BAW-109

LYNCHBURG TEST REACTOR

Critical Experiment
Hazard Evaluation

November 1959

By
T. C. Engelder
P. F. Schutt
Critical Experiment Laboratory
Physics and Mathematics Department

SUBMITTED TO THE
UNITED STATES ATOMIC ENERGY COMMISSION
BY
THE BABCOCK AND WILCOX COMPANY

ACKNOWLEDGMENT

The authors wish to thank D. V. P. Williams and other members of the Critical Experiment Laboratory for many helpful discussions regarding the design of the experiments and the preparation of this report. We particularly thank L. G. Barrett, R. H. Clark, and J. M. Funderburg for supplying the detailed design features of the Lynchburg Pool Reactor (LPR) and information on operating experience. The bulk of the nuclear calculations were performed by W. A. Wittkopf and A. E. Paulson, and the engineering design was coordinated by H. G. Sommer.

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ABSTRACT

This report summarizes the hazards associated with the operation of the Lynchburg Test Reactor (LTR) critical experiments. The LTR critical experiment program is outlined and the design of the experiment is described. It is proposed to construct a basic assembly of MTR-type fuel elements and beryllium reflector elements, with four 6 in. sq. test holes, normally filled with aluminum. The critical mass and extensive flux distributions will be obtained with shim rods in and out and with simulated test loops inserted in the test holes. The experiments will provide basic data to check two-dimensional calculational methods and provide information needed in the design of the LTR. The experiments will be performed in the LPR, using the LPR fuel elements, control system, and instrumentation.

Described in detail are operating procedures, normal operating hazards, health physics procedures, and steps to be taken in the event of unusual conditions. The Laboratory Emergency Plan is described, and operating limitations are stated.

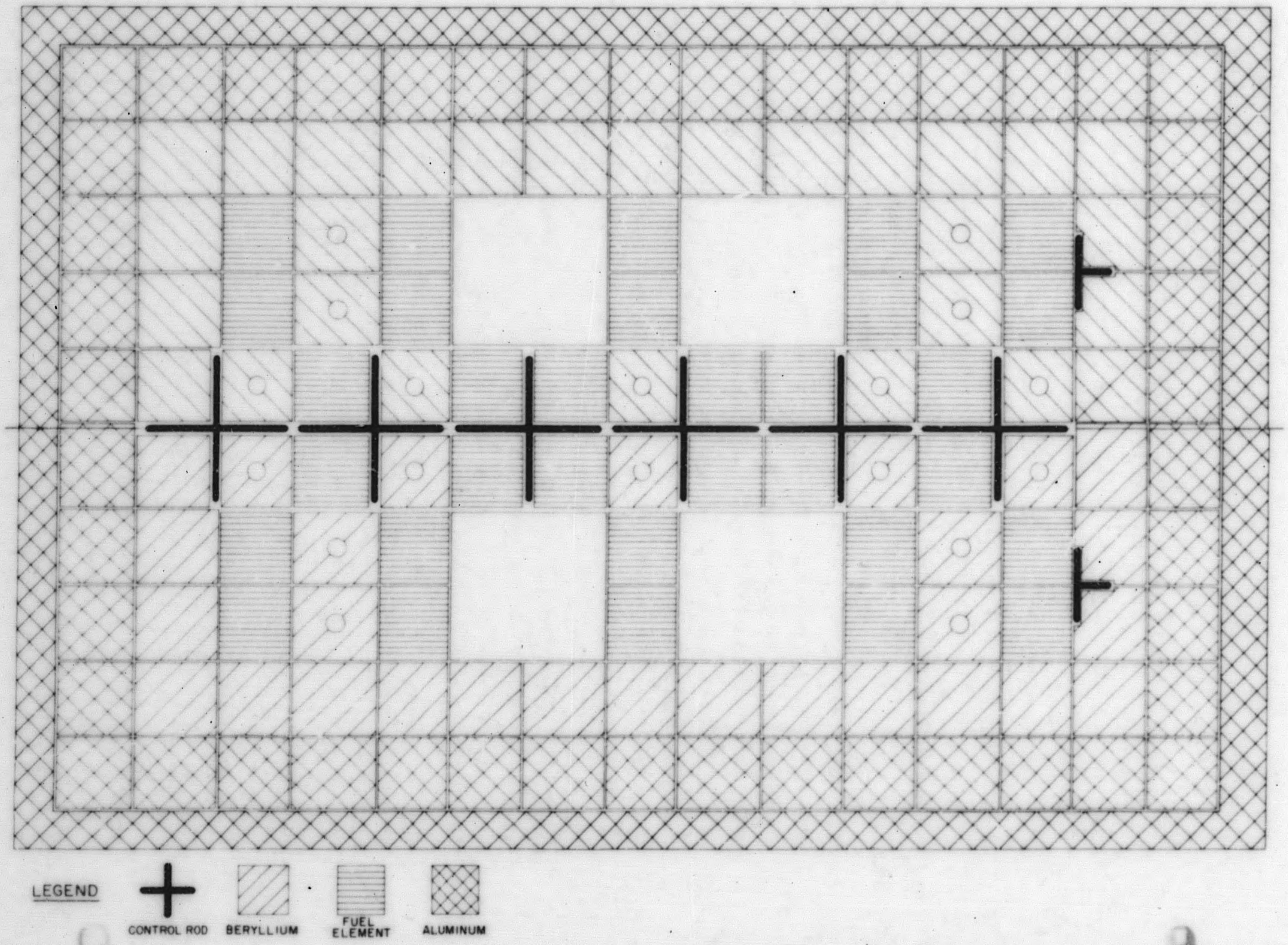
Various accidents are discussed, and the maximum credible accident is postulated as the sudden addition of 2% excess reactivity coincident with the plugging of some coolant channels and melting of some of the fuel. The release of volatile fission products from the pool and laboratory building is calculated under various conservative assumptions. It may be concluded that a 2% excursion would not endanger the nearest permanent resident off the site or the employees outside the laboratory.

