements and beryllium reflector pieces in a 7 x 9 grid with moderation and cooling provided by forced circulation of demineralized water. The reactor tanks are submerged in a barytes concrete pool, filled with water, which serves as a biological shield. Experimental facilities include two 18" diameter "Engineering Test Facilities" and six 6" diameter beam holes. In addition, access to the core is available through the water of the pool.

The result on the surrounding population of release to the atmosphere of a large fraction of the radioactive material in the core has been computed by two methods. It is shown that under certain conditions off-area personnel could be subjected to greater than the maximum permissible exposure. An analysis of the maximum hazard caused by the release of the entire contents of the core to the local watershed indicates that the resulting incident could be quite serious, but with proper monitoring and supervision would probably not constitute a lethal hazard. The probability of the occurence of a catastrophic release of activity of sufficient magnitude to cause widespread hazard to life is quite small and it is believed that the measures taken to lessen this probability are adequate.

An Appendix, Volume II, contains supporting information for this report, and is also intended to serve as a reference for future use.

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2. INTRODUCTION

The history of the ORR (ORNL Research Reactor) Project began in the early part of 1950 when it was determined that a permanent reactor facility was required at the Oak Ridge National Laboratory in order to carry out successfully the various research programs, then being undertaken. The reactor selected was a simplified version of the MTR (Materials Testing Reactor) to operate at a power level of five megawatts, which would require a minimum of design and development. Preliminary design of the ORR has been supported by the Atomic Energy Commission for the past three years, during which period the facility has undergone a number of design changes as required by the shift in emphasis on the experimental programs. With the operating experience of the MTR and other reactors as a guide, and the additional data now available in the field of reactor technology a number of improvements have been made in the basic design.

The need for improved facilities for experiments requiring neutron beams, expanded facilities for isotope production, and the necessity for an irradiation facility to permit engineering tests of the materials and components of liquid fuel reactors has become increasingly urgent. Construction and operation of the reactor proposed (ORR), in conjunction with existing facilities, should meet the requirements for neutron irradiation space of both the basic and applied research programs of the Laboratory and the requirements for radioisotope production at this location insofar as they can be anticipated at this time.

A Preliminary Proposal[1] which outlined the basic features of the reactor was submitted to the Atomic Energy Commission in March 1954. This proposal was approved, and a directive was issued permitting the use of funds to secure the services of an Architect Engineer. A contract, calling for the definitive design of the building, reactor shielding structure, and cooling system was excecuted, on July 21, 1954, with The McPherson Company of Greenville S. C. On August 25, 1954 a modification of this directive was issued authorizing design and procurement of the reactor and controls by the Oak Ridge National Laboratory. Construction of the building, reactor shielding structure, and cooling system is to be performed by a contractor who will be selected on a "lump-sum" bid basis. It is anticipated that drawings will be available for inspection by construction firms on December 15, 1954. The present schedule calls for the selection of a contractor by February 1, 1955. The scheduled date for completion of construction and installation of the reactor is August 1, 1956.

Copies of the Preliminary Proposal were sent to members of the Advisory Committee on Reactor Safeguards. A further description of the facility was given at a meeting of this Committee held at Cincinnati, Ohio on April 21, 1954. The purpose of this report is to make available to the Committee further information concerning the ORR and to present estimates of hazards resulting from the operation of the reactor, in order that these hazards may be reviewed by the Committee.

3. DESCRIPTION OF THE FACILITY

3.a GENERAL DESCRIPTION OF THE REACTOR

The Oak Ridge National Laboratory has proposed [1] that a research reactor of intermediate power be built and operated in support of its basic and applied research programs. It is proposed that this system be operated initially at a power level of five megawatts corresponding to an estimated maximum thermal flux of 8×10^{13} neutrons/cm.^{2}sec. It is further proposed that at such times as adequate cooling is available the system be operated at power levels up to ten megawatts.

The design of the O.R.R. incorporates a heterogeneous core which utilizes enriched uranium fuel with ordinary water as coolant and moderator. The reflector will be a relatively thin layer (approximately 3") of beryllium metal, backed by a thick layer (approximately 4') of water. A plan view of the core is shown in Fig. 1.

The core arrangement, fuel elements, and methods of shim and control are similar to those used in the MTR[2]. The reactor core is housed in the center section of a series of aluminum tank sections approximately fifteen feet in overall height and approximately six feet in diameter as shown in Fig.2. Control drives are operated from below the reactor through a large shielding plug as illustrated in Fig.3. This assembly, together with a top cover plate, constitutes the Reactor Unit which may be completely removed should the occasion arise, see Fig. 4. The design of this reactor unit incorporates many features which have already been tested or which are now being used in other installations.[2],[3],[4],[5]

The reactor unit is located in a pool of demineralized water which

is approximately twenty-one feet long, ten feet wide, and twenty-~~six~~ nine feet-eight inches deep, as illustrated in Figs. 3, 5, and 6. The twenty-~~one~~ four feet of water above the core centerline provides the main shielding above the reactor during operation and also provides the shielding required for the transfer of fuel elements, control rods, and vertical experiments from the reactor to the storage areas of the pool. These storage areas are adjacent to the reactor pool itself and are separated from it by removable aluminum gates. It is proposed to line the entire series of pools with $\frac{1}{4}$" welded aluminum plate in order to help maintain high water purity and to insure against leakage through the pool walls and floor.

Although the pool water is normally independent of the reactor cooling system, provision is made for thermal circulation of this large volume of water through the reactor core should such a procedure be desirable.

The structure above the sub-pile room ceiling and the adjacent storage pools is designed to meet biological shielding requirements during reactor operation and also to permit safe transfer of fuel and experimental equipment between pool sections. The pool walls are to be constructed of high density barytes concrete. The shielding will be a minimum of seven feet thick in the vicinity of the reactor core. Immediately adjacent to the six beam holes, the shield is nine feet thick.

Each of the three pool sections is approximately the same size (see Fig.6) with a total capacity of 120,000 gallons. The sections are separated by removable aluminum gates and can be filled and drained independently. There is sufficient water provided between the reactor and the walls in most cases to prevent serious activation of the concrete. Those walls

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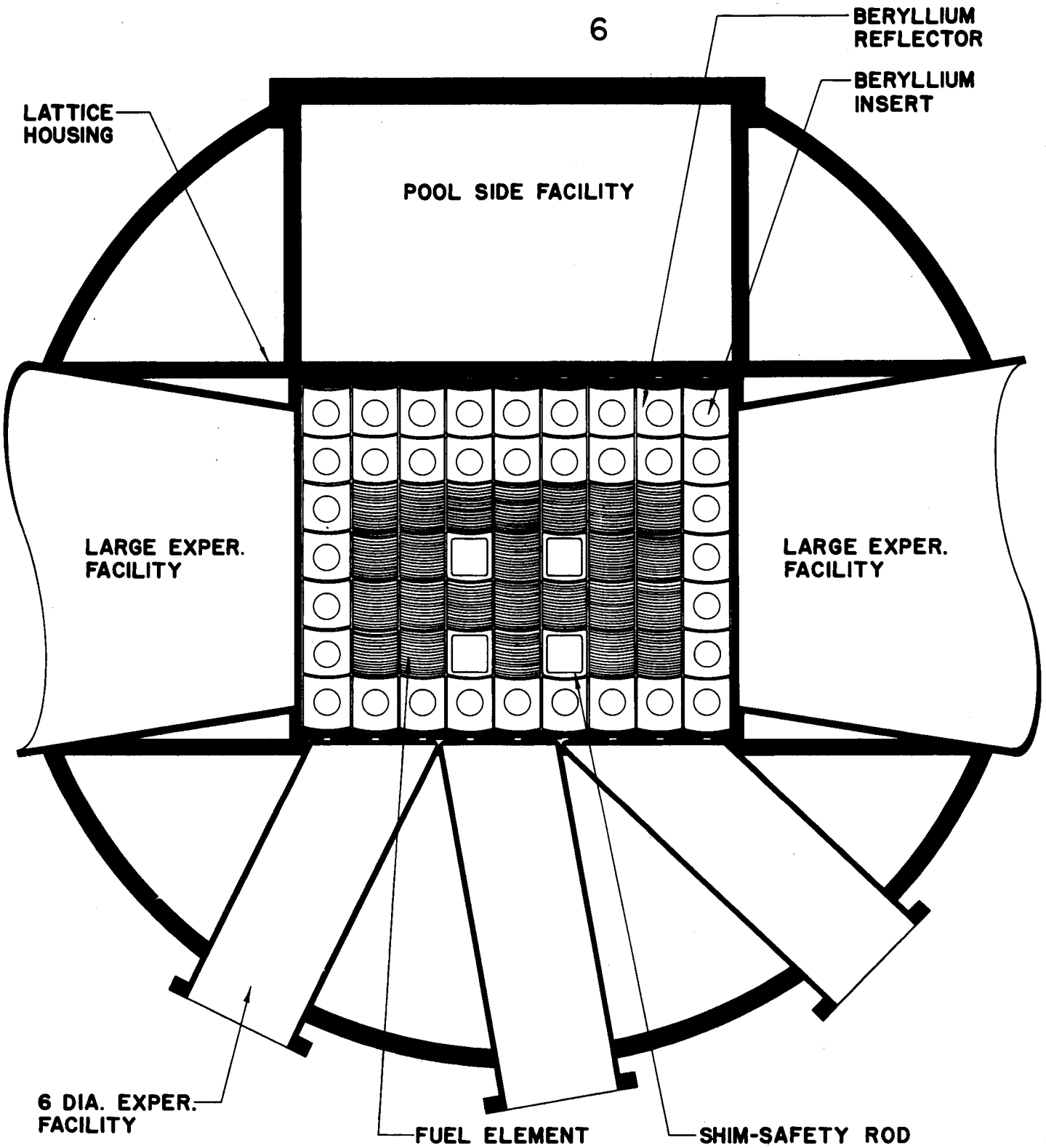

ORR PROJECT
TYPICAL LATTICE
ARRANGEMENT

0 5 10 20
SCALE IN INCHES

Fig.-1

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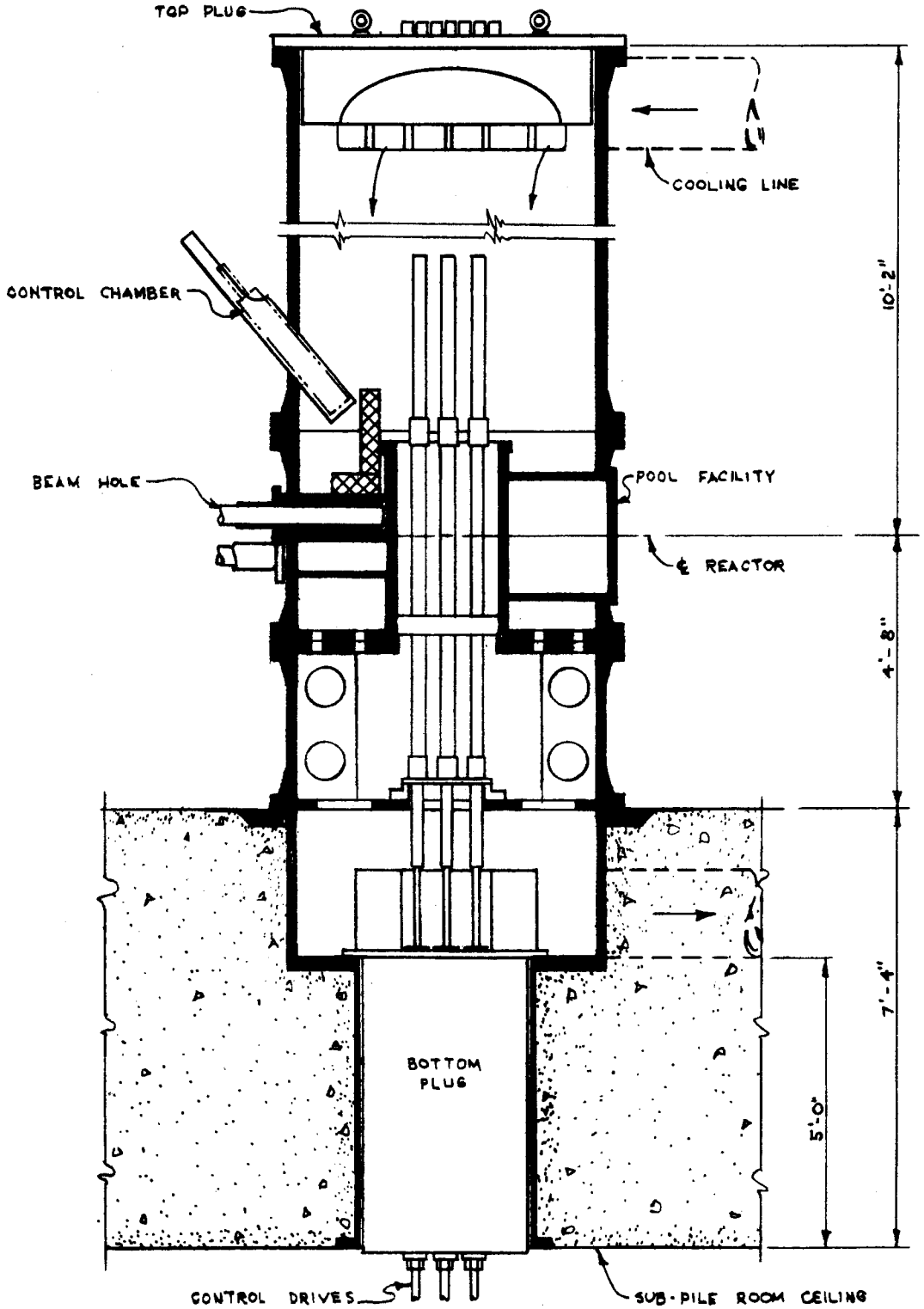

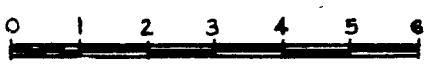

SCALE IN FEET

REACTOR - VERTICAL SECTION

ORR PROJECT

FIG.-2

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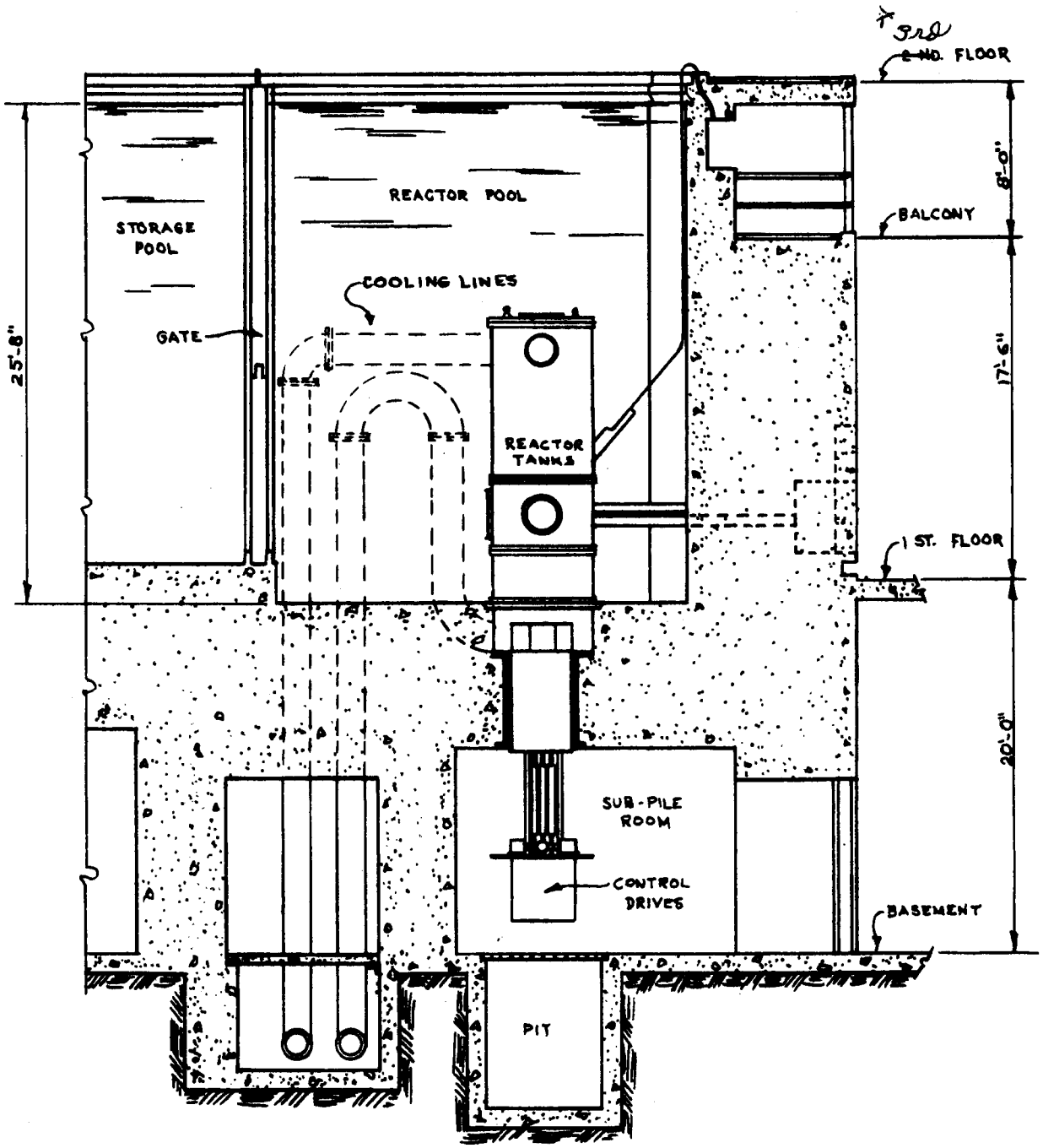

0 5 10 15 20

SCALE IN FEET

REACTOR STRUCTURE - VERTICAL SECTION

ORR PROJECT

FIG.-3

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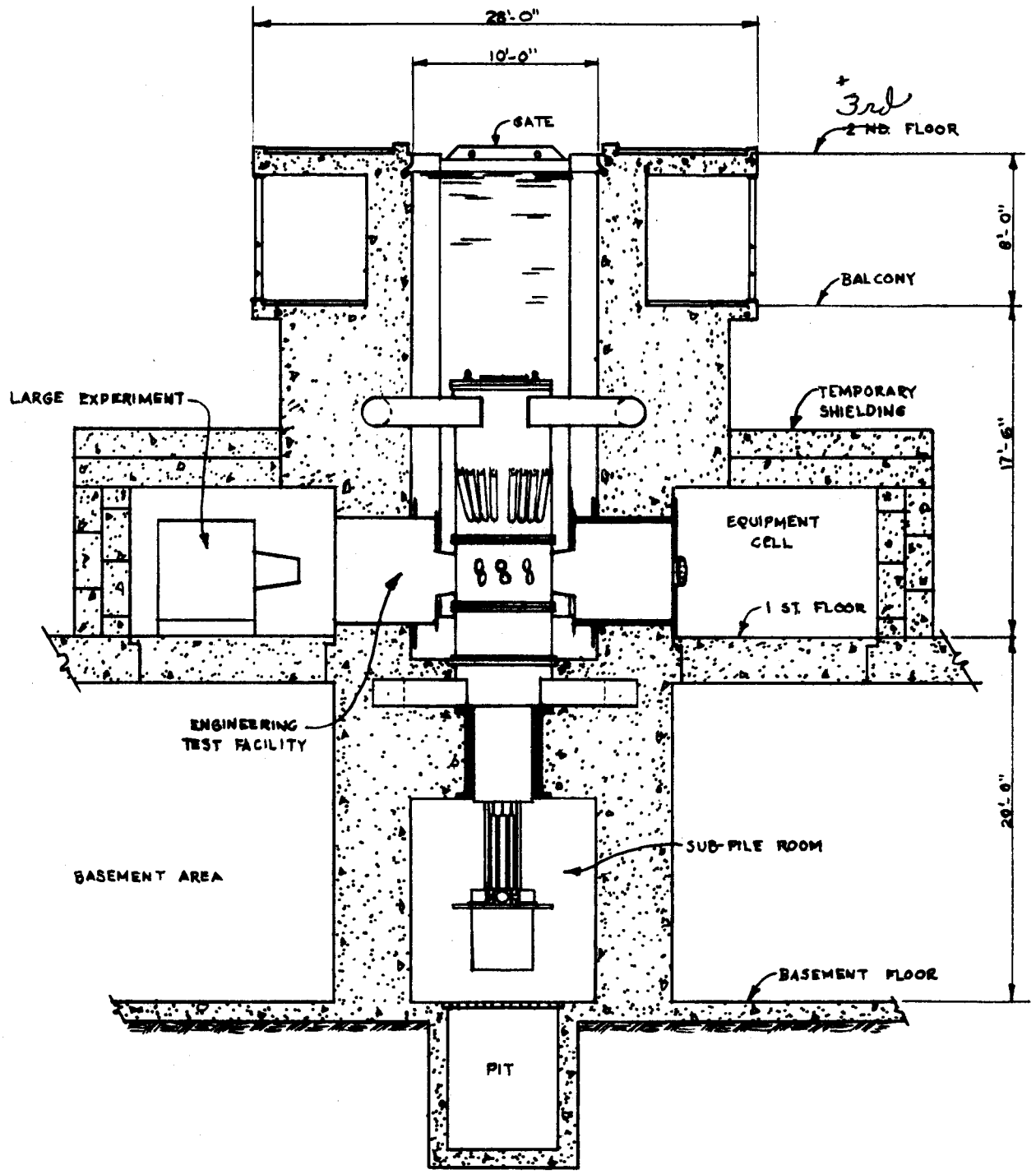

0 5 10 15 20

SCALE IN FEET

REACTOR STRUCTURE - N-S VERTICAL SECTION

ORR PROJECT

FIG.-4

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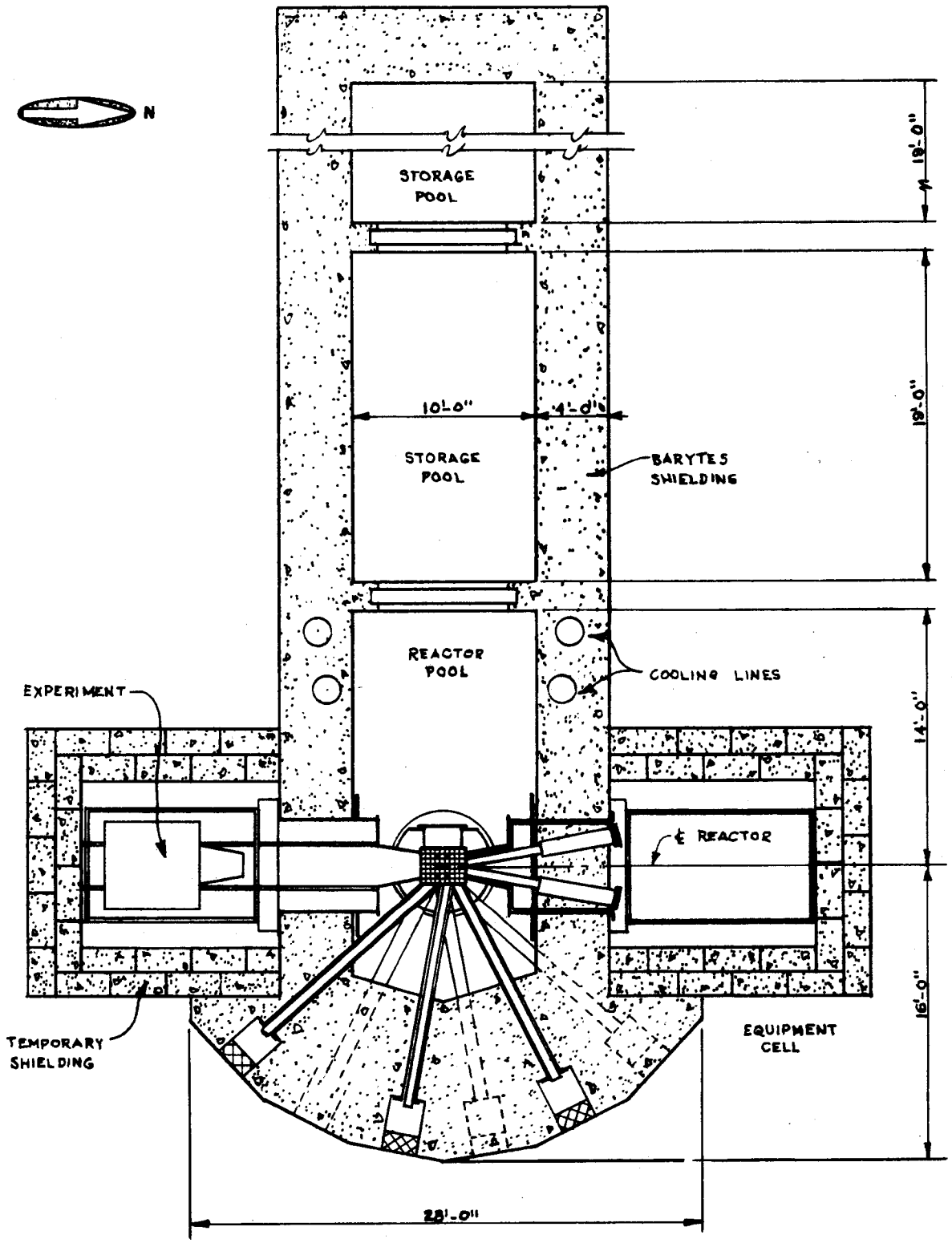