I. INTRODUCTION

In July of 1959, the United States Atomic Energy Commisiion selected, as a basis for negotiation,[1] The Babcock and Wilcox Company's (B&W) proposal to the Division of Reactor Development for irradiation services. To supply these services B&W proposes to construct and operate a Nuclear Fuels Testing Reactor to be known as the Lynchburg Test Reactor (LTR). In its conceptual design, the LTR is a 50 MWh reactor utilizing beryllium as reflector material and light water as coolant and moderator. A flexible arrangement of ETR-type fuel elements and beryllium reflector elements provide various sized test holes for test loops and capsule irradiations. A typical core arrangement with four large irradiation spaces is shown in Figure 1.

A series of critical experiments has been proposed to provide accurate knowledge of the flux available in the test holes per unit of reactor power. (The reliability of nuclear calculations on cores of this type is not established.) These experiments will also provide design information on rod worths, reactivity effects of various materials in the test holes, and the required fuel element loading. The critical experiments — scheduled to begin December 15, 1959 — will require several months time. They will be performed in the pool of the LPR, located at the B&W Critical Experiment Laboratory near Lynchburg, Virginia.

FIG. 1: TYPICAL LTR CORE ARRANGEMENT



II. SITE LOCATION AND LABORATORY DESCRIPTION

A. SITE LOCATION

The Critical Experiment Laboratory is located in Campbell County, Virginia, approximately 4.5 air miles east of Lynchburg, the nearest principal city. A map of the site is shown in Appendix A.

B. LABORATORY DESCRIPTION

The major experimental areas of the Critical Experiment Laboratory are two shielded bays and a reactor wing. Other facilities include control rooms, subassembly rooms, physics laboratory, chemistry laboratory, counting room, change room, electronics shop, machine shop, and offices. A photograph and floor plans of the laboratory are presented in Appendix B.

Critical experiments for the N. S. Savannah reactor are currently being performed in Bay No. 1.[2] Bay No. 2 contains two critical experiment facilities: a water tank used for studies on water moderated thorium-oxide uranium-oxide fuel pins,[3] and a split bed assembly for the graphite moderated LMFRE critical experiments.[4] Both experiments utilize a common control console.

The reactor wing houses the LPR, a 200-kw swimming pool research reactor.[5] Recently an application has been submitted to the U. S. Atomic Energy Commission to increase the power level of the LPR to 2 MW; however, the ultimate disposition of this application will in no way affect the program described.[6]

The LTR critical experiments will be performed in the LPR, utilizing a major portion of that facility. The LPR is described further in Appendix B. The modifications of this facility and the design of the LTR critical experiments are discussed in Chapter IV.

C. LABORATORY ADMINISTRATION

The Critical Experiment Laboratory is operated by the Physics and Mathematics Department of B&W's Atomic Energy Division. The Laboratory Supervisor, responsible for all operations within the laboratory, will assign a Group Supervisor who will direct all LTR critical experiment operation. Other administrative details specifically directed to the LTR critical experiments are discussed in Chapter V.

D. LABORATORY OPERATIONS

The Critical Experiment Laboratory began operation in 1956. Since then, six major critical experiment programs have been undertaken without incident — a total of 2200 operating runs. In addition, the LPR has been operated approximately 250 times since it initially reached criticality on September 19, 1958. Of the 40 members on the Critical Experiment Laboratory Staff, 20 are scientists — including nine AEC-licenced operators — and the remainder are technicians and supporting personnel.

III. EXPERIMENTAL PROGRAM

A. GENERAL OBJECTIVES

The major objectives are (1) to construct a critical assembly resembling as nearly as possible, the typical LTR core arrangement shown in Figure 1, (2) to obtain criticality and flux data that can be used to check two-dimensional calculational methods, and (3) to provide experimental information for the preliminary design of the reactor.

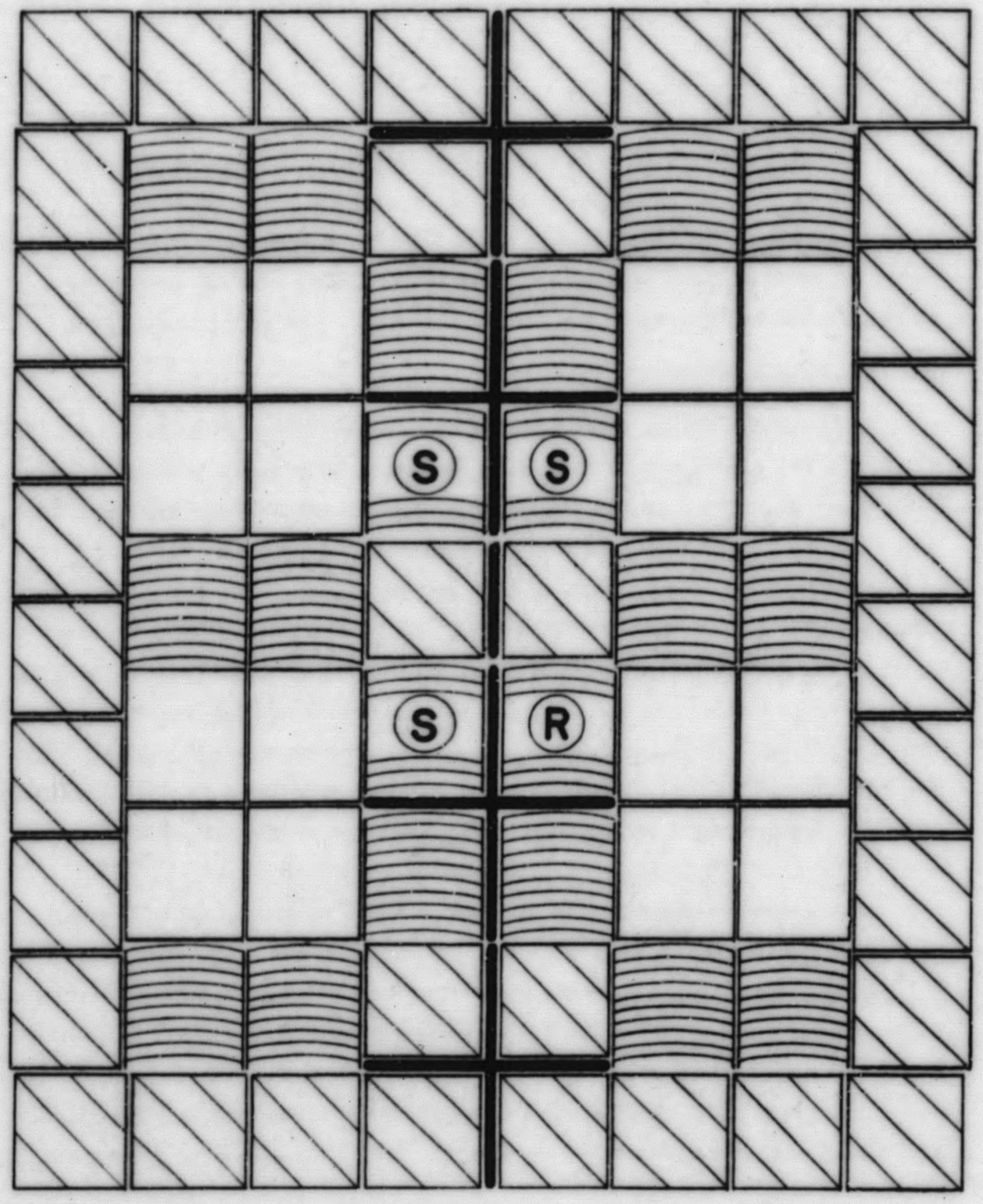
B. BASIC ASSEMBLY

Although the basic assembly to be studied (Fig. 2) doesn't exactly duplicate the expected LTR loading shown in Figure 1, it does contain the major features, and represents a reasonable nuclear mockup consistent with the availability of fuel elements and beryllium.