REACTOR AND POOL STRUCTURE

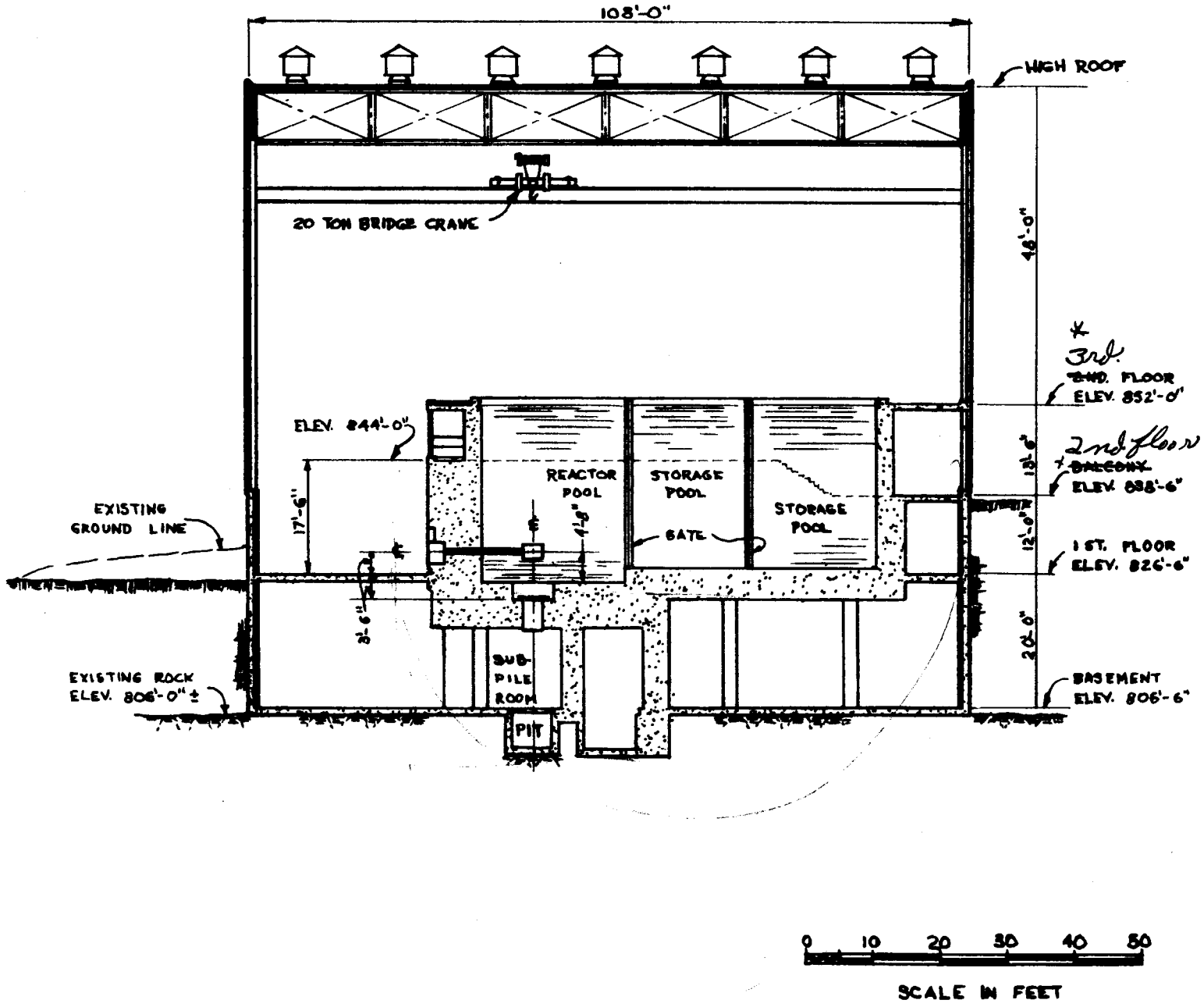
HORIZONTAL SECTION

ORR PROJECT

0 5 10 15 20

SCALE IN FEET

FIG.-5



SECTION AT ℄
ORR PROJECT

FIG.-6

+ See errata

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closest to the reactor will be protected by a thermal shield to prevent excessive heating.

3.b REACTOR SITE

The site proposed for the ORR is located in the north central portion of the Oak Ridge National Laboratory as indicated in Fig.7. This location is approximately 650 feet south of the nearest periphery fence and is inside the exclusion area fence which limits access to the reactor to those with legitimate business in the area. Other buildings in the general vicinity (see Fig.8) are as follows:

Building	Location
Bulk Shielding Facility (3010)	80' NE
X-10 Graphite Reactor (3001)	110' W
LITR (3005)	80' N
Rolling Mill (3012)	125' E

The building housing the ORR is to be located on the side of a hill which slopes in an easterly direction from elevation 835' to a small creek at elevation 800', approximately 350' east of the reactor site. This particular location was chosen because of the availability of services already in the area, integration of operations with currently existing similar facilities, and the availability of space for future expansion and for conducting neutron experiments at some distance from the neutron source. The cooling system for the pool water is located * east of Bldg. 3010, the Bulk Shielding Facility ~~adjacent to the NE corner of the building~~, and the cooling system for the core is located approximately 300' NE of the reactor building as shown in Fig. 8.

3.c REACTOR CONTROLS

The reactor control and safety system resembles the system installed on the MTR[2],[6]. Fewer channels of instrumentation will be installed,

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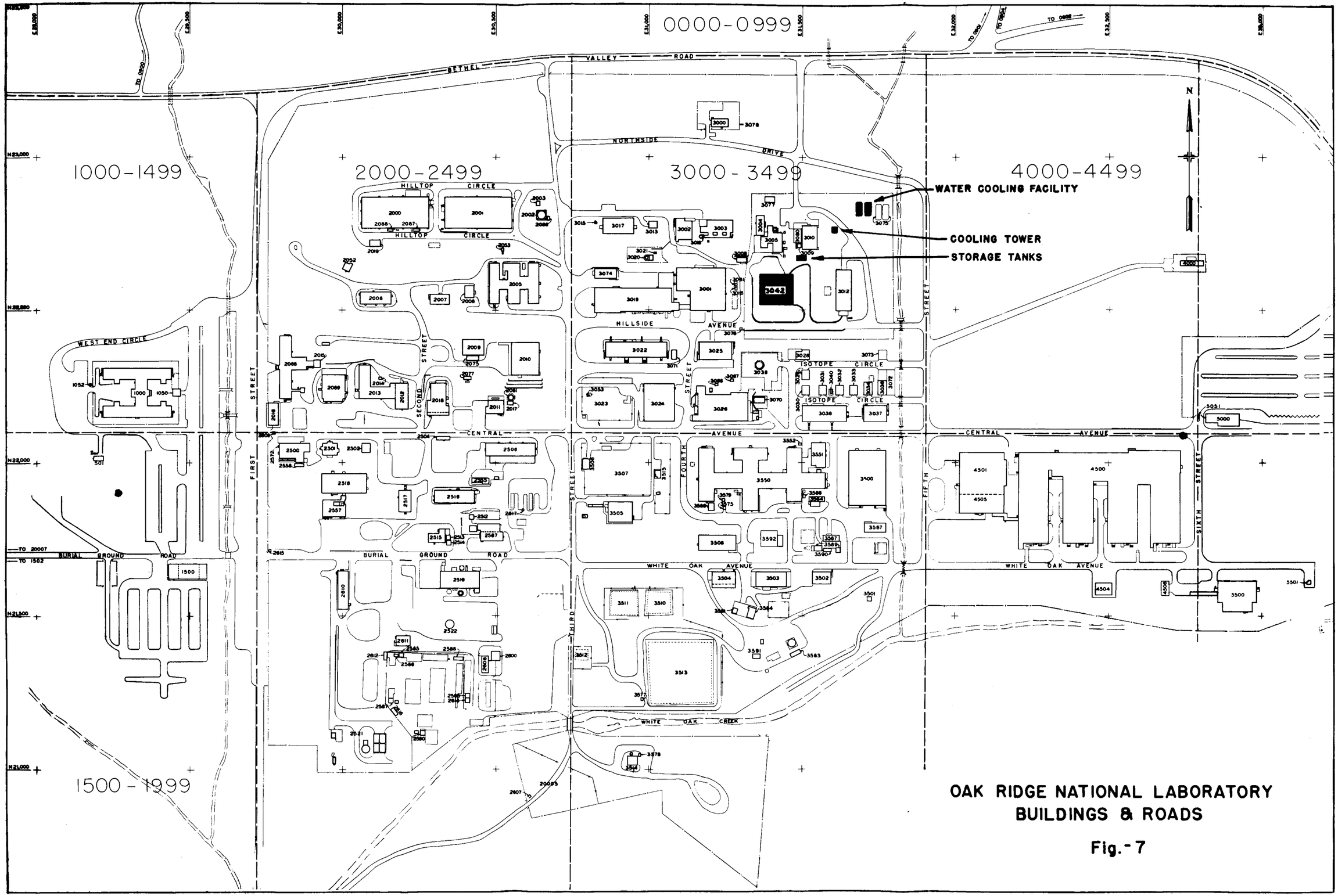

OAK RIDGE NATIONAL LABORATORY
BUILDINGS & ROADS

Fig.-7

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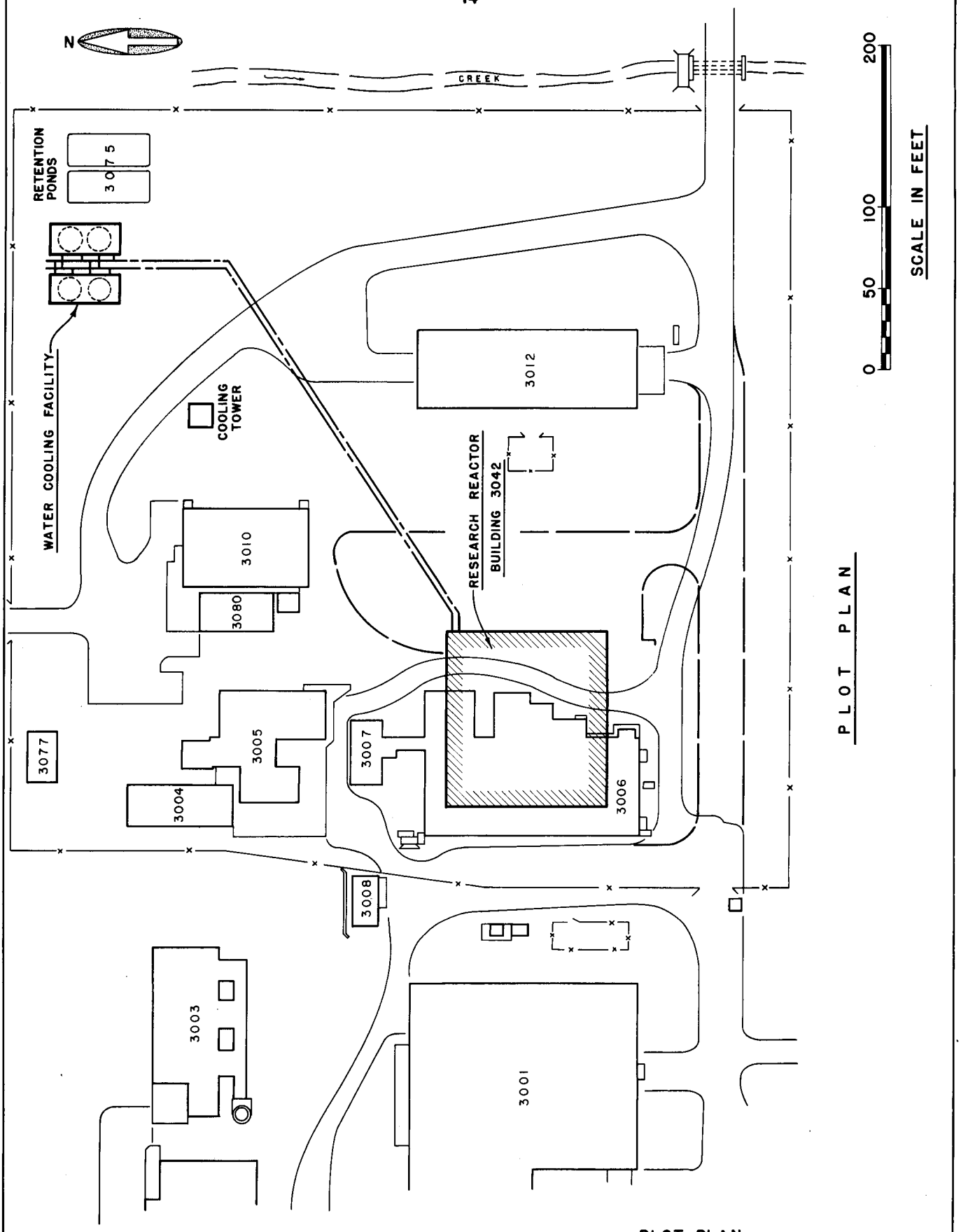

PLOT PLAN

O. R. N. L. RESEARCH REACTOR BUILDING 3042

OAK RIDGE NATIONAL LABORATORY

FIG.-8

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since considerable experience and confidence in the instruments have been gained since the first installation, and since the consequences of an accidental shutdown will be less costly on the ORR than on the MTR. The basic philosophy of control involves an operator, who supervises the actions taken by the control system, which in turn supervises the action taken by the operator. The Level Safety System is counted on to shut the reactor down in case of malfunction of operator or controls, but considerable effort is made to forestall a scram caused by an operating error or a false signal by equipping the control system with less drastic modes of corrective action, which can correct many potentially dangerous situations before a scram becomes necessary.

Reactivity is controlled by means of four movable control rods. For initial operation these will consist of four shim-safety rods, similar to those in the MTR[2]. These rods contain cadmium poison in the upper section and fuel in the lower section. The lattice is arranged to permit up to twelve shim-safety rods to be installed; for any given loading a suitable number of rods will be installed at locations appropriate to the fuel disposition. Rod drive mechanisms, located in the sub-pile room, will be installed so as to correspond with the selected rod positions. The rod drives terminate in quick release mechanisms which engage sockets on the rods as shown in Fig. 9. This mechanism was developed jointly by the North American Aviation Corporation and the Oak Ridge National Laboratory for use on the proposed UPR and the ORR. A more detailed description of the release mechanism and the results of tests performed at the Oak Ridge National Laboratory to prove the reliability and speed of response is given in the

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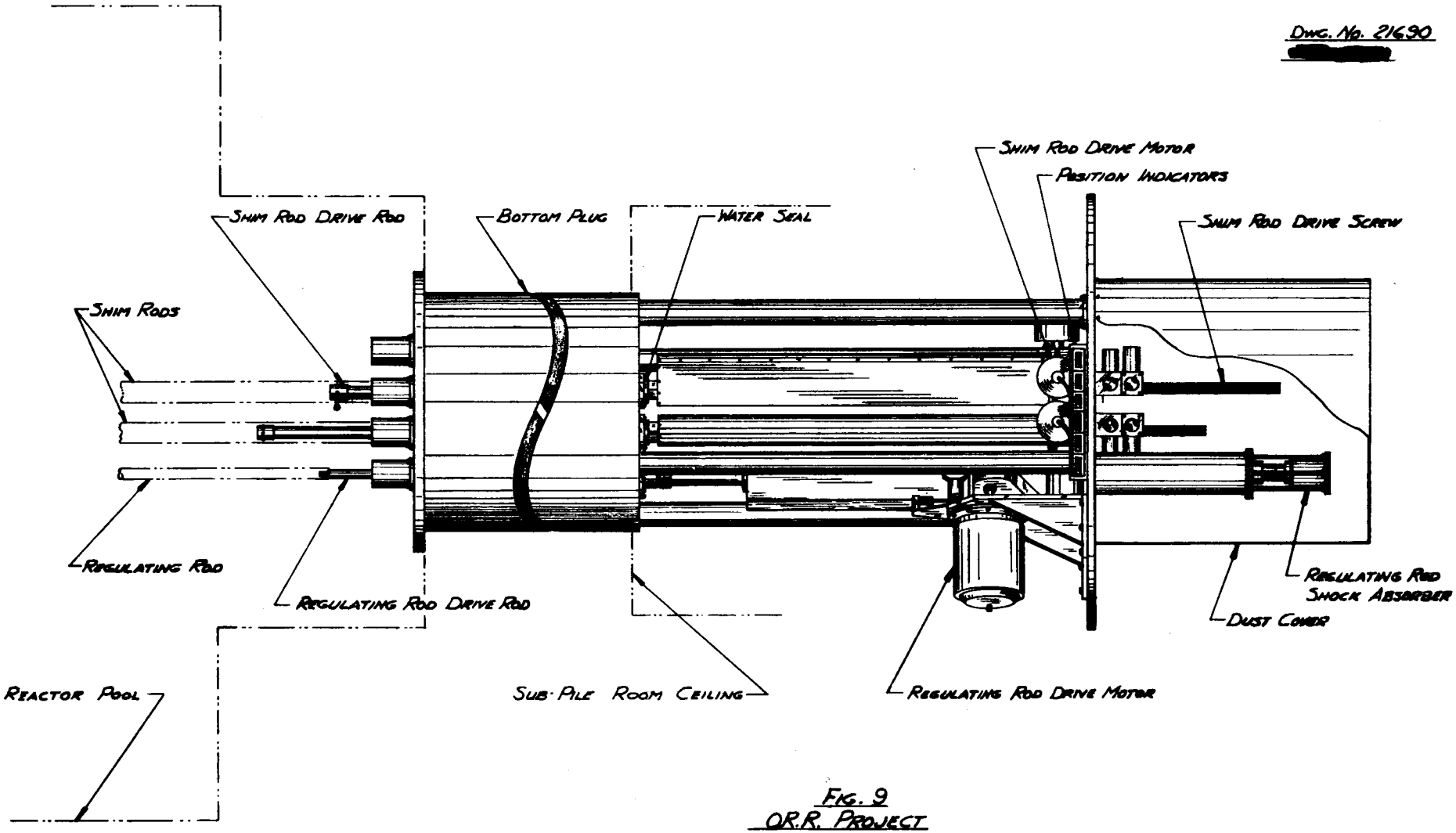
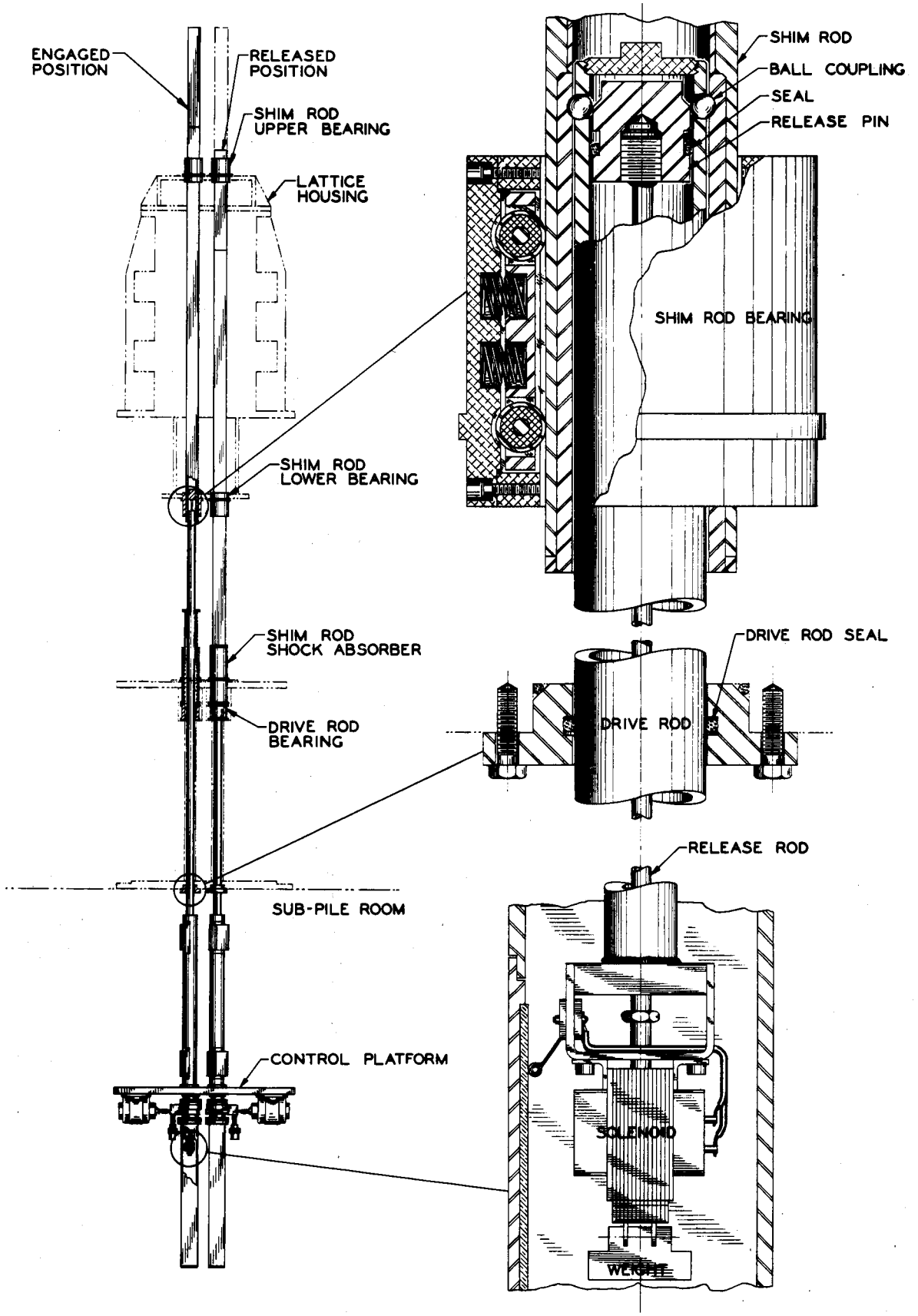


Fig. 9
O.R.R. Project
Control Drives Assembly



ORNL RESEARCH REACTOR
SHIM ROD RELEASE MECHANISM

SK-123 [redacted]

Fig. 9a

appendix.

The release mechanism operating solenoid is connected to the reactor safety system as in the LITR and MTR[6]. Rod motion, other than scram insertion, proceeds from a drive motor operated from the a-c line. Alternating current motors have been specified in order to minimize the possibility of greater than normal speeds. Single phase power will be used throughout to avoid potential troubles encountered by opening one wire of a polyphase line.

Four limit switches are provided on each shim-safety rod as follows:

(1) Upper limit on drive mechanism.
(2) Lower limit on drive mechanism.
(3) Clutch switch showing aspect of release mechanism.
(4) Seat switch showing whether or not the rod is actually seated.

In addition to these limit switches, a double synchro pair is provided for each rod, the coarse synchro to rotate 270° for the full rod travel of 30"; the fine synchro to rotate 360° for each inch of travel. It will be possible to indicate the position of the rod drive mechanism to the nearest 0.01 inch and still avoid any ambiguity as to rod position.

The four rods control a $\delta k/k$ of 34 °/o in a 5 X 5 or 3 X 9 lattice[7]. The maximum rate of addition of reactivity is 0.1% per second and occurs when the rods are simultaneously withdrawn at their maximum speed of five inches per minute[8].

Nuclear information is transmitted to the operator and to the control and safety systems from several channels as follows:

(1) Level Safety. Three PCP chambers supply current to three Level Safety Channels[6].

(2) Period. One compensated chamber supplies current to a Period Channel[6].

[redacted]

(3) Linear. One compensated chamber supplies current to a Leeds and Northrup Micromicroammeter.

(4) Counting Rate. One fission chamber supplies pulses to a Linear Amplifier, Pulse Height Selector, Scaler and Log Count Rate Meter.

The three types of chambers designated above are to be of the same type as those currently in use on other ORNL reactors. Both the PCP and compensated chambers will be installed in a manner similar to those in use on the Tower Shielding Facility, that is, "sealed" chambers which do not require a flow of gas. The fission chamber will be of the Westinghouse sealed type as used on the STR and TSF.

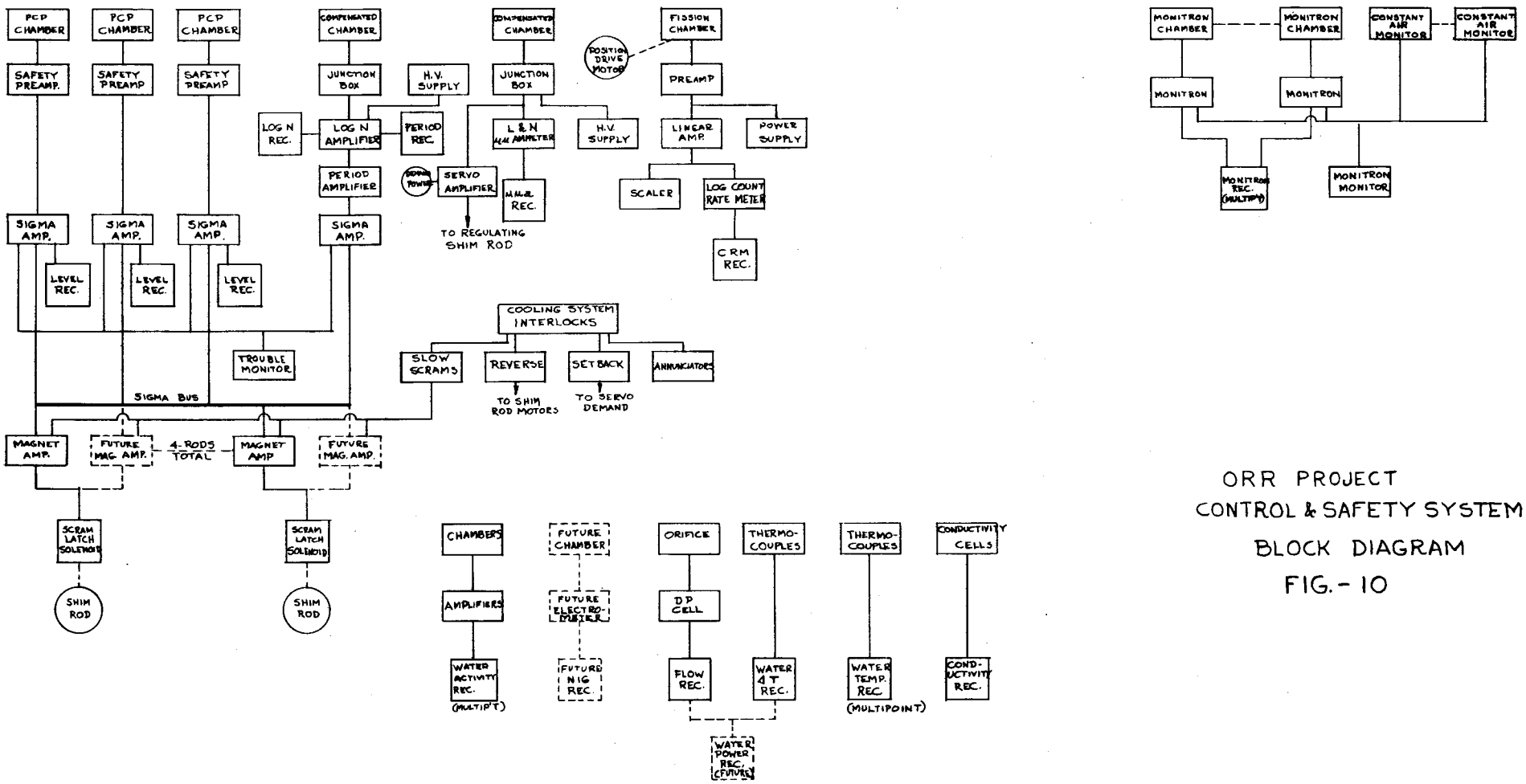
Control of the cooling system is accomplished through the use of conventional instrumentation as regards parameters such as flows, temperatures and pressures. Radiation detectors are provided to give warning of contaminated water.

It is planned to provide a servo system to maintain the flux at constant level, using one of the shim-safety rods as a regulating rod. The δk available to the servo for control purposes will be limited to a safe amount, approximately 0.5% $\delta k/k$. The rate of rod motion initially will be identical with that available to the shim control, although it may be desirable to change the maximum rate for the servo-controlled rod by a factor of as much as two in one direction or the other. A maximum rate of about 0.05% $\delta k/k$ per second is the highest now contemplated.

A block diagram of the control and safety systems is given in Fig.10.

3.d EXPERIMENTAL FACILITIES

Figures 4 and 5 are sections through the reactor structure showing the arrangement of the major experimental facilities. Six 6" diameter holes



ORR PROJECT
CONTROL & SAFETY SYSTEM
BLOCK DIAGRAM
FIG. - 10

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penetrate the biological shield to one face of the lattice and will be used mainly for the semi-permanent type beam experiments, in which the length of the beam hole is not a major consideration and where the bulk of the equipment can be mounted on the floor area, the cubicle, or the face of the shield. Examples of such experiments are: neutron or gamma ray spectrometers, neutron velocity selectors, neutron diffraction, nuclear alignment studies, etc. Shielding in this region consists of a minimum of six feet of water and nine feet of barytes concrete.

The two large penetrations to the reactor lattice are designated as "Engineering Test Facilities" and are designed for a number of anticipated installations. Two large water tight bulkheads approximately five and one half feet in diameter pierce the shield to a point just outside the reactor tanks opposite the lattice, where they are reduced in size and tapered to a diameter of 18" at the lattice face. Initially these holes will house water tight dummy plugs which will duplicate the main shielding in this vicinity and will contain two smaller experimental facilities, each eight inches in diameter, and eight feet long. These facilities are being designed specifically to meet the needs of the type experiment now required for the fluid fuel reactor loop programs, where it is desirable to have the plug diameter as large as possible and the length of the plug as short as possible.

The cell immediately outside of the engineering test facilities will be erected as required for the shielding of the auxiliary equipment required for the experiment and will have provisions for service connections from the basement area and for instrumentation and control connections at the

balcony level. Space is provided for the construction of disassembly cells to be located either on the main floor or the basement floor. Sufficient permanent shielding is to be provided so that the equipment cells can be entered during shutdown periods, provided the type of experiment is such as to allow entry.

Removal of the beam hole plug from either side of the reactor will allow for the insertion of what may be termed "major engineering experiments", in which all the primary components for the test such as the fuel loop, pumps, heat exchangers, etc., are to be exposed to the high flux region of the reactor. In this proposal, the reactor would be completely unloaded, the original plug placed in a storage cell and the experiment inserted as a package. It is believed that with the proper selection of construction and shielding materials, primarily aluminum and chemically pure lead, in both the reactor and facility design, and with fuel and reflector removed to a storage pool, insertion of these experiments can be accomplished with a minimum of remote handling equipment.

By locating the control rod drive mechanism below the reactor, it is possible to consider insertions into any of the sixty-three possible fuel or reflector spaces, depending upon the lattice arrangement. Small experiments which require outside connections can be inserted in tubes and guided into the reactor through sleeves located in the top plug. Storage of these tubes on the side walls of the pool during refueling operations can be accomplished without breaking the external instrument or service connections. Larger experiments such as small fuel loops can be inserted in such a manner that they can remain in the reactor during the refueling operation. The con-

nections here are to be made through flanges on the top cover plate.

The fourth face of the reactor will provide initially for three vertical experimental facilities six inches square with access through the top plug. In addition this face can be converted into a major facility for the installation of experiments from the pool side, for shielding studies, a thermal column, engineering type tests, etc.

It is also planned that two hydraulic and two pneumatic type rabbit tubes will be included in the design of the installation.

3.e WATER SYSTEMS

The ORR water system can be divided into two essentially separate units; the reactor cooling loop and the pool system. Both systems are to be constructed of 63S aluminum pipe with 3S welded or flanged fittings. All-iron valves will be used where valve life is not essential to the continued operation of the system; in the more strategic locations, aluminum valves of 356 cast aluminum alloy with 316 stainless steel trim will be used. Cast iron pumps with stainless steel trim will be used in both systems. Heat exchanger tubing will be 3S aluminum appropriately clad with 72S aluminum. Large storage vessels and large diameter pipe will be fabricated of 54S aluminum. A flow sheet of the water system is given in Fig. 11.

THE REACTOR COOLING LOOP

The reactor cooling loop is designed to circulate a minimum of 2500 g.p.m. of demineralized water through the reactor and air-cooled heat exchanger for operation at five megawatts. Higher rates of circulation are provided for in the form of additional pumping equipment for use at higher power levels and as standby pumping capacity. The design inlet temperature

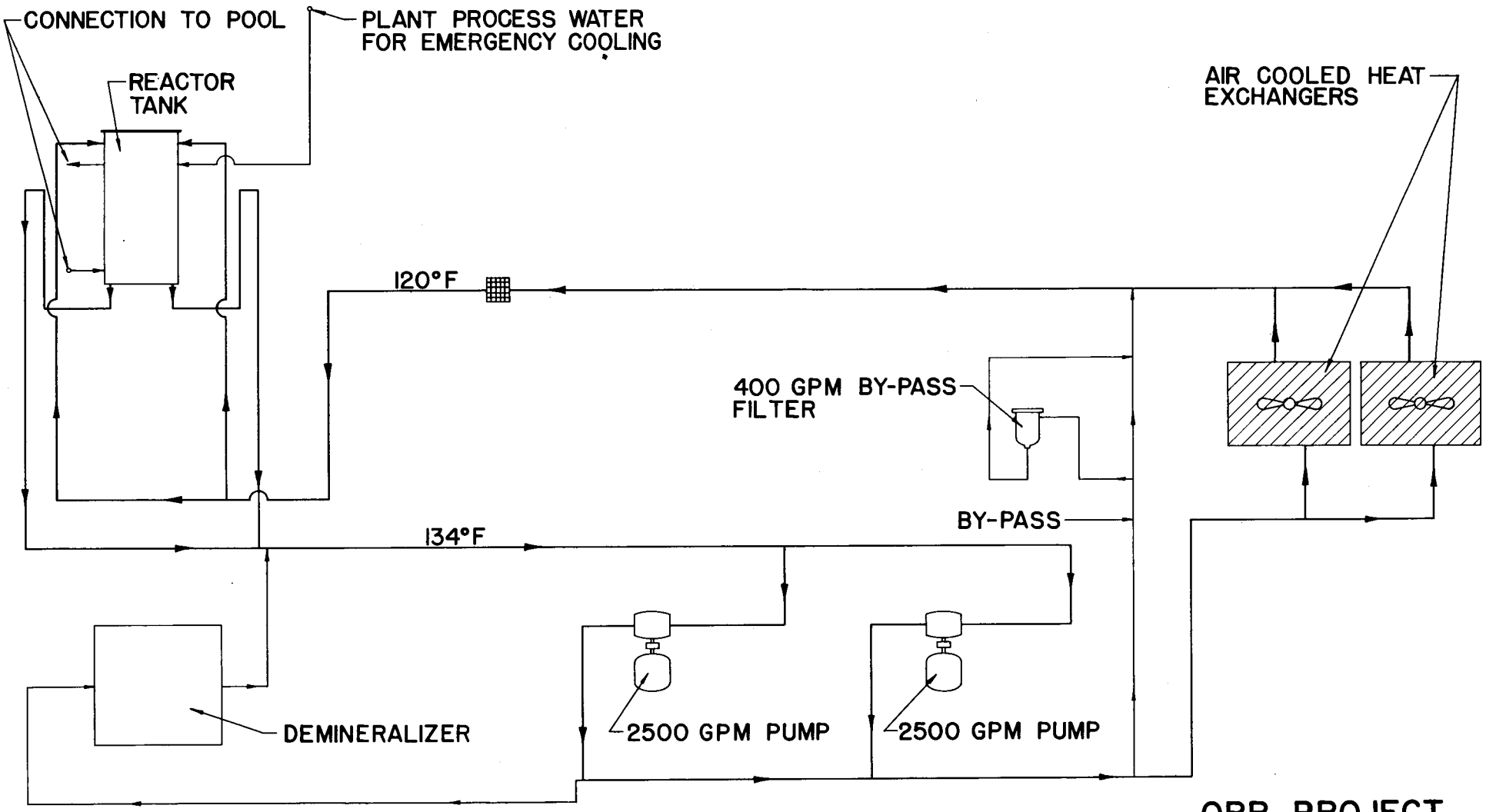

Fig.-11a

ORR PROJECT
COOLING LOOP
FLOW DIAGRAM

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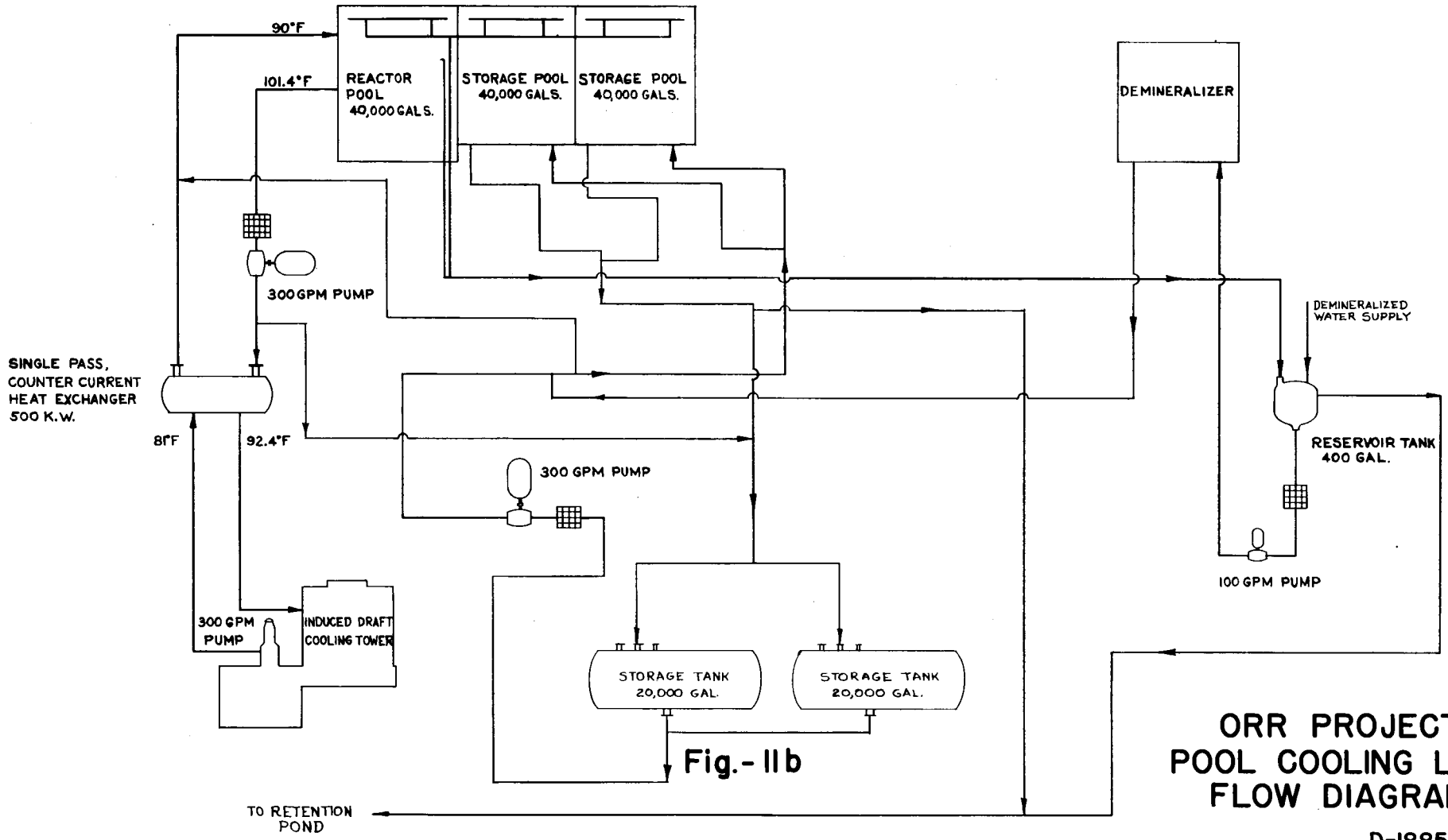

Fig.- 11b

ORR PROJECT POOL COOLING LOOP FLOW DIAGRAM

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to the reactor is 120° F and the outlet temperature is 134° F. These operating conditions are established in ORNL-CF-54-3-159 (see appendix) and are designed to allow a maximum temperature of 200° F in the water in contact with the fuel elements. Since the water pressure in the lattice will be equivalent to the head maintained by the surrounding pool (21 ft.) local boiling in the coolant is excluded.

The main circulating pumps, located in the basement of the building, draw a flooded suction on the bottom of the reactor lattice. Two 18" diameter pipes convey water from the bottom reactor tank to the pumps. Each of these pipes has a vertical loop extending approximately ~~two~~ * three to four feet above the reactor lattice. These pipes and loops are encased in the concrete of the pool wall and floor. A one inch line connects the top of the loop to the top reactor tank, thus forming an air break on the suction leg of the loop in the event of a pipe rupture. These 1" lines are also connected to the pool through a 3" overflow line * terminating 3-4 feet above the reactor lattice and to the off-gas system through a ball-float trap located near the surface of the pool. Thus the pool acts as an expansion chamber for the reactor cooling loop and as a standpipe establishing the static pressure at the exit from the reactor as the height of the pool at this point. During steady state operation there will be little net interchange of water between the systems.

The two 18" outlet lines merge into a single 24" line which leads to the main circulating pumps in the northeast corner of the building. The two main circulating pumps are designed so that each one can deliver 2500 g.p.m. when they are operated simultaneously. At five megawatts only one pump need be operated. Both pumps are required for ten megawatt operation. These

*See errata

pumps are located in cells of stacked concrete block, each having a ceiling either of poured barytes concrete two feet thick * or of ordinary concrete three feet thick. This ceiling forms a gallery on which the auxiliary equipment, pool piping and pool pumps are located. In the cell with one of the main circulating pumps is an auxiliary pump for circulation of a portion of the reactor cooling water through the experimental facilities * and cooling loop demineralizer; in the cell with the other pump is a line by-passing the air-cooled heat exchangers together with a large capacity (400 g.p.m.) bypass filter. Provision for installation of a third main circulating pump at some future date is also made.

Water from the main circulating pumps is discharged into a twelve inch line which leads through an earth-filled trench to the air-cooled heat exchangers which are located approximately 300" NE of the reactor building and are sufficiently isolated that they may be fenced off to prevent access during operation.

The water returns from the heat exchangers in another twelve inch pipe entering the building in the vicinity of the pump cells. In this area is located the orifice for the measurement of flow to the reactor and a strainer to protect the reactor from solids, such as scale particles, in the circulating stream. At a point upstream from the main orifice a small portion of the flow (300 g.p.m. maximum) is diverted for use in cooling the experimental facilities.

After the strainer the 12" line expands to a 24" pipe which lies beneath the basement floor and leads to a pipe room under the reactor pool. At this point the 24" pipe branches into two 18" lines which lead through the reactor pool structure to the upper reactor tank.

* See errata

In order to maintain the desired purity in the reactor cooling loop, a small amount of water (approximately 20 g.p.m.) is bypassed through a mixed-bed demineralizer which is composed of a cation resin similar to Rohm and Haas Cation Resin IR 120 and a strong basic anion resin similar to Rohm and Haas Anion Resin IRA 410. This bed will produce effluent of approximately 1.5×10^6 ohm resistivity. The unit is designed for a maximum flow of eighty gallons per minute for rapid purification in case excessive contamination develops. Waste from the ion exchange columns is led to the plant "hot" waste system.[12]

The reactor cooling system is illustrated schematically in Fig. 12.

THE POOL SYSTEM

The pool surrounding the reactor is an aluminum-lined concrete basin 10' wide, 60' long, and 28' deep. This pool is divided into three sections, each twenty feet long, by removable aluminum gates as shown in Fig. 6. The first pool contains the reactor and associated facilities. The remaining two sections are designed for the storage of experimental facilities and used fuel assemblies.

The pool containing the reactor, experimental facilities, and reactor cooling loop piping will be heated not only by conduction of heat from the exposed surfaces of the reactor tanks and cooling loop piping, but also by the absorption of gamma radiation from the core. For five megawatt operation the heating due to gamma absorption has been estimated to be 200 KW. This heating is linear with power level. For the design cooling loop temperatures and a pool temperature of 85° F, the heat transferred from the exposed surfaces has been estimated to be 300 KW. This heat transfer is

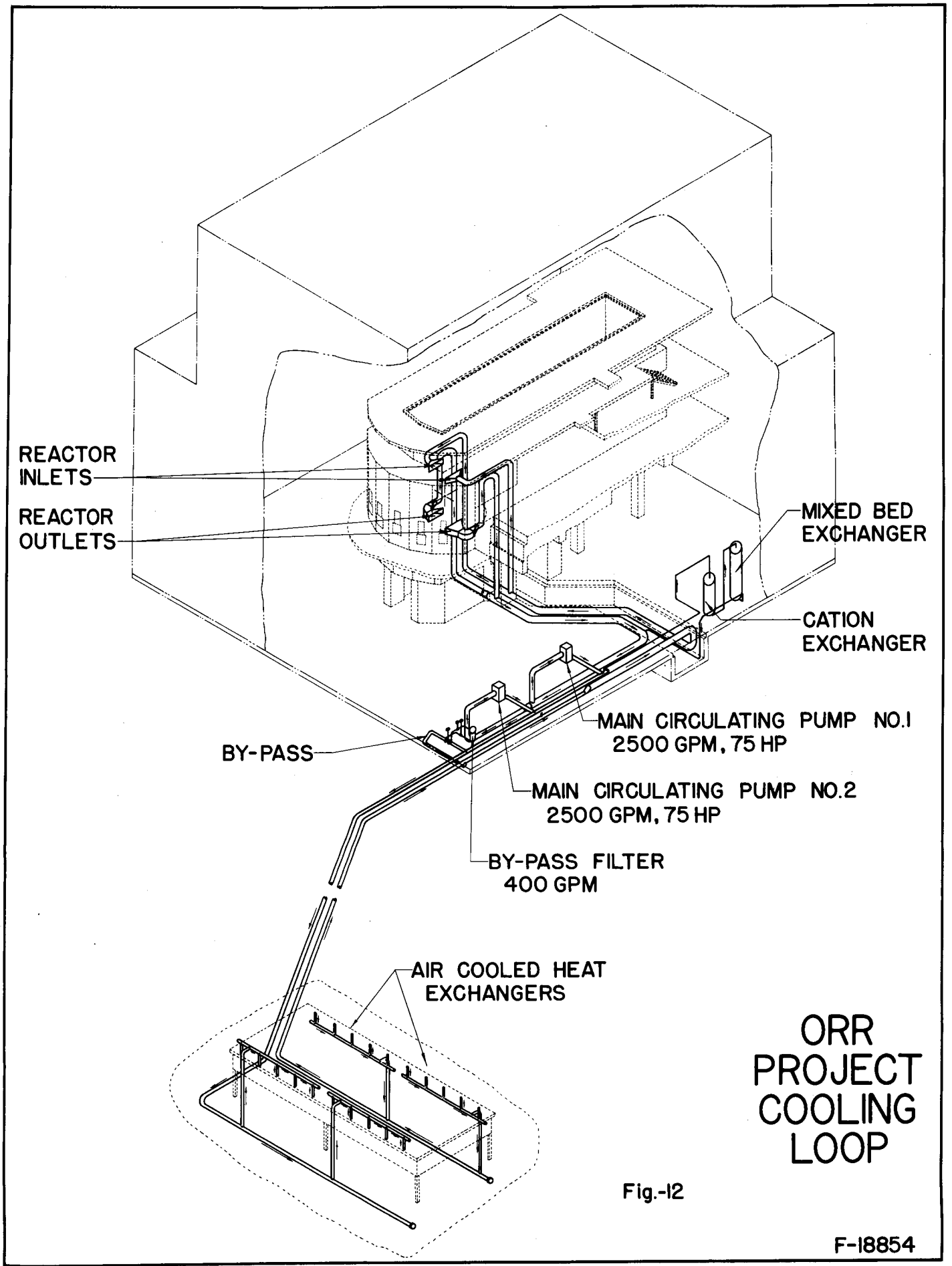

Fig.-12

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essentially independent of power level.[9]

Since the ambient humidity in the building is rather insensitive to changes in pool temperature, particularly when high humidity conditions exist in the inlet air to the building[10] a temperature of 100° F has been chosen as a practical design temperature for pool operation. In order to maintain this temperature, 300 gallons of water per minute is circulated through the tubes of an aluminum, U-tube, shell and tube heat exchanger. Through the shell an equal quantity of plant process water is circulated. This plant process water is in turn cooled in an induced-draft spray tower. The design temperatures are as follows:

Outlet from pool	101.4° F.
Inlet to pool	90° F.
Outlet from spray tower	80° F.
Inlet to spray tower	91.4° F.
Ambient wet-bulb (design)	76° F.

All connections for either inlet or outlet streams in the reactor section of the pool are above the level of the top of the reactor tank. Temporary piping must be installed in the event this section is to be drained directly. The remaining two sections of the pools have inlet and outlet lines located on the floor of the pool. The four inch inlet lines are protected from inadvertent draining by check valves immediately below the pool, and the gates between the pools are built in two sections. The bottom section will normally be in place during operation. The two inch outlet lines are sized to prevent rapid draining in case a drain is inadvertently opened.

Both of the storage sections and the upper portion of the reactor section of the pool have outlet lines leading to two 20,000 gallon aluminum storage tanks located outside of the building. Water from either of the storage tanks may be pumped into any of the pools or into the reactor

cooling loop. This same pump, by an alternative valving arrangement, can be used to facilitate draining of the pool. The pool can also be drained directly to a pair of 20,000 gallon retention ponds which are connected to the plant "warm waste" disposal system.

Connected to the fill line on the two storage pools and the reactor pool is a 2" line from the pool demineralizer. By an appropriate valving arrangement, this flow can be directed to any of the pools or to any combination of the pools. The pool system demineralizer is a two-bed unit similar to that used in the cooling loop. The maximum flow through this unit will be 100 g.p.m. The effluent will have a resistivity of approximately 700,000 ohms.

The overflow from the pools will be collected in a trough at the pool edges. The flow is through a screen into a six inch overflow header leading to a 400 gallon aluminum reservoir tank located in the pump gallery. This tank will provide a flooded suction for a 100 g.p.m. pump conveying water to the pool demineralizer system. The reservoir tank contains a float operated valve in a line supplying demineralized water from the plant system. This supply provides makeup water to the pool system by maintaining a minimum level in the reservoir tank. Overflow of the pools in excess of the demineralizer flow runs into a 6" line from the reservoir tank to the retention ponds. A schematic diagram of the pool system is shown in Fig.13.

3.f BUILDING AND SERVICES

The facility will be housed in a structural steel framed, insulated metal panel sided, mill-type building 108 feet long, consisting of a central crane bay area 60' wide with a service wing on each side 20' wide.

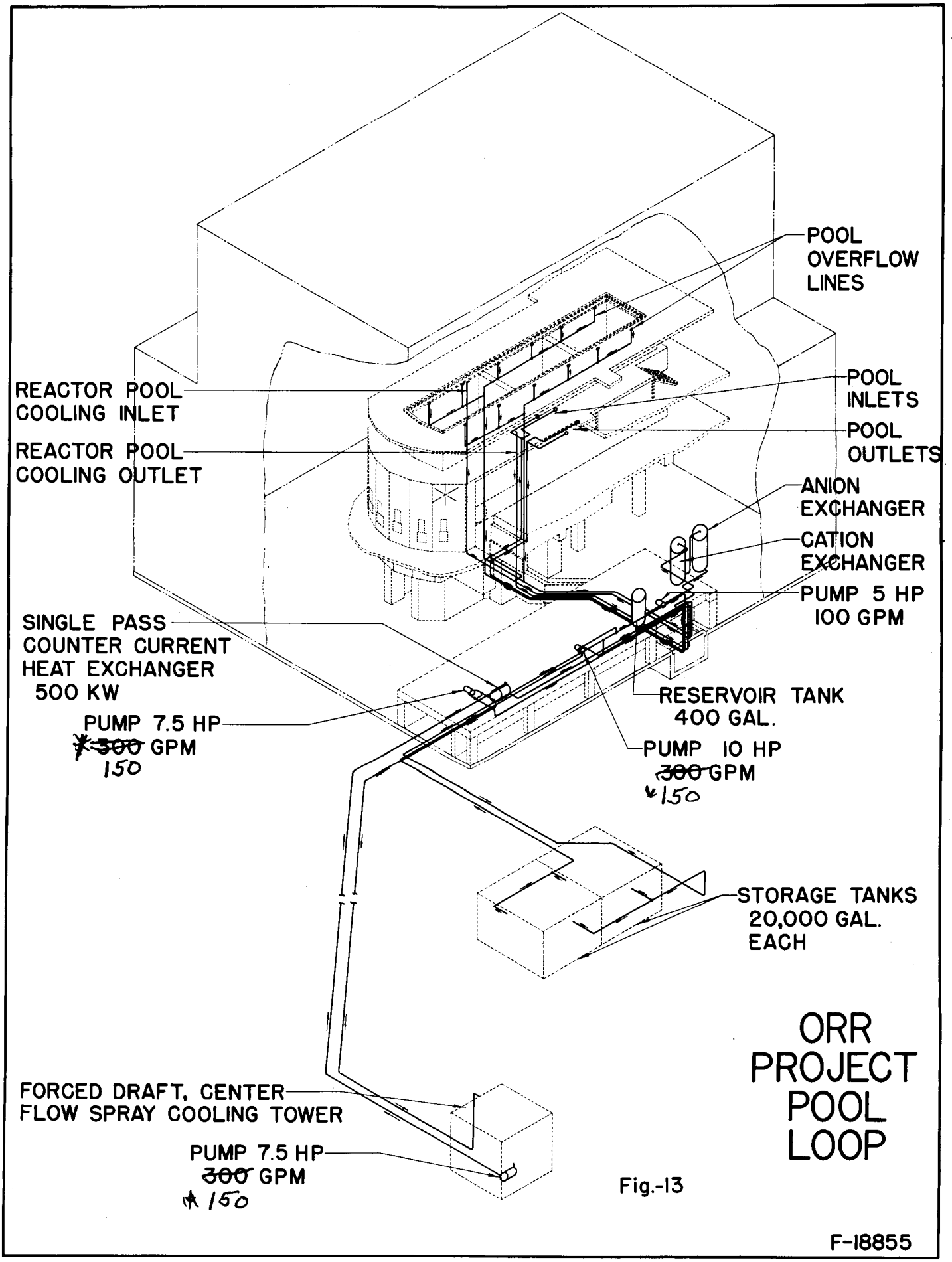

Fig.-13

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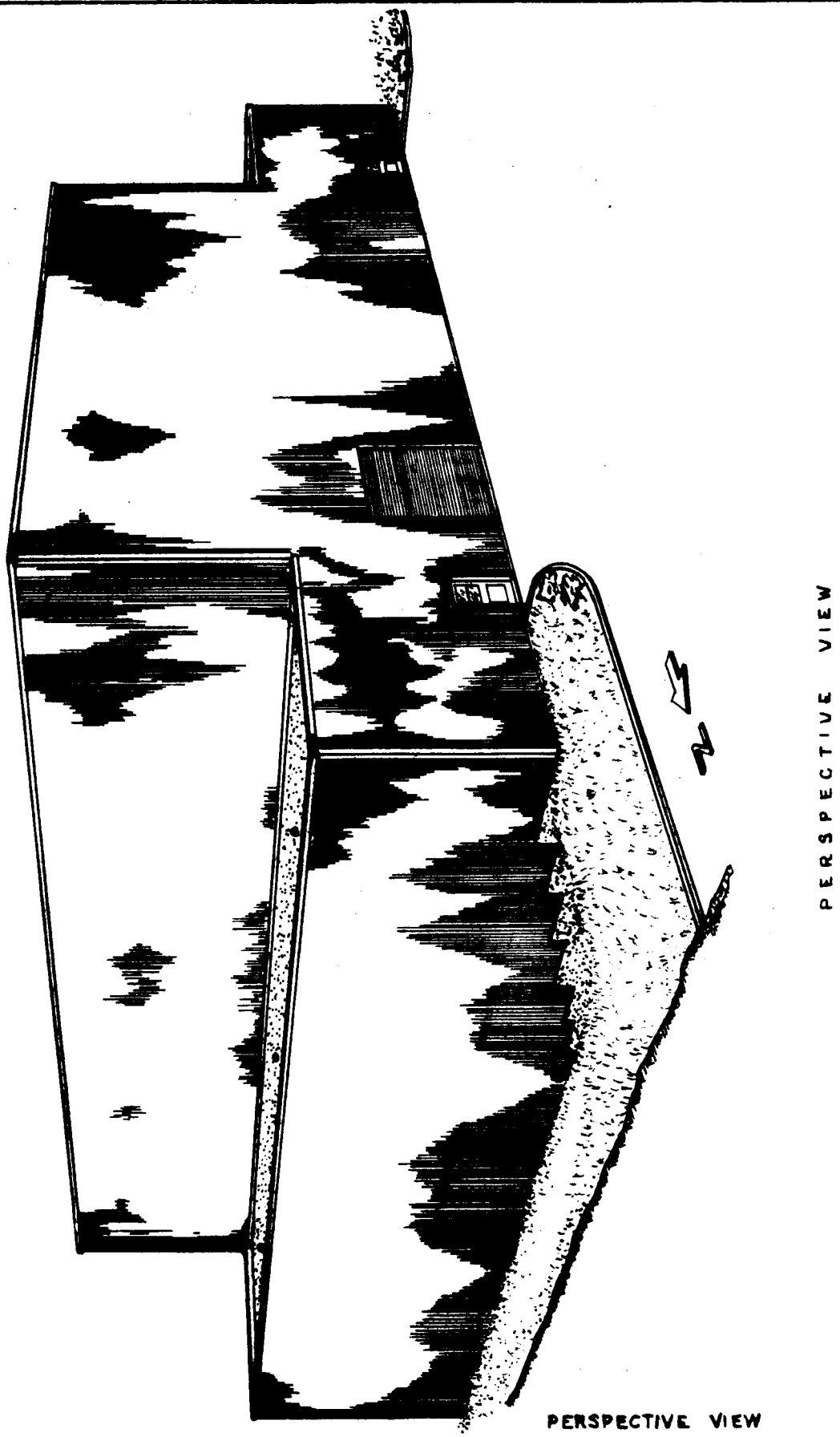
* See errata

The height of the mill type structure, measured from the first floor, (see Fig. 14) is about 36' for the service wings and about 71' for the crane bay. The area grade contour is such that a truck entrance to the building at street level has been provided at the second floor level on the west side of the building, and a second street-level truck entrance on the east side of the building at the first floor level. All construction below grade level will be reinforced concrete. The basement floor is 20' below the first floor level. Approximately 30' of shale and earth below the existing grade will be excavated to place the structure on a rock foundation.

The reactor pool structure will be of barytes concrete with walls of various thickness. Thick walls of reinforced concrete enclosing the basement sub-pile room and pipe chase support the reactor pool, and concrete columns support the storage pools.

The building will be divided into four working levels as follows: (1) A full basement of 11,400 square feet (see Fig. 15) which will include a fan and filter room, the sub pile room, shielded pump room and operating gallery, reactor sub-structure, and future cell areas. (2) A first floor of 9500 square feet (see Fig. 16) which will contain the reactor beam hole areas, general experimental area, and truck access. (3) A second floor of 3,000 square feet (see Fig. 17) which will contain truck access to the building, office~~s~~*, rest rooms, and balcony access around the reactor pool at an intermediate level 5'-6" above the second floor. (4) A third floor of 6,800 square feet (see Fig. 18) which will contain four offices, six laboratories, control room, equipment and instrument repair shop, access around the pool, and a small hot cell.

* See errata



PERSPECTIVE VIEW

ORR PROJECT

FIG.-14

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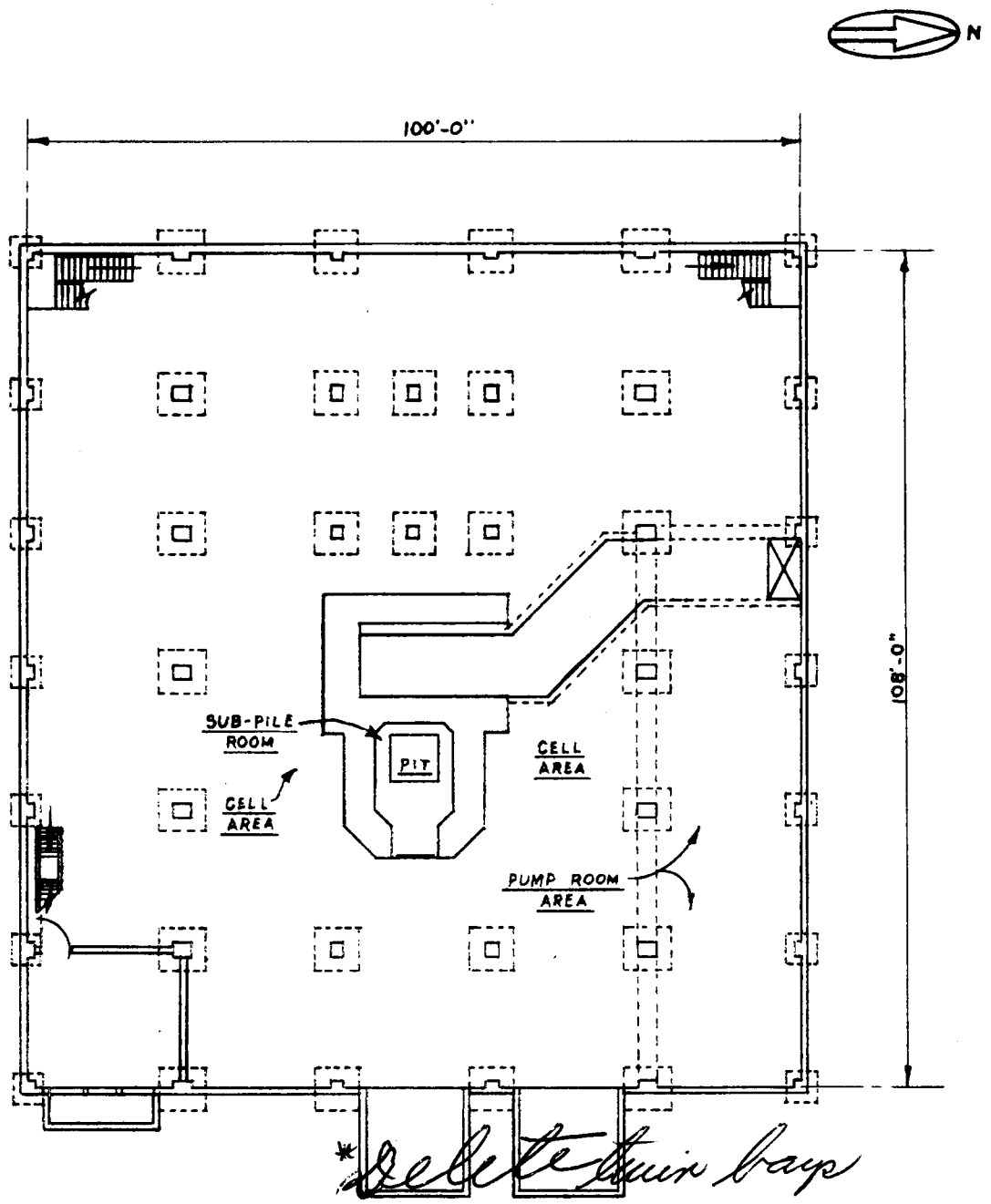

0 10 20 30 40 50

SCALE IN FEET

BASEMENT PLAN

O.R.R. PROJECT

FIGURE 15

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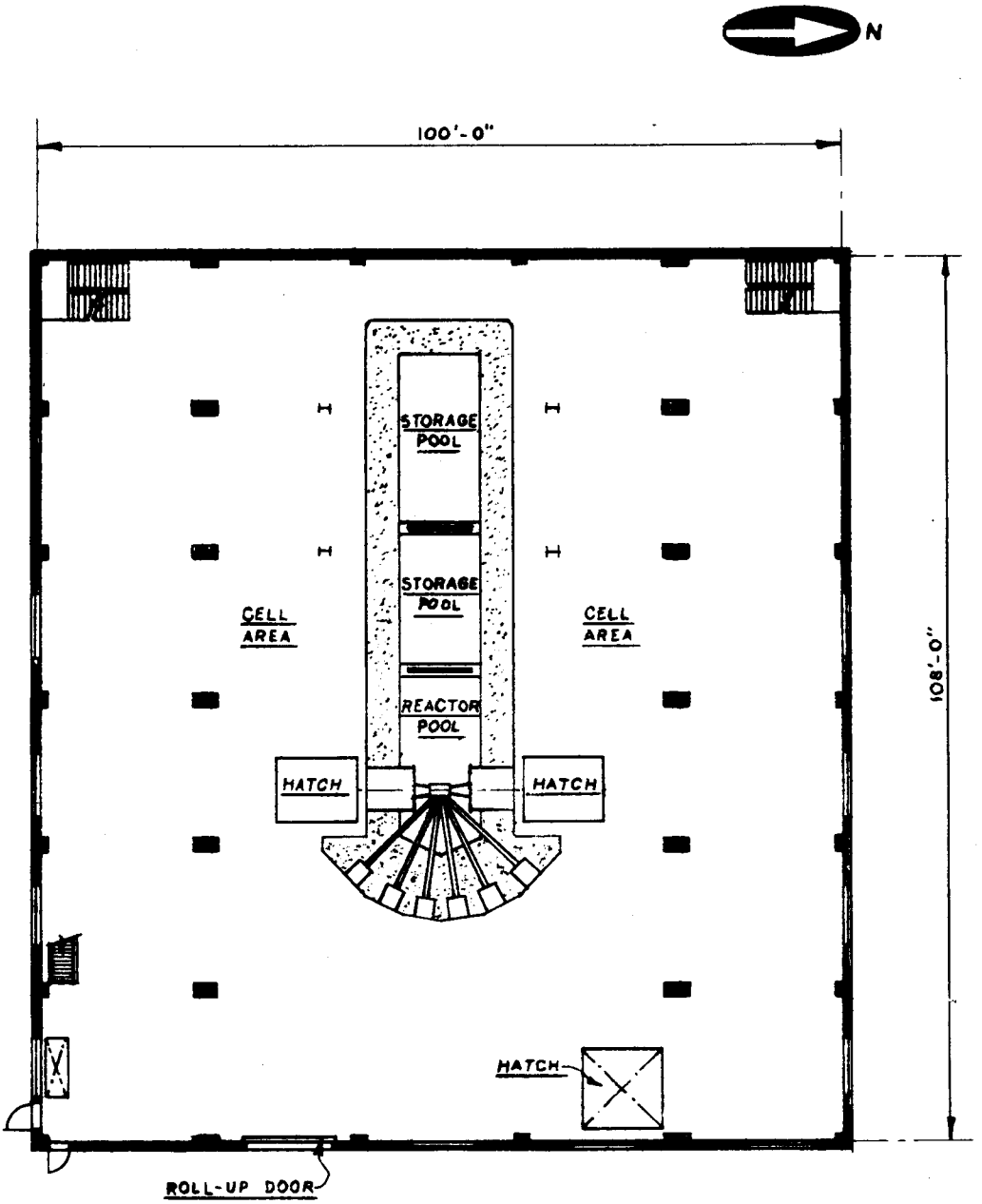
* See errata



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SCALE IN FEET

FIRST FLOOR PLAN

O.R.R. PROJECT

FIGURE 16

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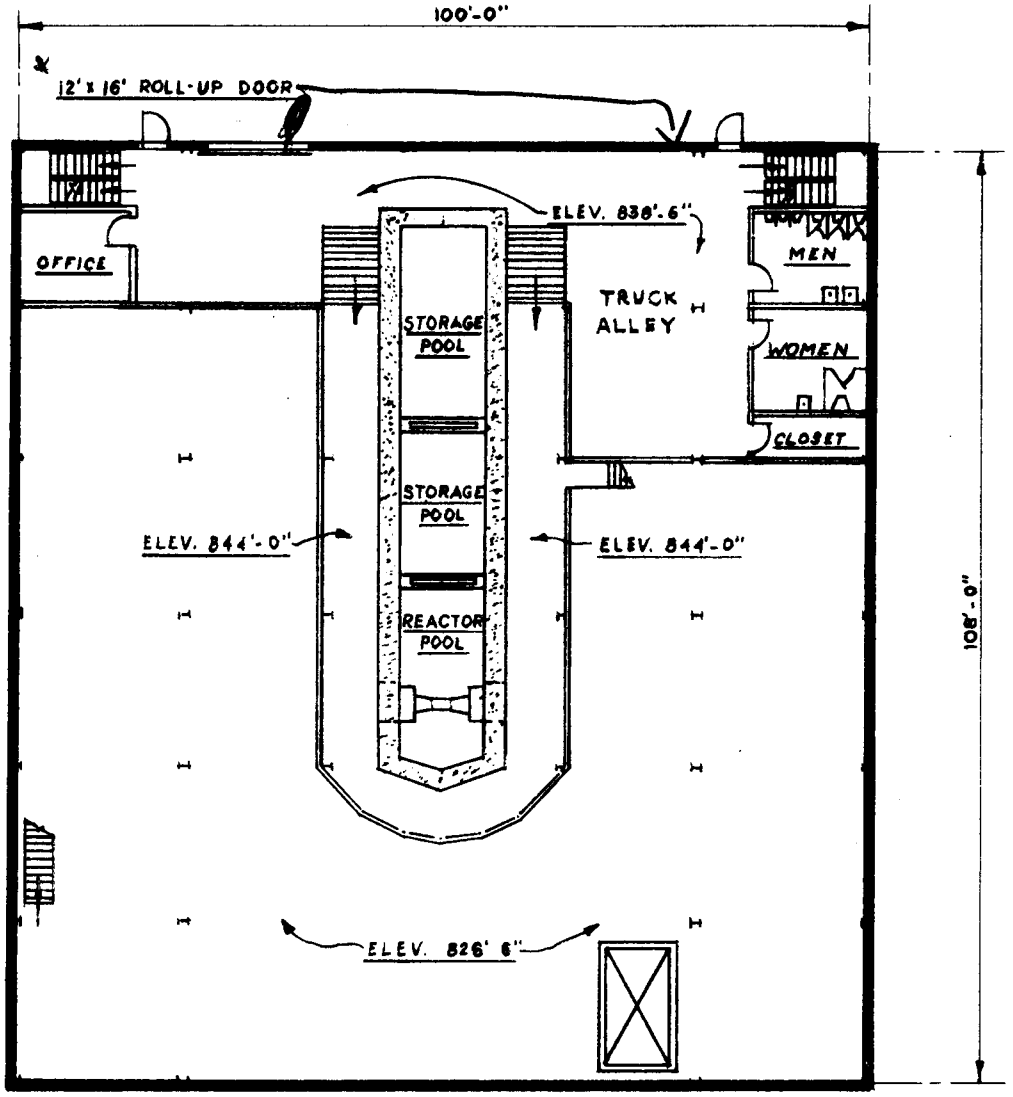

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SCALE IN FEET

SECOND FLOOR & BALCONY PLAN

ORR PROJECT

FIGURE 17

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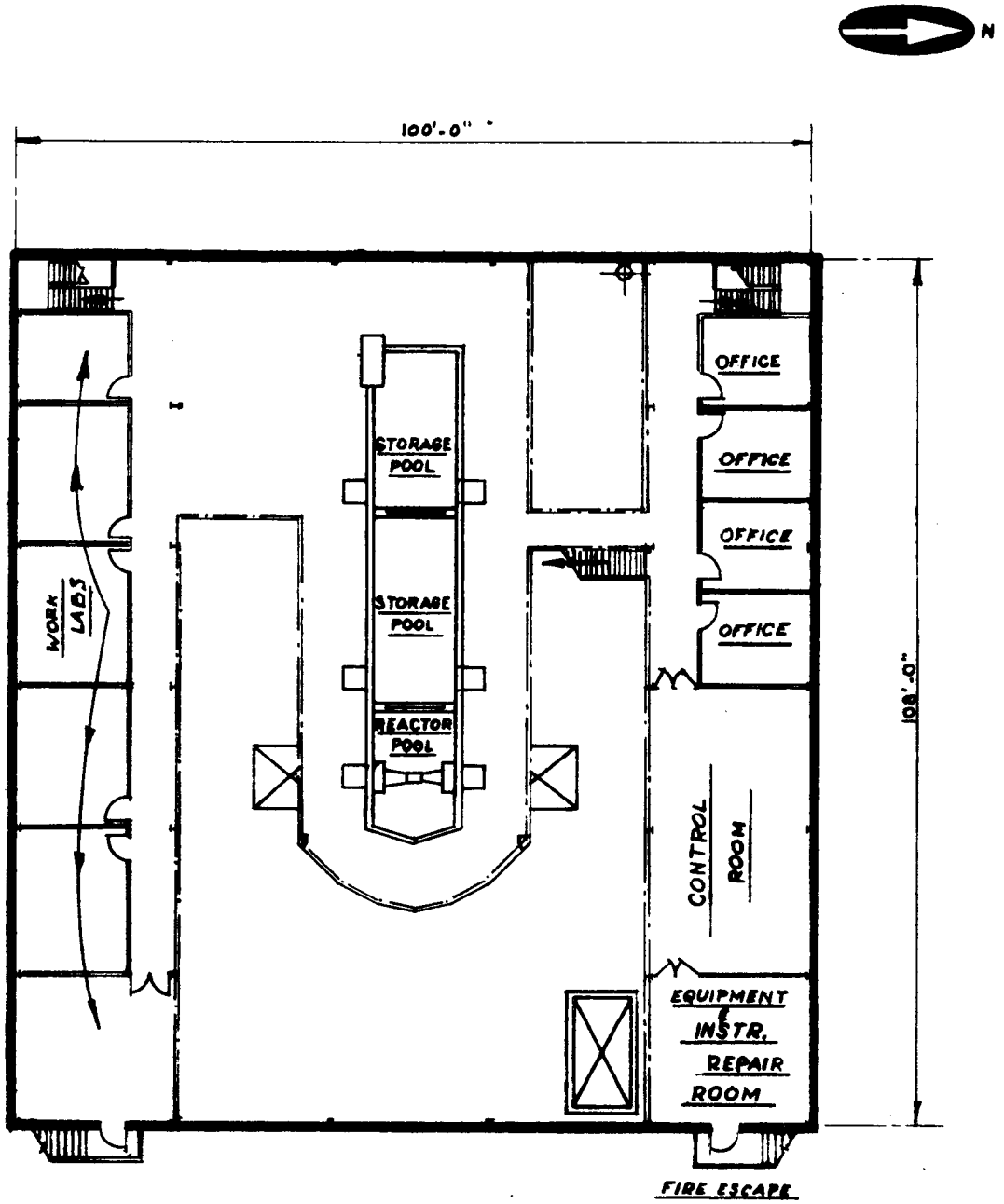

SCALE IN FEET

THIRD FLOOR PLAN

O.R.R. PROJECT

FIGURE 18

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The total floor area is 30,700 square feet and the gross building volume is 831,000 cubic feet.

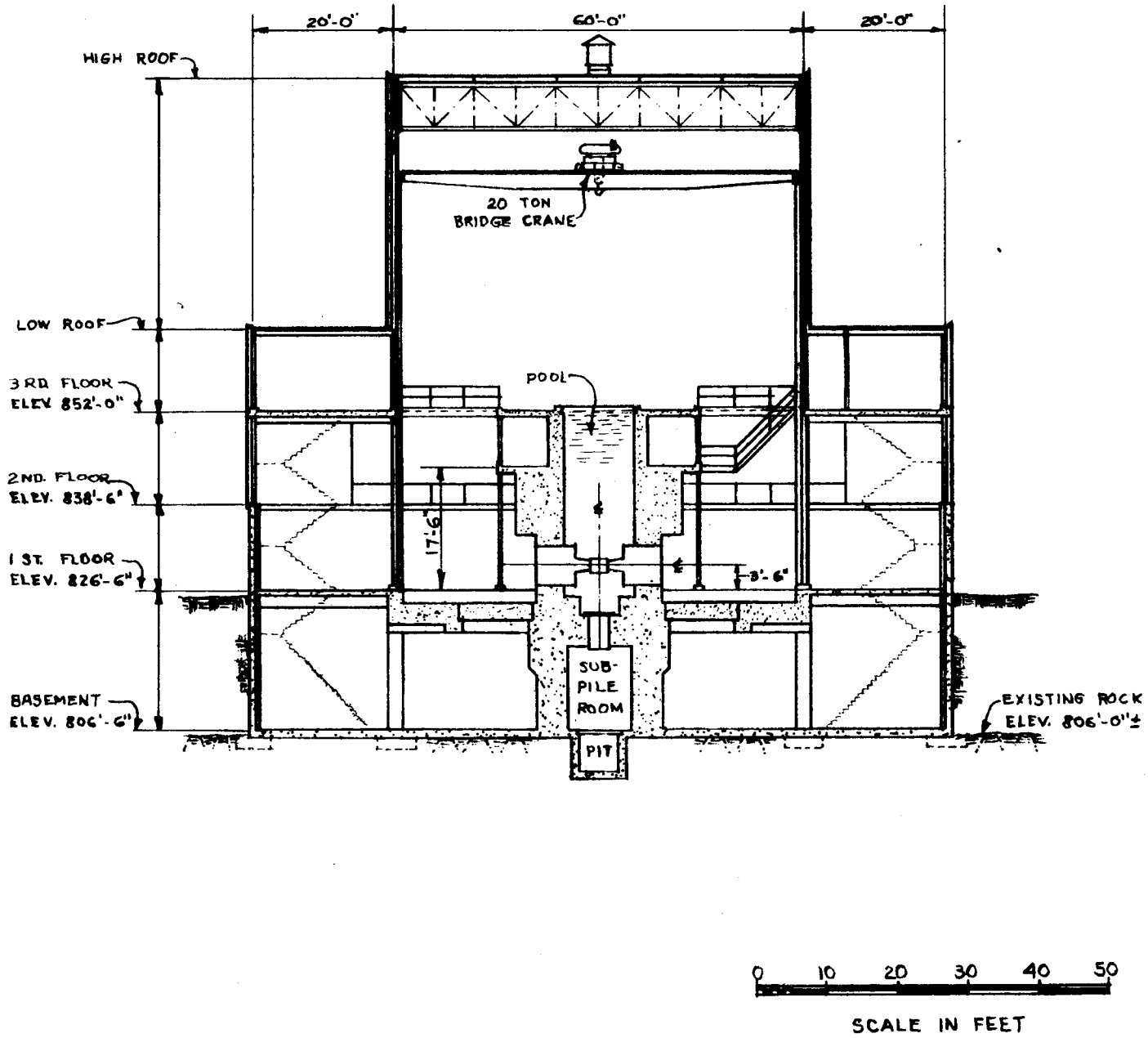
A twenty-ton bridge crane with auxiliary one-ton hook will be provided as illustrated in Fig. 19.

Heating will be provided by steam unit heaters and by fin type radiators in isolated areas. Ventilation will be provided by forced draft from the fan and filter room, located in the basement. As a fire prevention measure, the basement is sealed off from the floors above by hatch covers and fire walls around the stair wells. Air in the basement will be exhausted through the pump room and thence out of the building, through a vertical tunnel. Air conditioning will be limited to the third floor control room, offices, and equipment and instrument repair rooms.

Services provided to the reactor include plant process water, demineralized water, compressed air, "hot" off gas exhaust which connects with the existing "3039" 250' stack,[11] "hot" drains which connect with the existing ORNL waste disposal plant,[12] and process drains. These services are provided to convenient locations on the third floor walkway around the pool, on the second floor at the back of the pool convenient to the "rabbit" stations, and to all the experimental stations around the face of the reactor structure at the first floor level. Supply and return lines from the main reactor cooling circuit are provided to the first floor experimental stations for cooling purposes. These lines are imbedded in the pool wall in order to provide adequate shielding from N^{16} activity. Both 440 volt and 120/208 volt convenience outlets are liberally provided around the reactor structure. In addition, ten 50 amp. distribution boxes on the 150/230 volt special

Power exhausted on the roof.

* See errata



SECTION AT ℄
ORR PROJECT

FIG. 19

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service line are provided to the experimental areas for instrument plug-in. All services in the experimental areas are arranged to provide a high degree of flexibility. Numerous sleeves, pipe chases, and empty conduits are provided.

Initially, services to the six third floor laboratories will include plant water, compressed air, special and normal electrical services and process drains.

4. OPERATING PROCEDURE

4.a NORMAL OPERATION OF THE REACTOR

It is anticipated that the ORR will be operated on a basic 24 hour day seven days a week, however it will be necessary to modify this schedule in accordance with the requirements of the experimental program.

Operation of the reactor will be the responsibility of the Reactor Operations Department of the Oak Ridge National Laboratory Operations Division. This department is currently responsible for operation of both the LITR and the X-10 Graphite Reactor. Operating personnel will be chosen from this organization which has gained several years experience in operating a reactor of similar design, the LITR, and will report through their supervisor to the department superintendent.

At least two persons will be present at all times during operation; an operator to perform the required manual tasks, including manipulation of the control mechanisms, observation of instruments, recording of operational data, etc.; an experienced reactor foreman will be available to oversee the reactor operations and give instruction in the event of some atypical behavior of the machine. Since the Reactor Operations Department maintains a crew of four or five on all shifts, assistance is available for the performance of operations requiring additional personnel.

Initial operation of the ORR will be accomplished by the Reactor Operations Department with the assistance of experienced members of the Laboratory staff. Major changes in operating procedure, modifications of fuel loading, etc., will be reviewed by the Reactor Operations Review Committee which consists of five senior members of the Laboratory staff.

4.b OPERATION OF THE WATER SYSTEM

THE REACTOR COOLING LOOP

Satisfactory operation of the reactor cooling loop will be a prerequisite to reactor operation at any substantial power level.

Both the pool and reactor cooling loop systems will be filled initially with demineralized water from the plant supply. The reactor loop will be vented at the pumps, the strainers, and the air coolers. Venting from the reactor tank is automatic.

The valves on the system will be arranged in their normal operating positions and one of the main circulating pumps will be isolated from the system by closing the appropriate valves. The outlet valve on the other pump is then closed and the pump started. The discharge valve of this pump is then opened, permitting circulation through the loop as indicated by a flow indicator located in the pump gallery and a flow recorder in the control room. Ammeters in both control room and pump gallery indicate satisfactory operation of the pumps themselves.

It is estimated that any flow in excess of 2500 g.p.m. is permissible for five megawatts. For ten megawatt operation, use of the second pump is required.

Temperature elements will indicate inlet and outlet temperatures, and the temperature rise across the core on recorders located in the control room. Interlocks are provided to insure operation within normal limits. Control of fan speed and the position of the louvers on the air cooled heat exchangers will be accomplished from the control room. This control is required in order to provide heat removal at a rate consistent with the operating power

level and the prevailing ambient temperature.

Loss of coolant flow during operation at power will result in shutdown of the reactor through the interlock system; the heat released after shutdown will be transferred to the water in the reactor pool through the aluminum reactor tank. The large heat capacity of the pool will absorb this after-heat with a negligible temperature rise even if the pool coolant system is not in operation.

POOL COOLING SYSTEM

Concurrent with the startup of the reactor cooling loop, the pool cooling loop is put into operation. This requires that the turbine pump on the spray tower and the pool circulating pump be put into operation. Some degree of control of the cooling rate will be afforded by two-speed control of the cooling tower fan, however the response of the loop will be extremely sluggish due to the extremely large heat capacity of the pool.

In the event that the pool cooling system fails the rate of temperature rise in the reactor section of the pool at five megawatt operation will be of the order of 5° F per hour until an equilibrium temperature of 135° F is reached. At ten megawatts the rate of rise will be less than 7° F per hour and the equilibrium temperature is only slightly higher than that for five megawatts.

4.c FUEL ELEMENT HANDLING

Assuming uniform burnout of 15%, the life of a fuel loading, based on continuous operation at five megawatts, would be approximately 115 days. Because of the possibility of extending the allowed burnout and, to a lesser degree, because of scheduled shutdowns for maintenance and modification of

experiments it appears that the life of a loading would be more of the order of six to eight months.

It is not contemplated to change an entire loading at any one time. A tentative fueling plan, similar to that used in the LITR, calls for the shifting of the elements, at intervals of about two weeks, in such a way as to achieve uniform burnout. At the time this shifting is done those elements having a calculated burnout such that the minimum excess reactivity required will be unavailable before the next scheduled shutdown will be replaced.

Elements having a burnout equal to the maximum determined to be permissible will be hung in racks in one of the storage pools for a time sufficient to permit cooling to a level which will allow their safe transfer in shielded shipping containers to the processing plant at the National Reactor Testing Station at Arco, Idaho.

5. EXPERIMENTAL PROGRAM

5.a TYPES OF EXPERIMENTS TO BE PERFORMED

The ORR is to be constructed to serve as a permanent general purpose irradiation facility for use in the experimental and production programs at the Oak Ridge National Laboratory. The experiments anticipated, therefore, may be expected to be quite varied, including the conventional "in-pile" irradiation of small samples as well as complicated circulating fuel loops which release large amounts of energy during irradiation.

The following types of experiments are expected to form a large part of the operational load of the ORR:

(1) Dynamic loop tests to determine the effect on fluid fuels, and on the engineering equipment associated with them.

(2) The irradiation of fluid fuels in quantities large enough to permit chemical studies of continuous fuel processing systems.

(3) "Conventional" neutron experiments in which the reactor is utilized as a source of neutrons.

(4) Radio-isotope production

(5) Solid-State physics and metallurgy investigations of the effects of radiation on engineering materials and basic studies of the properties of metals, alloys and ceramics.

5.b LIMITATIONS TO BE PLACED ON EXPERIMENTS

Generalization of the limitations to be placed on experiments would serve little purpose here. It is obvious, however, that in addition to the specific problems directly concerned with the safety of the experimental apparatus, any effect that the experiments may have on the reactor must be governed by the same basic safety considerations as govern operation of the

reactor itself. The reactor control and safety system will be provided with means whereby information concerning the experiments can be introduced. Some measure of control will be exerted over the experiments in order to avoid violation of the basic safety criteria. The experiments must be treated individually and collectively with respect to their effect on the overall operation of the facility.

In order to insure the safety of the experimental program, all proposed experiments will be reviewed by the Oak Ridge National Laboratory Experiment Review Committee, a permanent committee which consists of five senior members of the Laboratory Staff.

6. DESCRIPTION OF THE SAFETY MECHANISMS

6.a NUCLEAR SAFETY

The primary nuclear safety device is the Level Safety System[6]. Three ionization chambers supply current to amplifiers whose outputs form an auctioneer circuit: the largest signal governs the response of the system. When the neutron flux at any chamber exceeds a preset level (usually 1.5 times full power; much lower during critical experiments), the Sigma Bus voltage (output of the auctioneer) will vary so as to cause a decrease in the current in the rod release solenoids which in turn will release the rods.

Following Newson[13], one calculates the shortest period occuring during a startup accident wherein the rods are simultaneously withdrawn at their maximum rate of 0.1% $\delta k/k$ per second. The period is found to be approximately 60 milliseconds. The rod release solenoids and mechanisms introduce a delay of 21 milliseconds[14], between the time the level reaches 1.5 times full power, and the time the rods actually begin to move. If one assumes a downward rod acceleration of 1 g, the power excursion to be expected from this situation will have a maximum value of 3.5 times full power.

The start-up accident described above uses a built-in potentially dangerous mechanism --shim rod withdrawal-- and the installed safety system prevents ill effects. Other nuclear accidents may be postulated which add various amounts of reactivity at various rates, due to, for example, sudden changes in installed experiments. The insertion of experiments or proposed changes in the reactor must be controlled by administrative action consisting in review of all proposed changes in operating procedure by the Operations Review Committee, and review of all proposed experiments by the Experiment

Review Committee. The possible interaction between experiments and the reactor is investigated by these committees, and particular attention is given to making sure that no conceivable accident can introduce reactivity in amounts or at rates exceeding those which can be adequately handled by the safety system.

During startups, or at low power, a period shorter than one second will result in a scram. This is done so that malfunction of equipment or maloperation at low levels need not raise the power above rated full power before a scram results.

Less drastic means are provided for correcting a potentially dangerous situation before it requires a scram. In order of decreasing severity they are:

(1) Reverse- Motor-driven insertion of all rods.

(2) Setback- Automatic decrease of requested flux level on servo system.

(3) Alarm- This requests the operator to initiate manual corrective action.

These corrections may be applied in echelon, such that an increasingly dangerous condition calls for an increasingly drastic cure, culminating in a scram if necessary. The level safeties are independent of other corrective actions.

6.b NON-NUCLEAR SAFETY CONSIDERATIONS

The reactor and the people associated with its operation must be protected against harm which could result from malfunction or misuse of non-nuclear equipment.

Health physics instrumentation is provided to warn personnel of con-

tamination or high radiation. These will include the following:

(1) One hand-and-foot counter,-personal contamination

(2) Two constant-air-monitors,-particulate air borne contamination (Five may be installed eventually)

(3) Six monitrons,-neutron and gamma ray intensity. (Thirteen may be installed eventually)

(4) Beta-gamma survey instruments of various types,-for operation and maintenance work in potentially dangerous areas.

Many experiments interact with the reactor in such a way that difficulty with the experiment could ruin the experimental device, cause an increase in reactivity, or even endanger the reactor structure. Detection of such conditions must be made to call for appropriate remedial action by the experimental device and by the reactor control system. Provision is made for interconnecting the experiment instrumentation with the reactor controls, so that more or less drastic corrective action may be taken as required. Depending on the nature of the experiment this may consist of simply an alarm to the reactor operator, or circumstances may require a full array of scrams, setbacks and alarms.

The integrity of the reactor may be jeopardized by the failure of the cooling system. It is therefore provided with appropriate safety devices, as follows:

(1) If the level of the water in the reactor pool or in either of the storage pools is lowered by an unsafe amount, an alarm will sound to warn personnel of the condition.

(2) If for some reason all water is lost from the reactor cooling loop and also from the pool, plant process water may be sprayed onto the lattice in order to dissipate afterheat. Valves for this purpose will be provided on the first floor and also outside the building.

(3) The activity of the reactor cooling water and of the reactor pool water will be monitered continuously; excessive activity will sound an alarm in the control room.

(4) The cooling system instrumentation will be interconnected with the reactor controls in such a way that, at power levels where cooling is needed, cooling failure will initiate a reactor shutdown. In particular, low flow or high reactor outlet temperature will initiate a scram.

Other interlocks will supervise the operation of the cooling system to minimize the effects of its malfunction or misoperation on reactor operation.

Chattanooga (1902), Sweetwater (1914), and Lenoir City (1918). The comparative freedom from seismic shocks of the Valley of East Tennessee, in which Oak Ridge is located, is due to the general stability of the earth's crust in this area.[15]. In the unlikely event that an earthquake severe enough to rupture the pool walls occured, it is not believed that other than local damage would be encountered.

7.b HAZARDS DURING NORMAL OPERATION

The potential hazards which exist during the normal operation of the reactor, but against which no automatic devices other than alarms are provided for the protection of the reactor and its operating personnel are:

1. Cracking of the shield due to thermal stress.
2. High radiation intensities due to inadequate shielding.
3. High radiation intensities due to contamination of the pool water.
4. High radiation intensities due to contamination of the reactor cooling loop.
5. Leakage from one or both water systems.

The steps taken to minimize these hazards are as follows:

(1) A radial temperature gradient will be created in the shield, particularly at a plane passing horizontally through the core center, due to the deposition of energy by the attenuation of gamma rays. It has been estimated [16] that the maximum temperature reached in the main shield will, at five megawatts operation, be insufficient to cause excessive stresses in the concrete. Thus, with the exception of that portion of the shield immediately adjacent to the large experimental facilities, no thermal shielding is contemplated initially. A two-inch lead thermal shield will be installed in the vicinity of the large facilities. Since the temperature in the interior of

the shield is a very sensitive function both of the conductivity of the shield and of the heat transfer coefficient at the inner wall, thermocouples will be installed in the shield in order to determine the actual temperature distribution and to insure against excessive heating. Should measurements indicate that excessive temperatures will develop, additional thermal shielding will be installed.

(2) Estimates, based on data given in the appendix, indicate that with the exception of the pump cubicles and the area immediately adjacent to the reactor loop heat exchangers, no regions of high radiation intensity are available to personnel. In both of these cases the high intensity is the result of induced activity in the water. Access to the heat exchangers and to the pump cubicles will be prevented by means of locked doors during operation. In neither of these locations will it be necessary to approach the equipment during operation. Remote control valve handles penetrate the shielding around the pumps, and the heat exchangers are located a sufficient distance from other laboratory facilities so that fencing will provide adequate protection.

Radiation intensity above the pool may be caused by the upward convection of N^{16} produced near the surfaces of the reactor core tank. Conservative estimates indicate that this intensity will exceed tolerance levels by a factor of about three for five-megawatt operation. Should undesirable intensities actually occur, provision has been made to install baffling in order to prevent the direct upward flow of the N^{16} bearing water from the vicinity of the core and thus eliminate the hazard.

(3) Contamination in the pool water is expected to be largely due to

its use as a storage facility for contaminated or activated material. The amount of induced activity will be small since the thermal flux available outside the core tanks will be quite low. The control of activity in the pool is therefore largely an operational problem. It is anticipated that the system will be operated in such a manner as to restrict the maximum radiation intensity at the edge of the pool to less than the tolerance level. Because of the short half life of N^{16} it is believed that none of this material will leave the reactor pool.

In addition to these sources of radiation there is a minor interchange of water between the cooling system of the reactor and the pool system. This, too, is expected to be negligible compared to contamination from stored materials.

Adequate instrumentation will be provided to give warning of increases in activity of the pool water.

(4) Normally the water in the reactor cooling loop will contain small quantities of activity due to the activation of impurities and to reactions such as $Al^{27}(n,\alpha)Na^{24}$. An examination of the effluent from the LITR, operating at three megawatts yields the following estimates for the activity in the cooling water leaving the bottom of the fuel elements in the ORR.

NUCLIDE	HALF LIFE	ACTIVITY (mc./ft^3)
N^{16}	7.3 sec.	1.63×10^3
Na^{24}	14.9 hr.	0.36/megawatt
Mg^{27}	9.6 min.	0.50/megawatt
Al^{28}	2.3 min.	1.63/megawatt

Thus, in normal operation, shielding of the hard gamma ray from N^{16} is the

controlling factor for the first few seconds after the cooling water has left the core. The important consideration here is the flow rate since the concentration of N^{16} is essentially independent of the power level. Shielding of the pipe lines is adequate for flows several times greater than 2500 g.p.m., however, flow rates greater than 5000 g.p.m. would require the installation of additional shielding around the pump cubicles.

The addition of an appreciable amount of fission products to the cooling system would result in higher than normal intensities in the pump cubicles and at the heat exchangers. Since these locations are not normally entered during operation, little danger to personnel would result. The cooling loop will be provided with adequate instrumentation to detect abnormal rises in activity. In addition, such increases would be quickly detected by an increase in activity on the ion exchange beds of the demineralizer.

7.c COOLING SYSTEM FAILURE

It has been shown by E.T. Jurney[17] that a sudden removal of coolant concurrent with shut down of a reactor of the ORR type operating at five megawatts would not result in melting of the fuel elements in less than 13 minutes. Extrapolation of this calculation to ten megawatts indicates that about six or seven minutes would be required to supply sufficient afterheat to melt the core even if no heat transfer from the core were permitted.

In order to protect against a possible loss of cooling water, seal legs are provided on the exit water lines and provision is made to allow water from the pool to circulate in a free convection loop through the reactor core tanks. In addition to this, water may be added directly to the

core from the plant process system.

In view of these precautions it would appear that only a major catastrophe resulting not only in loss of water to the core cooling loop, but also rupturing the pool itself and at the same time disrupting the process water supply would be likely to result in melting of the core.

5.d LOSS OF SHIELDING DUE TO LOSS OF POOL WATER

Since the top shielding and to a lesser extent the side shielding of the ORR depend upon the fact that the core tanks are immersed in the pool, it is clear that any situation in which the level of the water in the pool is not maintained is dangerous.

In order to prevent accidental draining of the reactor section of the pool no provision has been made to drain this section below the top of the reactor tank. All connections, both inlet and outlet, are above this level. Temporary piping would have to be installed in the event this section is to be drained.

The remaining two pool sections have inlet and outlet lines located in the floor of the pool, however the gates between these pools and the reactor pool will normally be in place during operation. The inlet lines to the storage pools are 4" diameter and are protected from inadvertent draining by check valves immediately below the pool floor. The outlet lines are 2" diameter and approximately 3½ hours is required to drain the storage pools. This should be sufficiently long to detect and correct an operating error. An alarm system will be provided to warn when the level of any of the pool sections drops below the safe minimum.

7.e ACCUMULATION OF LARGE VOLUMES OF CONTAMINATED WATER

While the pool and cooling loop demineralizers are considered adequate to handle contamination in excess of the levels expected, there does exist the possibility that, due to malfunction of the system, it may become necessary to dispose of large quantities of contaminated water.

The two water systems contain a total of approximately 130,000 gallons. There are available at the Oak Ridge National Laboratory several methods of handling this water. The water could be sent directly to one or both of two 20,000 gallon retention ponds and then bled slowly into the Clinch River by way of White Oak Creek as is done with other "warm waste". In case additional capacity were needed temporary lines could be installed to either or both of two other retention ponds which have a total capacity of 400,000 gallons. The water could be impounded here for a time sufficient to allow for decay of the shorter lived components and then handled through the regular "warm waste" system.

Should the activity of the water be too great to handle by either of the above methods, temporary piping from the reactor building could be installed conveying the water to the ORNL tank farm (a distance of approximately 400') whence it could be led through existing piping to either of two 500,000 gallon lagoons where its volume would be dissipated by solar evaporation and the concentrate impounded indefinitely. This procedure is in current use for the disposal of large volumes of radioactive chemical waste from other processes at the Laboratory.

7.f SABOTAGE

Sabotage of the ORR would most readily be effected by use of an explosive or thermite bomb to release the accumulated fission products after the reactor had been operated at high power levels. It is not impossible that a clever saboteur could circumvent the security measures in force at ORNL, although such a possibility is quite remote.

Sabotage of the reactor safety system could conceivably be effected only by a person thoroughly familiar with the detailed layout and function of the system who also had the opportunity to make considerable alterations or additions to the physical system. In view of the reasonably large number of experienced personnel who will be in the area at all times, it seems unlikely that such an attempt would go unnoticed.

The site is classed as an exclusion area and is protected according to General Manager's Bulletin No. 153, Physical Security Standards for AEC Facilities. Normally a full complement of personnel will be on hand at the reactor twenty-four hours a day, seven days a week. Should the occasion arise when the reactor is not operated for one or more shifts, a standby crew will have charge of the building. In addition to this, regular hourly checks by the Oak Ridge National Laboratory Security Department would be maintained.

8. ULTIMATE CATASTROPHE

8.a THE CASES CONSIDERED

Ultimate catastrophe is considered to occur if a situation should arise in which a large fraction of the activity of the reactor core is released in the form of a cloud of gas and particulate matter, or is dissolved in water and released.

The arrangement and nuclear characteristics of the fuel region are such that a nuclear explosion would probably not follow a reactor runaway. In a large stepwise increase in reactivity, the periods expected are of the order of milliseconds[18], a time large compared to the propagation of sound in the core materials.

There exists, however, the possibility of a steam explosion in the core tanks due to the sudden conversion into high pressure steam of an appreciable fraction of the water contained therein. The energy for this transition could be supplied by a stepwise increase in power resulting in melting of the fuel elements and possibly followed by the initiation of the highly exothermic Al-H_2O reaction.

While very little specific data is available concerning the Al-H_2O reaction it is clear that such a reaction is possible, particularly if the fuel elements have been melted. Whether the reaction will take place between solid fuel elements and water has not yet been determined, but seems unlikely. The total energy release from the Al-H_2O reaction is approximately 2×10^5 calories per mole of aluminum. This heat is more than enough to vaporize a mole of aluminum, thus the reaction would be expected to be autocatalytic and could result in a severe accident.

If we make the assumption that the active portion of the lattice contain approximately 105 Kg of aluminum and Al-U alloy, approximately 2.5×10^7 calories are required to melt the fuel assembly. This corresponds to a pulse of about 105 megawatt seconds in a time short enough to preclude both heat transfer to the coolant and operation of the control mechanisms.

In the Borax experiment[19] where 4% excess reactivity was added to a subcritical MTR type lattice in about 0.2 seconds, a pulse of 130-180 megawatt-seconds was experienced. This resulted in an explosion, but since it is virtually impossible to add reactivity to the ORR by any operational procedure at a rate exceeding 0.1% per second, the possibility of an incident such as the Borax explosion resulting from misoperation is effectively excluded.

The possibility does exist that a sudden increase in reactivity might be caused by the sudden flooding of one or both of the large Engineering Test Facilities, had they previously been operated as voids. A maximum of about 3% $\delta k/k$ could be added in this fashion. For this reason, extreme care must be taken in the use of these facilities. Normally, if they are not occupied by experiments, the dummy plugs will be in place, thus preventing the possibility of flooding. Experiments for use in these facilities will be designed in such a manner as to prevent this type of occurrence.

For the purpose of estimating the maximum danger to be expected to the surrounding area in the unlikely event that ultimate catastrophe should occur, the following cases are considered: (1) the entire contents of the core is put in soluble form and conveyed to the watershed, (2) all or some of the core is entrained in a cloud of steam released to the atmosphere.

8.b SURFACE WATER CONTAMINATION HAZARDS OF ORR*

SUMMARY

Emergency values for the use of drinking water for periods of 1 day or of 30 days have been used to evaluate the hazards of contamination of the public water downstream from a major reactor accident to the ORR or HRT.

Such limited use of the water (1 da or 30 day) at the appropriate emergency MPC value would lead to a total internal dose of 25 rep.

The downstream water concentrations exceed these emergency values at the four points considered by the following factors:

	Factor of Emergency MPC	
	1 da	30 da
K-25	42.0	586.5
Kingston	10.5	169.3
Watts Bar	0.044	0.79
Chattanooga	0.008	0.17

These results indicate the necessity for containment of the major fraction of released active contaminants which may become waterborne.

The severity of a rain-out problem may be inferred and the economic factors involved have not been appraised.

INTRODUCTION

The following discussion is grouped under these headings:

1. Assumptions Regarding the Release of Contaminants into Clinch River.
2. Potential Hazards from Water Contamination.
3. Estimates of Peak Concentrations at Selected Locations.
4. Maximum Permissible Concentrations of Radioactivity in Drinking Water.
5. Other Factors Affecting Peak Concentrations and MPC Values.
6. Comments Regarding Potential Hazards from Surface Water Contamination.
7. Conclusions
8. Acknowledgements

* Prepared by THJ Burnett ORNL Health Physics Division

The most serious conditions of surface water contamination resulting from a major reactor accident involving the ORR or the HRT would be encountered if all or a major portion of the radioactive materials in the reactor were to be released and enter the waters of the Clinch River during a short period of time. As will be disucssed later, this is an extremely remote possibility and one which should be prevented by appropriate measures such as retention pits, etc., but it is examined here to provide a working basis for hazard estimates. In this discussion it is assumed that the waters of Clinch River may be heavily contaminated from the ORR or HRT and that possible downstream effects, particularly potential hazards through water supply systems, must be considered.

ASSUMPTIONS REGARDING THE RELEASE OF CONTAMINANTS INTO CLINCH RIVER

With respect to the initial contamination of Clinch River it is assumed for these preliminary calculations that:

(1) The reactor has been operating continuously for a period of 82 days at a power level of 10 megawatts and without replacement of fuel. Some shift in the relative amounts of different isotopes would occur for other operating periods and it is planned to recalculate the amounts of those isotopes of major concern and evaluate.

(2) By some coincidence of events all of the radioactive materials in the reactor and its container are released and dispersed (in solution or in fine suspended matter) into the shielding water of the ORR concrete pool.

(3) The water in the pool containing all the activity is immediately discharged (such as by rupture of one or both of the large experimental beam holes) escapes the building, and pours out onto the ground surface.

(4) By heavy rainfall and surface run-off all of the waterborne radioactive materials are flushed down the valley and into the Clinch River without significant loss or holdup.

(5) From the time of the initial release of activity from the reactor, the radioactive contaminants flow into Clinch River at the mouth of White Oak Creek - mile Cl 20.8. - are uniformly mixed with the concurrent river flow past that point during a period of 6 hrs.

(6) The dispersion of contaminants described above occurs at a time of "worst conditions" of water flow in the Clinch and Tennessee Rivers from the standpoint of estimated peak concentrations of radioactivity in the water at the intakes of downstream water supplies or of steam plants that might be seriously affected.

POTENTIAL HAZARDS FROM WATER CONTAMINATION

The significance of potential hazards from radioactive contamination of drinking water supplies depends upon a number of factors, including: (1) the number of people who might be exposed by use of the contaminated drinking water; (2) the kind and concentration of the various contaminants in the water used for drinking; (3) the period of exposure through continued use of the water; and (4) the counter-measures that could be effected if necessary. A potential hazard at steam plants taking boiler water from the river would be the exposure of personnel to external radiation from radioactive materials concentrated in water softening units or in boilers. Local concentrations of activity from the discharge of water softener regenerating solutions might also have to be monitored and controlled.

Four downstream locations have been selected for critical analysis of the conditions of water contamination that may be foreseen from the assumptions outlined above. These locations represent the river water normally used by three steam plants above the mouth of Clinch River and by four

domestic water treatment plants as listed below. In terms of river miles above the mouth of each river (Cl = Clinch River and Te = Tennessee River) these points are: mi Cl 13.2; mi Cl 00 = Te 567.7; mi Te 529.9; and mi Te 465.3.

1. K-25 - mi Cl 13-2

There is a water treatment plant at K-25 (intake approximately mi Cl 14.4) to provide drinking water for several thousand people during working hours and process water for the plant. There is also a steam plant and a power generating plant (intake at mi Cl 13.2) for which the water is taken from the Clinch River. A representative intake of these three plants is taken as mi Cl 13.2.

2. Kingston Steam Plant (TVA)

The mouth of Clinch River (mi Cl 00) is taken as representative of the Kingston steam plant and its water treatment facilities; actually the water intake for boiler water and for drinking water is located a short distance upstream on an embayment of Emory River. If necessary drinking water for the small operating force could be obtained from distant source.

3. Watts Bar Village

The intake of the water treatment plant for Watts Bar Village is located above Watts Bar Dam (river mi Te 529.9). The village is a resort community with perhaps 1,000 maximum population.

4. Chattanooga

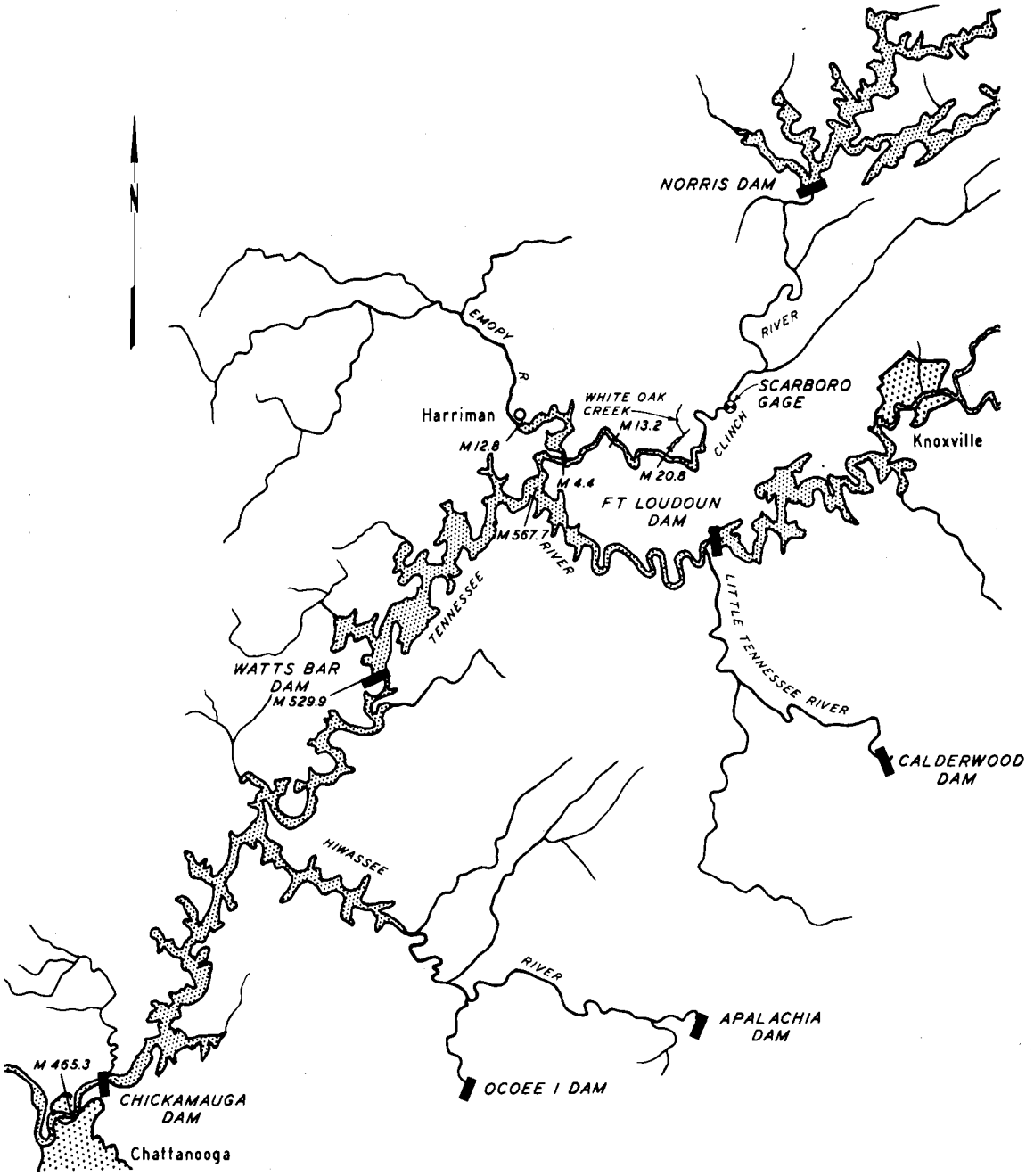
In the city of Chattanooga, approximately 131,000 population, a water treatment plant is located about 5 mi downstream from Chickamauga Dam (intake at mi Te 465.3). This plant provides the domestic and industrial water supply for the city of Chattanooga and several surrounding communities.

The town of Harriman, approximately 6,500 population, is located about 12 miles upstream on the Emory River, but contamination at the Harriman water plant is problematical since upstream flow from the Clinch River to Harriman occurs only occasionally under special conditions of stratification. The filtration plant intake is near the upstream limit in Emory River of backwater from Watts Bar Dam. If contamination should occur, it might be expected that the concentration of activity at the Harriman water plant would be considerably less than at the mouth of Clinch River.

The relative location of the several points in the Clinch and Tennessee River systems mentioned above are shown on the accompanying map. (Fig. 20)

ESTIMATES OF PEAK CONCENTRATIONS AT SELECTED LOCATIONS

The four locations in the Clinch and Tennessee Rivers which are outlined above were selected as the points of principal interest because of the usage of river water at these locations. The peak concentrations of radioactive materials in the water that might be expected near the point of entry into the Clinch River and at these downstream points will differ widely, depending upon the complex inter-related factors of stream flow and water elevations in the TVA reservoirs. Exact data relative to the combined effects of these diverse factors under various conditions are not available. However, through cooperation of the Division of Health and Safety and the Hydraulic Data Branch of TVA, very extensive tabulations of data with charts and estimates based upon calculations have been prepared during the past several months and have been made available to ORNL for use in the preparation of this report.[20] These estimates are considered by the TVA hydraulic engineers to be reasonably accurate. The times of water travel, given to the nearest 1/2 day, may be 10 to 20 percent <u>below</u> what will actually be observed in the field and values for dilution and stream flow are stated as ± 10 percent.



TENNESSEE VALLEY AUTHORITY
DIVISION OF WATER CONTROL PLANNING
HYDRAULIC DATA BRANCH

Fig. 20 AEC SPILLAGE STUDY

LOCATION MAP

Scale 0 10 20 30 Miles

OCTOBER 6, 1952

The essential values estimated and supplied by TVA include; time of water travel from the point of entry to the four selected locations; volumes of stream flow in CFS at the selected points; the percentage of reduction in the initial peak concentration of contaminants at the point of entry into Clinch River owing to dispersion into the stored water of the reservoirs downstream; and the volumes of dilution water that would further reduce the initial peak concentration because of increases of stream flow below the point of entry of the contaminants. Information was also obtained concerning the seasons of the year and the stream flows that would induce stratification in the Clinch River embayment and the Watts Bar reservoir and the effects of stratification upon the expected reduction in the peak concentrations by dispersion and dilution.

The analysis has been made by selection of appropriate available data and by calculations in line with the assumptions stated previously in order to determine the following:

(1) The initial peak concentation in Clinch River at the mouth of White Oak Creek (mi Cl 20.8) during a 6 hour period following the initial release of activity.

(2) The elapsed time until the peak of concentration would reach each of the selected locations downstream.

(3) The percentage reduction by dispersion of the initial peak concentration when the contamination reaches each of the four selected locations..

(4) The dilution factor effective in further reducing the peak concentration at the selected locations.

(5) The radioactivity from individual isotopes and from total contaminants in the water after radioactive decay during the elapsed time required to reach each of the selected locations.

(6) The estimated peak concentrations (by calculations based on the above factors) of individual isotopes and of total contaminants in μc/ml at the four selected locations.

The combination of "worst conditions" (conditions that would combine to give the maximum peak concentration at each of the selected locations) considered here are those river and lake conditions determined[21] by several calculations assuming various conditions of water level and stream flow. These trial estimates determined that the highest peak concentrations would be expected under the following conditions:

(1) Stratified conditions in the Clinch River embayment and in Watts Bar Reservoir downstream to Watts Bar Dam.

(2) Stream flows: in Clinch River from mi Cl 20.8 to mi Cl 00, 1000 cfs; in Tennessee River at Watts Bar Dam (mi Te 529.9) 18,900 cfs; in Tennessee River at Chattanooga (mi Te 465.3) 25,500 cfs.

(3) Pool levels (water surface elevations): at Watts Bar Dam 740 ft; at Chickamauga Dam 682 ft.

A summary of the pertinent factors under the conditions outlined above is given in Table I.

The next step in the analysis was to estimate and to tabulate the amounts of radioactivity (in curies) from individual radioisotopes and from the total fission product mixture that would be present (after decay) in the Clinch River at the mouth of White Oak Creek and at the four locations of principal interest downstream. As shown in Table I the periods of total

elapsed time from the initial release of activity until the peak of contamination will arrive at each of these five locations are 0, 3.25, 9.50, 17.0, and 25.5 days, respectively.

The radioactivity after these five time intervals was calculated using the assumptions stated previously and standard buildup and decay formulas with thelatest available fission chain yields (22)(23)(24). In the tabulation of the results of these calculations, the noble gases, Xenon and Krypton, and any isotope that accounted for less than 0.1 percent of the total activity were omitted. The curies of total activity for the 30 odd isotopes of hazard significance after the several time intervals, estimated by this method, are approximately as follows: initially 7.73×10^6c; after 3.25 days 4.24×10^6c; after 9.5 days 2.94×10^6c; after 17 days 2.31×10^6c; and after 25.5 days 1.90×10^6c.

The estimated amounts of radioactivity as explained above, the assumed immediate dilution in Clinch River by a flow of 1,000 cfs for a period of six hours, and the factors of reduction in the initial peak concentration by dispersion and by dilution as given in Table I, were used as a basis for calculating the peak concentrations at the selected downstream location. The estimated initial peak concentration of total activity at the mouth of White Oak Creek is approximately 12.63 μc/ml. The estimated peak concentrations for the individual and total fission products at the four selected downstream locations are shown in Table II. The totals of the factors by which these isotopes in their relative concentrations exceed the emergency values (which would give the maximum allowable emergency internal exposure of 25 rep) are shown at the bottom of Table II. For example, the anticipated worst concentration at K-25 exceeds the 1-day emergency use by a factor of 42. This means that the consumption of 2.2 liters of this water

thus contaminated would lead to a serious internal dose of 1,050 rep total, all of which, however, would not be to the same body organs.

MAXIMUM PERMISSIBLE CONCENTRATIONS OF RADIOACTIVITY IN DRINKING WATER

Values of maximum permissible concentration (MPC) of radioisotopes in drinking water for long-term use have been suggested by the National Committee on Radiation Protection in National Bureau of Standards Handbook No. 52. MPC values for individual isotopes and for unknown mixed contaminants in drinking water for temporary use during brief periods and in emergencies can be much higher than the suggested values for long-term use. A method of estimating MPC values for drinking water contamination that are acceptable for emergency conditions has been developed and described by Morgan and Straub.[25] These values are currently being revised to include the factor N = 5 for certain bone-seekers and some will be reduced thereby. By use of this method, emergency MPC values for individual radioisotopes in drinking water have been determined for 1 and 30 days of use, respectively (Table II). The use of drinking water for the specified period at these designated concentrations will lead to a total internal dose of 25 rep during the time the isotope remains in the body. On the basis of the estimated proportion of the individual fission products at the selected points of principal interest in the rivers, the factors by which these specific isotopes exceed the emergency MPC values are tabulated in Table III and the total factor by which their combined respective concentrations exceeds the emergency use limits is given at the bottom of both Table II and Table III.

These emergency MPC values are included as a guide in judging the degree of hazard through drinking water that may be caused by the release of mixed fission products and the levels of contamination discussed above. The

basis and validity of these estimates are explained by Morgan and Straub.[25] As a basis for judgment they probably are the best possible estimates that can be obtained at this time.

Unfortunately, data are not available for estimating the duration of the peak contamination at each of the four selected locations or the period during which the expected levels of contamination in Clinch River would exceed a given MPC value. It is reasonable to expect that in Clinch River at K-25 the duration of peak contamination would be brief and that the wave of heavy contamination would pass fairly quickly. At the mouth of Clinch River, representing the Kingston steam plant, it would be expected that the duration of the peak would be longer and that heavy contamination would persist for a few days. Comparison of the values for Watts Bar Dam and Chattanooga shows that the expected peak of contamination is such as to be of probable concern and perhaps to require emergency measures to provide alternate drinking water supplies and/or special water treatment measures.

OTHER FACTORS AFFECTING PEAK CONCENTRATION AND MPC VALUES

The values given in Table II have been estimated to represent the "worst conditions" and have taken into account only those factors for which a basis for quantitative estimates is available. There are several indeterminate factors which would combine to reduce the peak concentrations that would be expected at the various points downstream and to increase the emergency MPC values that might be considered acceptable. Although not included in the above estimates, these factors will be mentioned briefly as outlined below.

(1) The assumption that 100 percent of the activity released at the ORR or the HRT site would enter Clinch River is pessimistic. A holdup of some part of the radioactive materials by sorption in the soil and organic matter during the flow over the surface from the site* to the river would be expected even in a heavy rain.

(2) After dispersion in the river it would be expected that a certain part of the radioactive contaminants would be removed by adsorption, coagulation, and settling, particularly in the lower portion of Clinch River and in Watts Bar and Chickamauga Reservoirs. This factor would be effective in the case of certain isotopes that are readily adsorbed from solutions by clays and organic suspended materials which would partially settle and carry contaminants to the bottom; any radioactive solid particles would be even more likely to be removed because of agglomeration or attachment to larger suspended particles.

(3) Water treatment processes in community water supplies and at steam plants provide an added safety factor. Each of the community water supplies listed above includes water treatment by coagulation and filtration, and the two major steam plants employ water softening or deionization treatment of the boiler water. Data on the removal of a number of radioisotopes by water treatment procedures may be found in the literature.[26] A conservative value that may be applied to the removal of mixed fission products by alum coagulation and filtration procedures is 70 percent. Water softening or deionization units would be expected to remove a high percentage (perhaps 99 percent) of the radioactive ions present.

(4) In this case, with a release of contamination during a short time, the peak concentration of radioactivity in the river at a particular

* See errata

point would not persist for long periods and the average level for the period of exposure through use of the water would be lower than the peak shown in Table II.

(5) The emergency MPC values shown in Table II presuppose a use of the water 100 percent of the time for the period of 1 day or 30 days as stated. For those individuals, such as workers at K-25, who obtain only a part of their drinking water requirements from the contaminated supply, the MPC vaue for the stated period would be higher.

COMMENTS REGARDING POTENTIAL HAZARDS FROM SURFACE WATER CONTAMINATION

An over-all view of the conditions of contamination and the hazards that might result in the Clinch and Tennessee Rivers is provided by the location map (Fig. 20) and the values given in Table II. As the wave of contamination moves downstream the peak concentration would be reduced considerably by decay and even more markedly by dispersion and dilution. At this stage of decay the fission product mixture would contain considerable proportions of the longer-lived radioisotopes (for example I^{131}, Sr^{89}, Zr^{95}, Ru^{106}, Cs^{137}, Sr^{90}, and Ce^{144}) some of which are the most important drinking water contaminants.

With regard to exposures through drinking water, the K-25 plant, being closest, would be subjected to a higher peak of contamination than would the plants farther downstream. The estimated peak of 4.3 μc/ml is approximately 42 times the 1-day emergency MPC value for the total mixture of contaminants at this point and is about 586 times greater than the 30-day MPC value. It appears that a release of activity under the adverse conditions assumed would certainly make it necessary to discontinue the use of Clinch River as a source of drinking water for K-25 employees until survey measures and analyses proved its safety. Normal storage of 2,500,000 gal of treated water will supply the drinking water system (exclusive of air conditioning) for 24 to

26 hrs. For longer periods it may be possible for the filtration plant to take raw water from Poplar Creek through the process water system, subject to survey verification.

At the mouth of Clinch River the peak concentrations of activity would exceed the emergency 1-day MPC value by a factor of about 10 and would be higher than the 30-day MPC value by a factor of about 169. This may not present a serious problem since no communities take drinking water from the Clinch River near this point; employees of the TVA steam plant could obtain drinking water eleswhere, and the water supply of Harriman, if affected at all, would almost certainly be subjected to a much lower peak concentration.

At the Tennessee River locations of Watts Bar Dam and Chattanooga the peaks of contamination would be appreciable but for a short period of use would be within the acceptable limits of MPC values. The duration of appreciable levels of radioactivity that would result from a single discharge of contamination at ORNL would not be expected to be longer than two to four weeks at these Tennessee River points and would require sampling and analysis to evaluate the hazards.

Possible hazards from exposures to external radiation would be less serious than drinking water exposures but would have to be considered with the peak levels of radioactivity shown in Table II. Calculations and experimental measurements have indicated that radiation levels would be 5 to 10 mr/hr just above the surface of large volumes of water containing 1×10^{-2} μc/ml of a typical mixture of fission products. Hence, in the river opposite White Oak Creek a reading of 6 to 12 r/hr would be expected at the time of initial release. Trained personnel and water survey instruments suitable for downstream monitoring are available at ORNL, and in the past the follow-up of

minor releases of contamination into Clinch River has been undertaken by the Health Physics Division. In case a major release should occur, as visualized in this report, field survey individuals or teams would function to observe the progress and levels of contamination in the river and to render advice, assistance or supervision as necessary to prevent undue radiation exposures.

A quite serious problem might be presented by external radiation in proximity to water softener units at steam plants. For example, if it is assumed that a water demineralizer at the Kingston steam plant treating 350,000 gal per day of river would remove 95 to 99 percent of the radioactivity and that the concentration in the water is 1.15 μc/ml, the accumulation in the resin bed would be approximately 1440 curies of activity in one day. Data from pilot plant water treatment experiments at ORNL indicate that in such a case the radiation level at the outer surface of the demineralizer tanks would be of the order of 58 to 300 r/day. Radiation levels in proximity to water pipes and regenerant solution discharge pipes would be much lower. There should be very little concentration of radioactivity in the boilers from the use of demineralized water with frequent or continuous blowdown. The situation at the K-25 steam plant and power plant would be similar and correspondingly higher. Under these conditions it may be doubtful if the plants could continue to operate and advice and assistance to the supervisors and operators would be certainly necessary. This should include survey and personnel monitoring services while the wave of contamination is passing and afterwards until radiation levels have lowered to acceptable values.

CONCLUSIONS

From the assumptions and analysis given and discussed above, it appears that a total release of activity from the ORR or the HRT under the

most adverse conditions would cause serious contamination in the Clinch River. Although there might be time to survey and appraise the situation and to arrange for any counter-measures that might be necessary, precautions must be taken to control such release to 1 or 2 percent of the activity present until meausres can be taken to confine it for adequated decay and treatment. It is indicated that in the Tennessee River the levels of peak contamination would be lower and that some hazards would result, although they may be of an acceptable severity.

In the event of a sizeable release of radioactive material in the Clinch River it would be essential that facilities for measurements of radioactivity and radiation should be available and that necessary information and assistance should be provided to downstream communities or installations on the Clinch River that might be affected. It would be desirable that measurements of peak concentrations and times of flow in both the Clinch and Tennessee Rivers should be made systematically to provide assurance that there was no resultant damage and to obtain information which would be useful in case subsequent accidental releases of radioactive contamination should occur.

No evaluations have been attempted of the economic liability incurred due to the contamination of watercraft and dock facilities. Likewise the hazards and economic factors arising from the use of river water by stock animals have not been estimated. These may be readily seen to be important, however.

It may well be justified to consider some plan for the TVA to release extra water from its dams to alleviate the emergency conditions in event of a major reactor disaster such as postulated

Table I

Hydraulic and Hydrologic Factors for Estimating Peak Concentrations of Contamination

Point in River System	Pool Level Elevation (ft)	Stream Flow (cfs)	Time of Water Travel from mi Cl 20.8 (days)	Reduction of Peak Concentration at mi Cl 20.8 (%)	
				By Dispersion	By Dilution
K-25, mi Cl 13.2	-	1,000	3.25	38	0
Mouth of Clinch River mi Cl 00 = mi Te 567.7	-	1,000	9.50	76	0
Watts Bar Dam, mi Te 529.9	740	18,900	17.00	97	95
Chattanooga, mi Te 465.3	682	25,500	25.50	99	96

Table II

Estimated Peak Concentration of Radioactivity and Emergency MPC[a] Values for Drinking Water at Selected Locations in the Clinch and Tennessee River System

Isotope	Peak Concentrations at Indicated Locations - μc/ml: K-25 Steam Plant mi Cl 13.2 (3.25 das)	Mouth of Clinch R. mi Cl 00 & Te 567.7 (9.5 das)	Watts Bar Dam mi Te 529.9 (17.0 das)	Chattanooga Water Plant Intake mi Te 465.3 (25.5 das)	Emergency MPC for Drinking Water μc/ml: For Use 1 day	For Use 30 days
	$\times 10^{-3}$	$\times 10^{-3}$	$\times 10^{-5}$	$\times 10^{-6}$		
Sr^{89}	298.92	106.61	60.44	144.08	1.4×10^{-1}	5.6×10^{-3}
Sr^{90}-Y^{90}	4.36	1.69	1.05	2.81	1.5×10^{-3}	5.0×10^{-5}
Sr^{91}	2.18	.00	.00	.00		
Y^{91}	335.51	120.98	69.38	167.96	5.2×10^{-1}	2.0×10^{-2}
Zr^{95}	350.91	127.08	73.33	178.59	2.7×10^{-1}	1.1×10^{-2}
Nb^{95m}	3.61	1.33	0.77	1.89		
Nb^{95}	210.39	87.27	57.55	158.65	5.8×10^{-1}	2.6×10^{-2}
Zr^{97}-Nb^{97}	25.36	0.02	.00	.00	2.9×10^{-1}	2.9×10^{-1}
Mo^{99}-Te^{99}	241.66	19.55	1.62	0.63	8.4×10^{-1}	1.8×10^{-1}
Ru^{103}-Rh^{103}	221.50	76.94	42.19	97.09	6.6×10^{-1}	2.7×10^{-2}
Rh^{105}	20.98	0.46	0.01	.00	1.3×10^{0}	4.6×10^{-1}
Sb^{127}	10.29	1.30	0.21	0.13		
Te^{131}	45.11	0.53	.00	.00	5.6×10^{-2}	2.4×10^{-2}
I^{131}	236.49	55.51	18.35	23.74	8.8×10^{-3}	7.9×10^{-4}
Te^{132}	203.97	20.61	2.58	1.19	3.1×10^{-1}	5.9×10^{-2}
I^{132}	210.49	21.26	2.68	1.22	1.5×10^{1}	1.5×10^{1}
I^{133}	43.87	.00	.00	.00	3.7×10^{-2}	2.0×10^{-2}
Cs^{137}-Ba^{137}	2.68	1.04	0.65	1.73	5.0×10^{-1}	1.7×10^{-2}
Ba^{140}-La^{140}	463.12	127.93	53.21	89.50	1.8×10^{-1}	1.2×10^{-2}
Ce^{141}	416.16	141.26	75.46	168.45	1.2×10^{0}	5.3×10^{-2}
Ce^{143}	109.97	1.80	0.02	.00	5.5×10^{-1}	2.2×10^{-1}
Pr^{143}	509.78	147.09	62.96	109.31	1.0×10^{0}	6.4×10^{-2}
Ce^{144}-Pr^{144}	95.98	36.59	22.45	58.62	2.0×10^{-1}	6.9×10^{-3}
Nd^{147}	183.70	48.48	19.13	30.35	7.6×10^{-1}	5.5×10^{-2}
Pm^{147}	11.08	4.54	2.96	8.10	4.3×10^{0}	1.4×10^{-1}
Pm^{149}	42.22	4.59	0.15	.00	9.9×10^{-1}	2.9×10^{-1}
Sm^{151}	0.08	0.03	0.02	0.05	1.4×10^{1}	4.8×10^{-1}
Total Conc.	4300 $\times 10^{-3}$	1154. $\times 10^{-3}$	567. $\times 10^{-5}$	1244. $\times 10^{-6}$		
Factor above MPC for emergency use for:						
1 day	42.0	10.5	0.044	0.008		
30 days	586.5	169.3	0.79	0.17		

(a) Maximum Permissible Concentration

Table III

Factors by which the Estimated Peak Concentrations of Radioactivity for Specific Isotopes Exceed their Corresponding Emergency MPC Values for Drinking Water at Selected Locations in the Clinch and Tennessee River Systems.

Isotope	Ratio of Concentration to Emergency MPC for 1 Day's Use				Ratio of Concentration to Emergency MPC for 30 Day's Use			
	K-25	Kingston	Watts Bar	Chattanooga	K-25	Kingston	Watts Bar	Chattanooga
Sr^{89}	2.135	0.762	.00432	.00103	53.379	19.037	.108	0.26
Sr^{90}-Y^{90}	2.907	1.127	.00700	.00187	87.200	33.800	.210	0.56
Y^{91}	0.645	0.233	.00133	.00032	16.776	6.049	.035	.008
Zr^{95}	1.300	0.471	.00272	.00066	31.901	11.553	.067	.016
Nb^{95}	0.363	0.150	.00099	.00027	8.092	3.357	.022	.006
Zr^{97}-Nb^{97}	0.087	.00	.00	.00	0.087	.00	.00	.00
Mo^{99}-Tc^{99}	0.288	0.023	.00002	.00	1.343	0.109	.00	.00
Ru^{103}-Rh^{103}	0.336	0.117	.00064	.00015	8.204	2.850	.016	.004
Rh^{105}	0.016	.00	.00	.00	0.046	0.001	.00	.00
Te^{131}	0.806	0.010	.00	.00	1.880	0.022	.00	.00
I^{131}	26.874	6.308	.02085	.00270	299.354	70.266	.232	.030
Te^{132}	0.658	0.066	.00008	.00	3.457	0.349	.00	.00
I^{132}	0.014	0.001	.00	.00	0.014	0.001	.00	.00
I^{133}	1.187	.00	.00	.00	2.200	.00	.00	.00
Cs^{137}-Ba^{137}	0.005	0.002	.00001	.00	0.158	0.061	.00	.00
Ba^{140}-La^{140}	2.572	0.711	.00296	.00050	38.593	10.661	.044	.007
Ce^{141}	0.347	0.118	.00063	.00014	7.852	2.665	.014	.003
Ce^{143}	0.200	0.003	.00	.00	0.500	0.008	.00	.00
Pr^{143}	0.510	0.147	.00063	.00011	7.965	2.298	.010	.002
Ce^{144}- Pr^{144}	0.480	0.183	.00112	.00029	13.910	5.303	.033	.008
Nd^{147}	0.242	0.064	.00025	.00004	3.340	0.881	.003	.001
Pm^{147}	0.003	0.001	.00001	.00	0.078	0.032	.00	.00
Pm^{149}	0.043	0.005	.00	.00	0.145	0.016	.00	.00
Totals	42.018	10.502	.04356	.00808	586.474	169.319	.794	.167

8.c ESTIMATES OF DISPERSION OF AIRBORNE ACTIVITY FROM THE ORR*

INITIAL CONDITION

Release of airborne activity from the ORR will occur only if one of two types of accidents occurs.

One of the accidents would involve the melting of some or all of the fuel plates which would release the volatile fission products which had been formed during the operation of the reactor. It is not at all certain that it would be possible for all the contained total or volatile fission products to become airborne, since some may stay in the water and some may condense on the structural materials in the vicinity of the reactor. To estimate the maximum hazard, it has been assumed that all the volatile fission products in the reactor escape and become airborne. The cloud of all the halogens and noble gases will be referred to as the cold cloud.

The other type of accident which could result in the liberation of airborne contamination is the reaction of the aluminum in the fuel plates with the water in the moderator. This is a high temperature reaction and would result in sufficient energy to disperse the stored activity in the fuel plates into the air. Whether or not this reaction would occur depends on a variety of conditions which include the rate of heat generation in the fuel plates, the impurities in the metal, the rate of dispersion of the metal into the water, and the effects of oxide formation on the metal surfaces.

Doubtless there is only a small chance of the reaction going to completion. In view of the many unknown variables concerned in the case of a nuclear accident, it has been assumed that a very large temperature rise in

*Prepared by D.R. Purdy and R.F. Myers, U.S. Weather Bureau, Oak Ridge, Tenn.

the fuel plates is followed by an aluminum-water reaction. The resulting cloud of vapor containing the fission products will be called the hot cloud.

THE HEIGHT OF RISE OF THE HOT CLOUD

The hot cloud, initially about 30 cubic meters in volume at a pressure of about 33 atmospheres with a temperature of about 2570°K, will rise to some height where it will be in equilibrium with the atmosphere and spread by eddy diffusion. The height of rise has been calculated by Sutton's method used in a previous report[27]. In the night time or stable case, the hot cloud will rise to about 800 meters with an initial volume of 125 cubic meters. In the day time or unstable case, the cloud is assumed to level off at about 1500 meters above the ground which is 300-500 meters above the average convective cloud base in daytime at this site. The initial volume of the daytime hot cloud at this altitude will be about 1350 cubic meters.

INITIAL CONCENTRATION OF FISSION PRODUCTS IN THE CLOUD

The activity in the reactor following shutdown, calculated from the equation given by Mills (18) contains no augmentation due to the runaway excursion. The equation:

$$A = 11.15 \frac{(\text{Power in Watts})}{t^{0.2}\,(\text{sec.})}$$

Where A is the activity in curies assuming an average fission product energy of 1 Mev.

At one second after shutdown, A is found to be 5.5×10^7 curies at 5 megawatts, or 1.1×10^8 curies at 10 megawatts.

The radii of the hot puff at various distances from the reactor site were calculated by the methods used by Hurwitz [28] in his development of Sutton's diffusion equation.

The parameters used in the calculations of concentrations and total dose are given in table 1:

TABLE 1

	Avg. night (Stable) hot cloud	cold cloud	Avg. day (unstable) hot cloud	cold cloud
h initial height (m)	780	25	1500	25
C virtual diff. coef.	.055	.13	.114	.224
n stability parameter	.35	.35	.23	.23
V_o initial volume of cloud	124	2.6×10^4	1360	2.6×10^4 cubic meters
u wind speed m/sec.	1.5	1.5	8.0	2.3
x_o distance correction (m)	118	1020	93	281

Initial concentration of activity in the cloud (curies/m^3) at 1 second

	Power	hot cloud, night	hot cloud, day	cold cloud
Total f.p.	5 mw.	4.4×10^5	4.0×10^4	
30 selected f.p.	5 mw.	3.5×10^4	3.2×10^3	
"	10 mw.	7.1×10^4	6.5×10^3	
Total f.p.	10 mw.	9.0×10^5	8.2×10^4	
Iodines, total	5 mw.			42
Iodines, total	10 mw.			85

GROUND CONCENTRATIONS

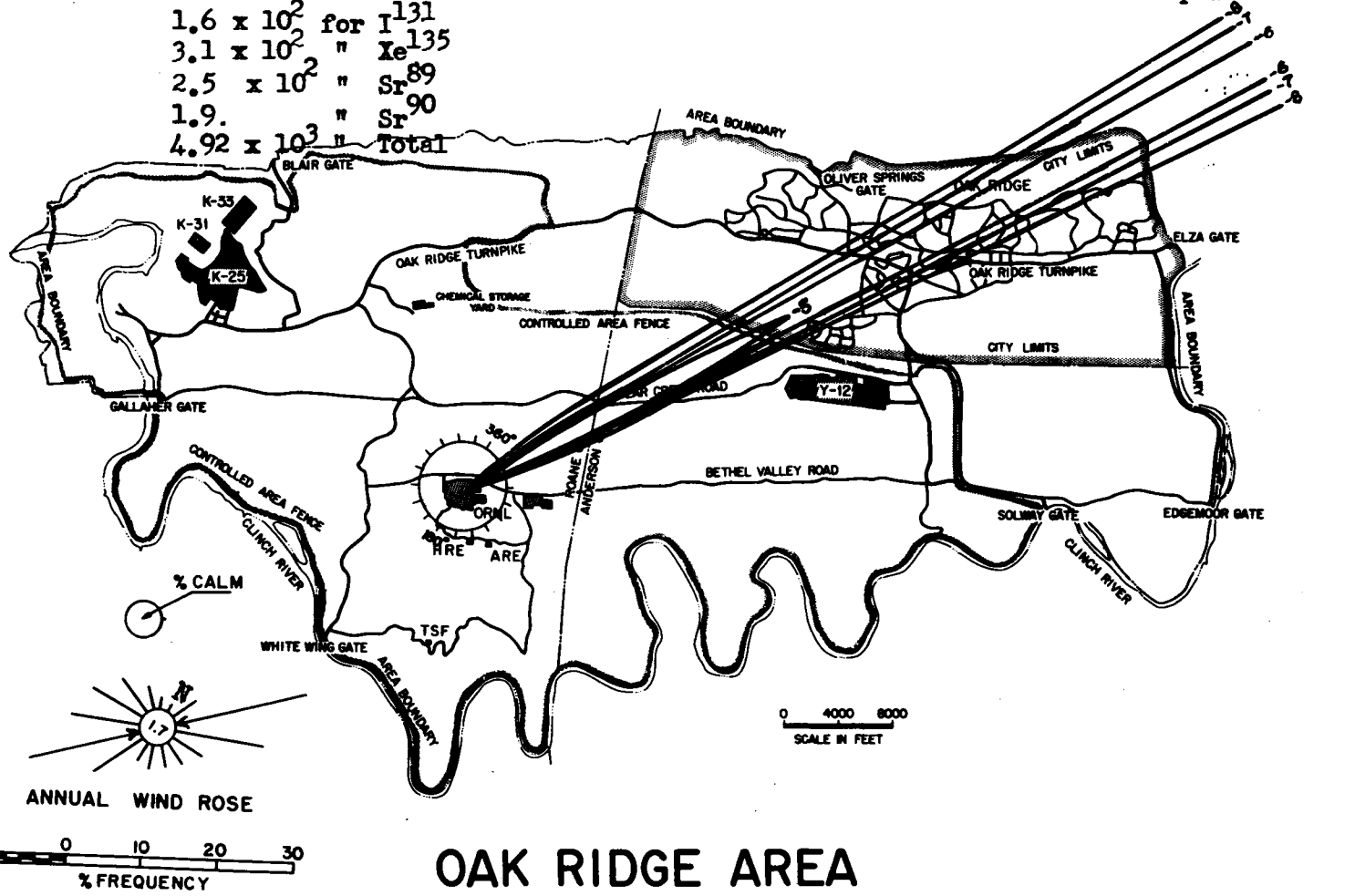
The contours of the maps (Fig. 21 and 22) represent the ground concentrations resulting from the diffusion of the cloud originating at the reactor site. The values on the curves indicate powers of 10 and when multiplied by the appropriate source strength as shown in the upper left hand corner of each map gives the concentrations in curies/$meter^3$. The curves are plotted in the direction of the nearest population center, which is not necessarily the direction of the most frequent wind. A wind rose showing the annual frequency of wind directions is displayed in the lower left hand corner of each map for the purpose of indicating the

COLD CLOUD
Average Night

For curies/meter^3, multiply by

1.6×10^2 for I^{131}
3.1×10^2 " Xe^{135}
2.5×10^2 " Sr^{89}
1.9 " Sr^{90}
4.92×10^3 " Total

numerals indicate powers of 10



OAK RIDGE AREA

Three major plant sites and the city of Oak Ridge. The total area encompasses 59,000 acres.

Figure 21

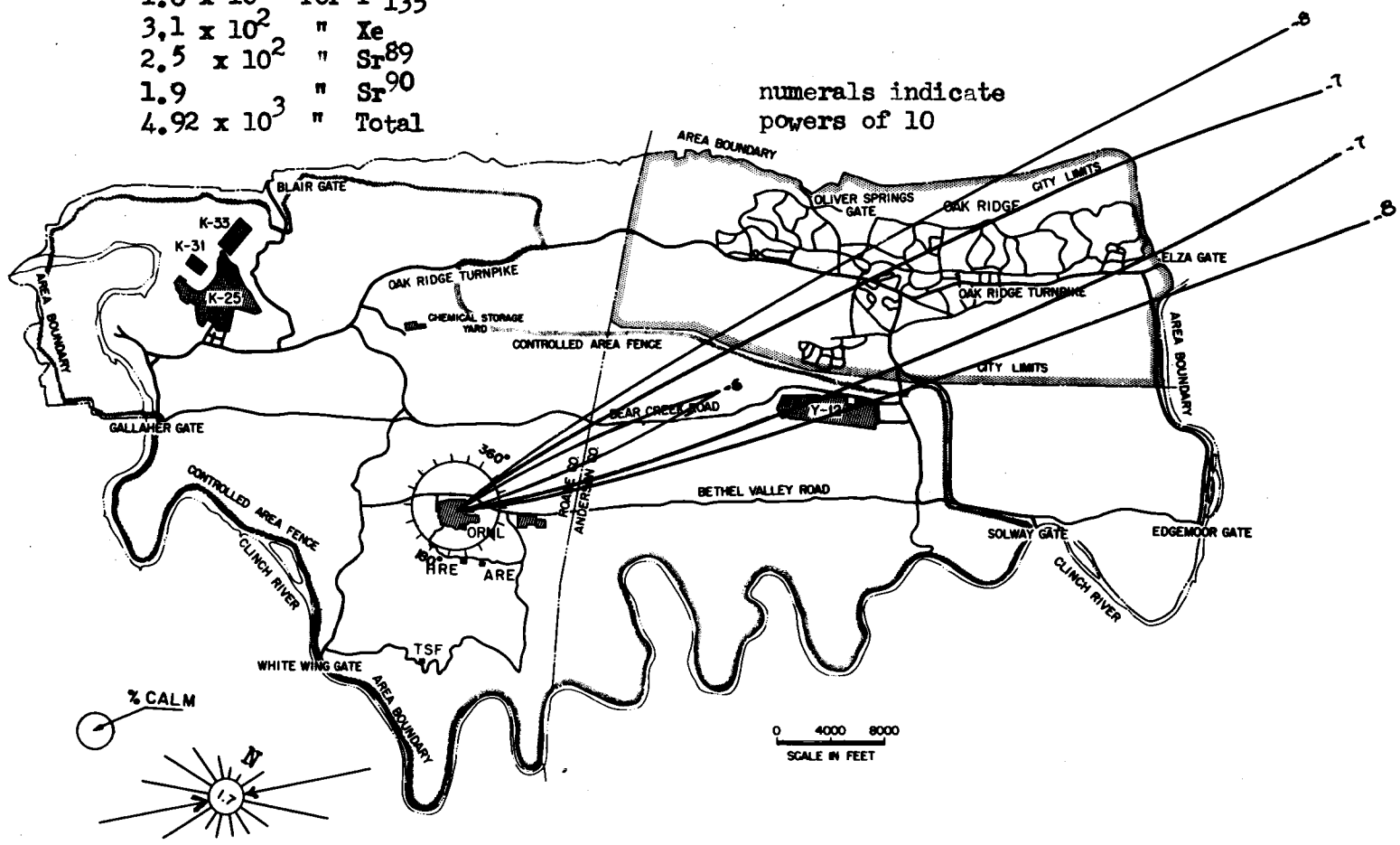
See errata

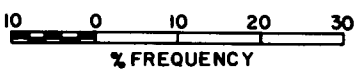
COLD CLOUD
Average Day

* For curies/meter^3, multiply by

1.6×10^2 for I^{131}
3.1×10^2 " Xe^{135}
2.5×10^2 " Sr^{89}
1.9 " Sr^{90}
4.92×10^3 " Total

numerals indicate powers of 10





OAK RIDGE AREA

Three major plant sites and the city of Oak Ridge. The total area encompasses 59,000 acres.

Figure 22

* See errata

frequency with which the clouds will travel in certain direction. It should be pointed out that the wind roses show the directions from which the wind is blowing, so the opposite direction is the direction toward which the cloud will travel.

For the cold cloud, two cases should be considered. In the first case, the volatile fission products released into the air of the building are contained by the building. The ventilating fans are shut off by the air monitors and the building evacuated. The activity then leaks out into the free atmosphere at a rate of 10% per day. Figures 21 and 22 show the resulting concentrations during typical day and night conditions.

In the second case, the volatile fission products are mixed with the building air and released to the atmosphere in a period of 15 minutes by the ventilating fans. The emission rate is increased by a factor of about 1000 but the exposure time is decreased by a similar amount, so the resulting distances out to which the MPC is exceeded remain the same in either case. The concentration contours are very similar to those of Figures 21 and 22, but the duration at any one spot is limited to 15-70 minutes out to 80 miles. Figure 23 shows the time required to inhale a 25 R dose as well as the time of passage of the cloud. The times were computed by dividing the MPD = 470 curies by the product of the breathing rate in $\frac{m^3}{min}$ and the concentration in curies/m^3. The scales on the graph fit the 15 minute case, but as mentioned in the previous paragraph the distances remain unchanged in either case. These distances, as can be seen from Figure 23 are, at night 38 mi. for 10mw and 25 mi. for 5mw. During the day, the distances are 7.4 mi. for 10mw and 4.7 mi. for 5 mw.

* See note

Table II shows the total integrated dose in curie sec/m^3 if the cloud were released as a single puff at 25 meters elevation and having initially the volume of the building. The maximum permissible dose of .94 curies sec/m^3 is exceeded ~~only~~ at night for distances under 36 miles and during the day under 6.5 miles for 10 mw of power. The probable consequences of the release of the volatile fission products in a cold cloud in some fashion lie within the brackets of the total integrated dose for the instantaneous puff and the slow leak of the building air for a few days.

Reference to Figure 24 shows that for the hot cloud case, the time of passage of the cloud is everywhere less than the time required to inhale the maximum permissible dose. There is thus no internal dose hazard in this case. Table II shows the external dose resulting from the passage of the hot cloud overhead. The calculations follow the work of Hurwitz (28). A plot of the results is shown in Figure 25, which shows that the exposure is above 25 R out to 5.5 miles at night and 0.65 miles during daytime conditions. Figure 25 also shows the results of calculations based on the method given in reference 31. The exposure is above 25 R out to 8 miles at night and 2.3 miles during the day. ~~These distances are compatible with those found by reference 30, which gives 3.6 miles at night and 0.9 for day time~~. All distances mentioned are for a power level of 10 mw. For the 5 mw operations, the intensity will be less by a factor of 2.

See errata

[redacted]

COLD CLOUD

* TIME REQUIRED TO INHALE 25 R DOSE

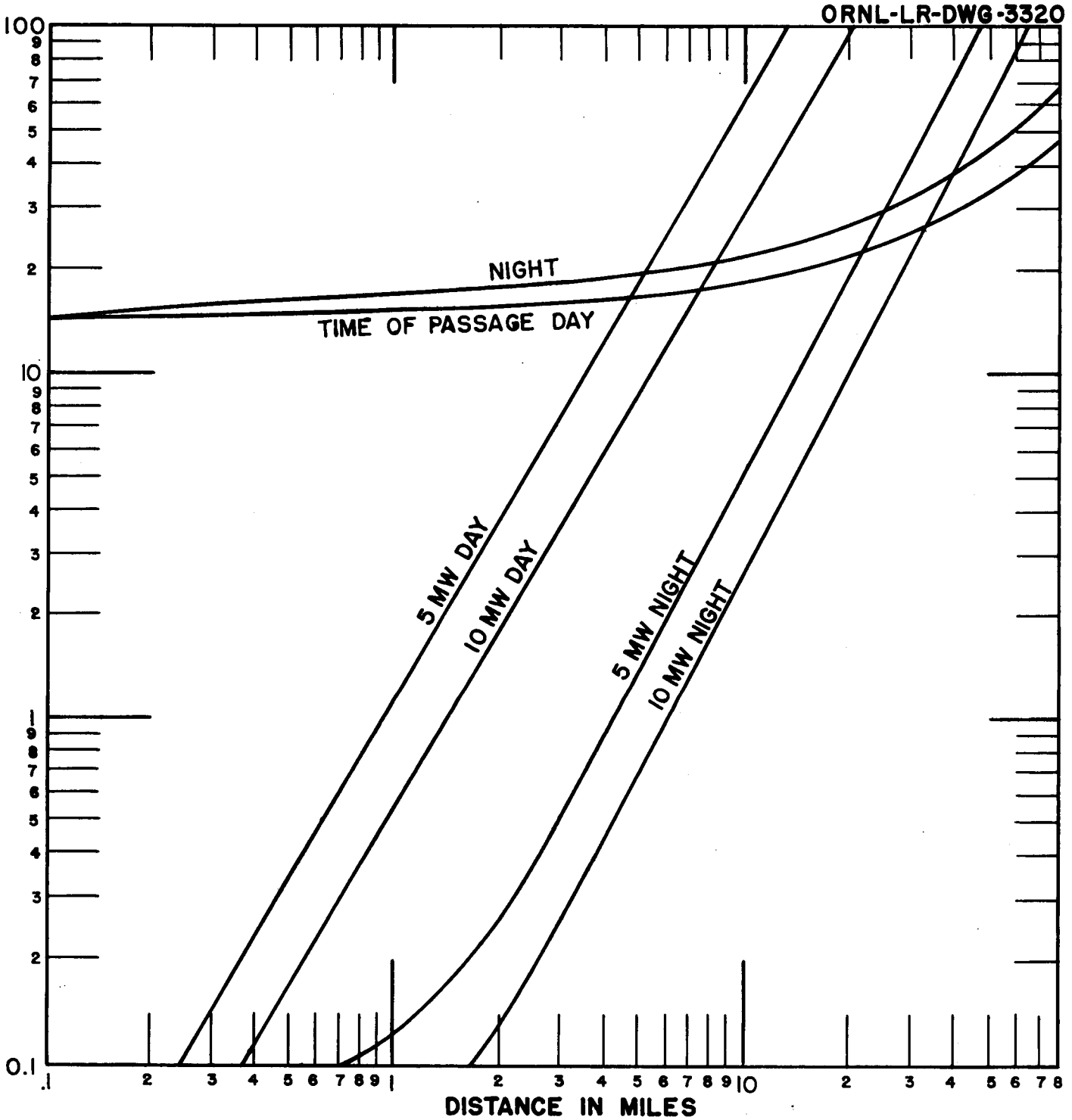


FIG. 23

* See errata

[redacted]

HOT CLOUD

TIME REQUIRED TO INHALE 25 R DOSE

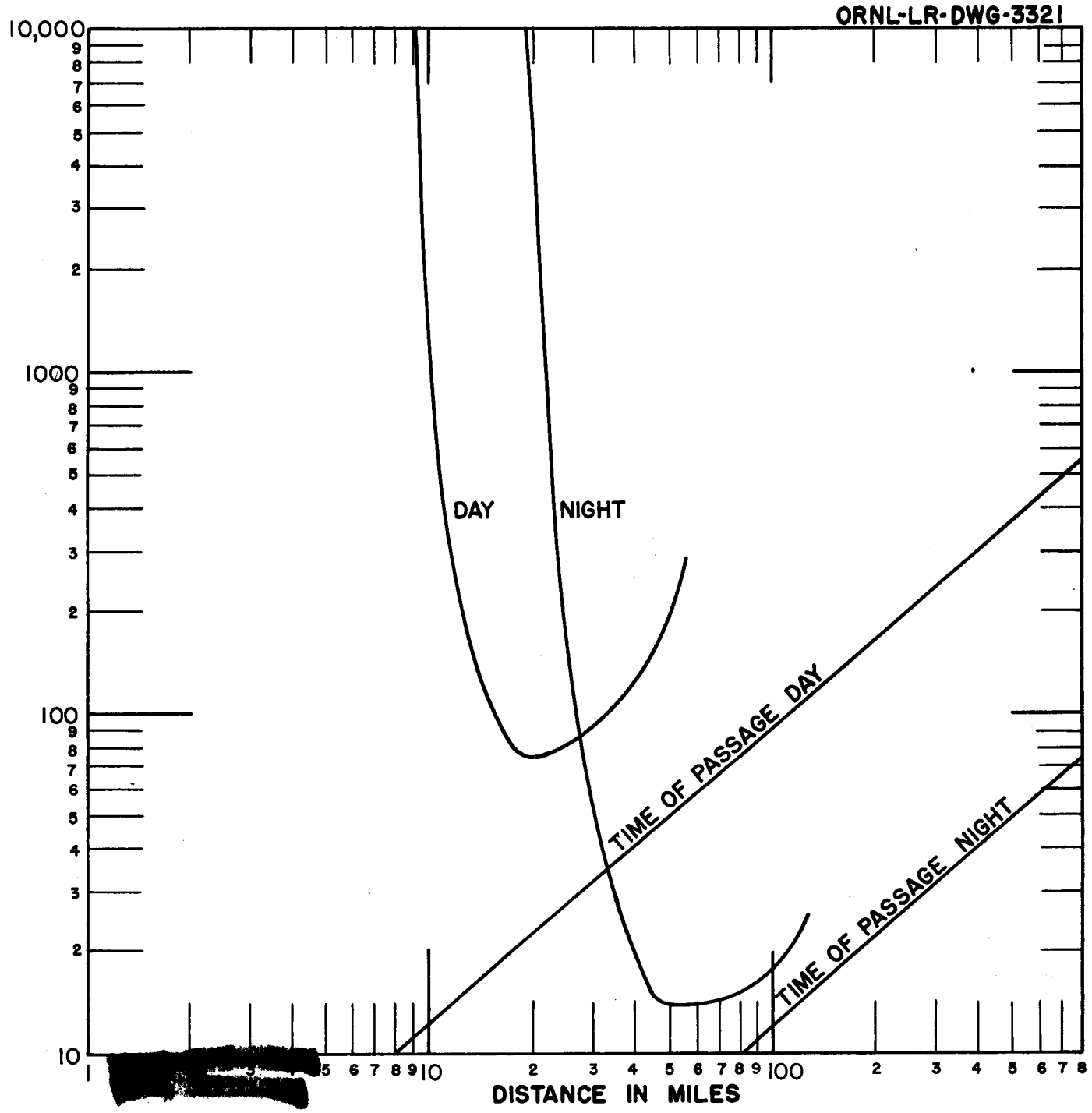


FIG. 24

[redacted]

See errata

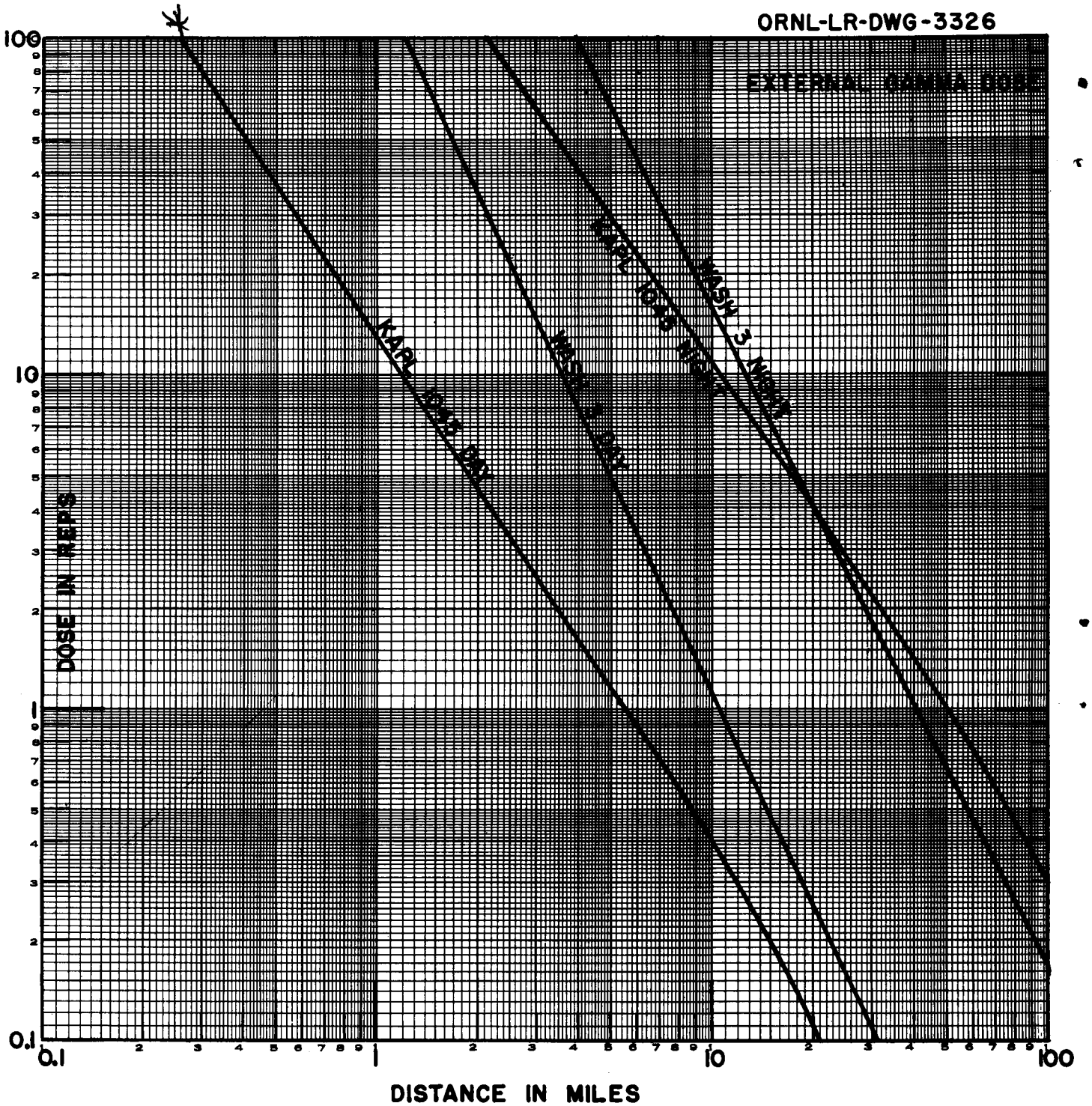


FIG. 25

[redacted]

[redacted]

*TABLE II

Internal Doses in Curie -sec/m^3 and External Doses in Reps

Average Night

(1) During passage of the hot cloud

Distance	5 mw	10 mw	External dose in reps due to 50% total activity (corrected for decay) 5 mw	10 mw
.61 miles			6.14×10^{1}	1.28×10^{2}
2.2			1.40×10^{1}	2.80×10^{1}
4.1			6.50×10^{0}	1.31×10^{1}
5			5.0×10^{0}	1.0×10^{1}
10	2.16×10^{-10}	4.4×10^{-10}	1.94×10^{0}	3.85×10^{0}
13	8.8×10^{-8}	1.8×10^{-7}	1.34×10^{0}	2.68×10^{0}
20	4.7×10^{-5}	9.4×10^{-5}	7.25×10^{-1}	1.43×10^{0}
50	1.1×10^{-3}	2.2×10^{-3}	1.72×10^{-1}	3.44×10^{-1}
80	9.6×10^{-4}	1.9×10^{-3}	7.45×10^{-2}	1.49×10^{-1}
150			2.50×10^{-2}	4.97×10^{-2}
300			6.80×10^{-3}	1.36×10^{-2}
Max	1.2×10^{-3}	2.4×10^{-3}		
MPE	.94	.94	25	25

(2) During passage of the cold cloud
TID in curie-sec/m^3

Distance	5 mw	10 mw
.19	1.67×10^{-1}	2.4×10^{-1}
.62	9.7×10^{-2}	1.9×10^{-1}
2.2	2.7×10^{-2}	5.4×10^{-1}
4.1	1.4×10^{-2}	2.4×10^{-2}
5	8.9×10^{-2}	1.8×10^{-2}
10	3.0×10^{-3}	6.0×10^{-2}
13	2.1×10^{-3}	4.2×10^{-3}
20	1.1×10^{-3}	2.2×10^{-3}
50	2.4×10^{-4}	4.8×10^{-4}
80	1.2×10^{-4}	2.4×10^{-4}
Max	5.0×10^{-1}	1.0×10^{-1}
MPE	.94	.94

Average Day

(3) During passage of the hot cloud
Internal dose in curies-sec/m^3 from inhalation of 30 isotopes

External Dose in reps due to total activity (corrected for decay)

* See errata

Distance	5 mw	10 mw	5 mw	10 mw
.61 miles			4.5×10^{0}	9.0×10^{0}
2.2			7.0×10^{-1}	1.4×10^{0}
4.1	8.4×10^{-13}	1.7×10^{-12}	2.8×10^{-1}	5.6×10^{-1}
5	4.3×10^{-9}	8.6×10^{-9}	2.0×10^{-1}	4.0×10^{-1}
10	2.8×10^{-3}	5.6×10^{-3}	6.0×10^{-2}	1.2×10^{-1}
13	1.3×10^{-2}	2.6×10^{-2}	4.1×10^{-2}	8.1×10^{-2}
20	4.5×10^{-2}	9.0×10^{-2}	1.9×10^{-2}	3.7×10^{-2}
50	3.9×10^{-2}	7.8×10^{-2}	3.01×10^{-3}	6.01×10^{-3}
80	2.1×10^{-2}	4.1×10^{-2}	1.26×10^{-3}	2.31×10^{-3}
Max	4.5×10^{-2}	9.0×10^{-2}		
MPE	.94	.94	25	25

*(4) During passage of the cold cloud

Internal dose in curies sec/m^3 from inhalation vol. f.p.

Distance	5 mw	10 mw
.10 miles	1.4×10^{-2}	2.8×10^{-2}
.61	2.1×10^{-2}	4.2×10^{-2}
2.2	3.0×10^{-3}	6.0×10^{-3}
4.1	1.1×10^{-3}	2.2×10^{-3}
5	7.7×10^{-4}	1.5×10^{-4}
10	2.2×10^{-4}	4.5×10^{-4}
13	1.5×10^{-4}	3.0×10^{-4}
20	7.1×10^{-5}	1.4×10^{-4}
50	1.4×10^{-5}	2.8×10^{-5}
80	6.1×10^{-6}	1.2×10^{-5}
Max	2.0×10^{-2}	4.0×10^{-2}
MPE	.94	.94

PARTICLE FALLOUT

In the event of a reactor catastrophe, there is reason to believe that particulate material may be formed in the chain of events leading to the reactor shutdown and escape of the active material into the air. This particulate matter will behave as a gas if the particle sizes are below 0.1 micron. For initial heights of several hundred feet or more, particles up to 10 microns in diameter will not settle rapidly enough to influence greatly the calculated concentration of a single puff of diffusing material which

* See errata

contains a wide variety of particle sizes.

The significance of the particle fallout is the almost certain elimination of the skip distance between the source and the point at which the first gas diffuses to the ground from the elevated puff of the initial cloud. Following Drinker and Hatch (29) the rates of fall have been calculated for various particles which might be produced in a catastrophe. The particles have been assumed to have a density of 3. The results are presented in Table III.

* TABLE III

Distances at which particles will reach the ground (miles)

Particle diameter (microns	1000	100	10	1
Hot cloud, day	1.1	14.7	100	1
Hot cloud, night	0.2	2.5	100	
Cold cloud, day	.02	0.1	12	100
Cold cloud, night	.02	0.2	17	100

It is evident that considerable contamination can occur as a result of particulate material out to as much as 20 miles if a large portion of the activity was present in the form of droplets or solid particles in the ranges shown above. No estimate is made here of the hazard since it is not possible at this time to assign any value to the size or activity distribution which would have a foundation in experience. Another case will be treated which will give some measure of the magnitude of this type of contamination if all the activity were deposited on the ground.

RAINOUT

If an accident occurs during a period in which some form of precipitation dominates the weather of the area, the cloud containing the activity will be subjected to scrubbing by the falling raindrops. Large scale downdrafts

* See errata

which occur with thunderstorms would also tend to bring the cloud to the ground. In any case an integration of Sutton's equation for an instantaneous puff would give the distribution of activity on the ground after any given time period with allowance for the spreading of the puff by eddy diffusion. This has been carried out by Holland (30) to give the formula for ~~complete~~ maximum deposition of the material in a single puff:

$$D = \frac{Q}{\pi^{\frac{1}{2}} e C_y x^{2-(n/2)}}$$

where D is the deposition in curies/meter^2
Q is the total activity in the cloud
C is the appropriate virtual diffusion coefficient (Table I)
x is the distance from source of accident
n is a stability parameter (Table I)
e is 2.71828

The deposition of activity, assuming the puff contains all of the contained activity for the hot cloud or all of the volatile activity for the cold cloud is shown in Table IV.

TABLE IV

Contamination Intensity in curies/m^2 for "rainout". (Multiply by 10 to get r/hr.)

Distance in miles	Hot Cloud				Cold Cloud			
	Day		Night		Day		Night	
	5 mw	10 mw	5 mw	10 mw	5 mw	10 mw	5 mw	10 mw
.61	29	58	84	168	8	16	24	48
2.3	3	6	8	16	.7	1.4	1	2
4.1	.8	1.6	3	6	.2	.4	.3	.6
5.0	.5	1.0	2	4	.1	.2	.2	.4
10	.1	.2	.5	1	.04	.08	.06	.12
13	.09	.18	.3	.6	.02	.04	.04	.08
20	.04	.08	.1	.2	.01	.02	.02	.04

* See errata

[redacted]

The case where a shower or thunderstorm occurs as the hot cloud passes overhead at night is certainly the most hazardous from the standpoint of external radiation. Out to about 10 miles from the source the dose rate would be above 10r/hr. with a total dose of more than 50 r. One may conclude that in event of rainout of the cloud with total deposition, the radius of hazard to the population would extend to about 8 miles.

These estimates are only order of magnitude calculations which could be influenced by the type of accident and by the characteristics of the raindrops which are falling through the cloud.

SUMMARY

Calculations of ground concentrations and exposures resulting from hypothetical accidental failures of the ORR are presented. Under the case where some or all of the fuel plates melt, releasing the volatile fission products, the maximum permissible internal exposure is exceeded for the cold cloud out to about 38 miles. In the event of rainout at the time of the hot cloud type of accident the radius of hazard in which the populace would be exposed to a total dose of more than 50 r is 16 miles.

For the hot cloud the 25 r external gamma dose is exceeded out to about ~~8~~ 3 miles.

* See errata

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