The various components of this assembly are described fully in Chapter IV. These experiments will utilize the fuel elements now being used in the LPR. The location (Fig. 2) of fuel elements containing the safety rods and the regulating rods is typical, and may be changed during the experiments providing their reactivity worths are consistent with the limitations set forth later. Solid beryllium elements, on loan from Oak Ridge National Laboratory, constitute the reflector, and several other beryllium elements, are located within the core.

In the basic assembly the five shim rod locations contain solid aluminum rather than cadmium shim rods in order to mockup the rod-followers. This mocks up the rod "out" condition. In all of the assemblies, the rigidly fixed shim rods cannot be moved until the fuel has been removed. The four test hole spaces are approximately 6 in. square, are filled with four solid blocks of aluminum. These are also rigidly fixed and cannot be removed before the fuel elements are removed.

FIG. 2: BASIC ARRANGEMENT OF LTR CRITICAL EXPERIMENT CORE



On the basis of the nuclear calculations summarized in Appendix E, this configuration would be somewhat super-critical. Therefore, before constructing the assembly, each full element will be loaded with a number of poison shims estimated to make the reactor critical with an excess reactivity less than the worth of the regulating rod when the core is completely assembled. The fuel element shims are described in Chapter IV, and the loading sequence and handling of the shims are covered in Chapter V.

After the basic assembly has been shimmed to criticality, the worth of the safety and regulating rods will be measured to assure safe operation.

C. FLUX MAPPING

In the basic configuration, and others to be described later, the power distribution in the fuel elements and the flux distribution in the beryllium and aluminum will be extensively mapped. Standard practices will be followed using foils or wires of gold, dysprosium, indium, U-235, etc. Flux mapping in the fuel elements and various water gaps will be done by mounting the foils or wires on lucite stringers which can be remotely loaded into the appropriate water gaps. The mapping in the beryllium and aluminum pieces will be accomplished by removing these elements and placing foils or wires in machined slots inside the elements. The sequence of removing and replacing these elements will follow that described in Chapter V. In some cases small holes may be drilled in the aluminum elements to facilitate wire irradiations without removing the element.

D. SHIM ROD EFFECTS

The reactivity worth of the cadmium shim rods will be measured in various combinations of "in", "out", and "partially inserted" positions. Each configuration will be considered as a new assembly, and incremental loadings will be made as described in Chapter V. Each assembly will require a different arrangement of shims in the fuel elements so that the final configuration has an excess reactivity of less than the worth of the regulating rod. The three types of shim rods representing the "in", "out", and " partially inserted" conditions are described in Chapter IV.

In the first set of assemblies the reactivity worth of each type of shim rod will be measured, individually in this order: the central blade, the outer cross-shaped blade, then the inner cross-shaped blade. In this sequence, fuel element shimming will be accomplished by removing poison shims, and, possibly, adding some fuel shims for the case of the more powerful inner cross-blade. As these assemblies are built, various flux traverses will be made, as discussed previously.

The importance of shim rod interactions will be studied in the next set of assemblies, which will be built with various combinations of two and three shim rods inserted. The number of combinations will be restricted by time and limited to configurations that can be made critical with 80 gm of U-235 per fuel element, in the form of fuel shims or 10% reactivity, whichever is less.

Some assemblies may be built with one or more shim rods "partially inserted". To mockup this condition, special shim rods containing a stainless steel and a cadmium section will be fully inserted and locked in the core. The shim rod will not be moved or partially inserted in any of these experiments.

E. TEST HOLE EXPERIMENTS

The reactivity worth of inpile sections of typical test loops, and their effect on the flux distribution, will be measured by replacing the four solid aluminum elements in one or more of the 6 in. sq test hole positions with the special test inserts described in Chapter IV.

Each configuration containing a test insert will be treated as a new assembly and loaded in increments. The first assemblies will have only one test insert, the other test holes being filled with solid aluminum, as in the basic assembly. If time permits, two or more test holes will be filled with test inserts to measure interactions.

F. ADDITIONAL EXPERIMENTS

Additional experiments may be necessary during the design of the LTR. These experiments would be similar to those already described and would not introduce new hazards. Typical possibilities are:

1. The addition of several solid aluminum reflector elements outside the beryllium reflector to obtain their reactivity effect and the flux change in the beryllium reflector.

2. Flux mapping in the water region around the assembly to obtain shielding data.

The safety of each experiment described as well as those to be added will be thoroughly reviewed in advance by the B&W Safeguard Review Committee, whose functions are described in Chapter V. No experiment will be started without approval of the Committee.

IV. DESIGN OF EXPERIMENT

The LTR critical experiments will be performed in the LPR, utilizing LPR equipment unless otherwise stated. A new grid plate will be installed adjacent to the present LPR grid plate. The only other major change from LPR operations involves the nature of the core itself.

A. POOL AND GRID PLATE

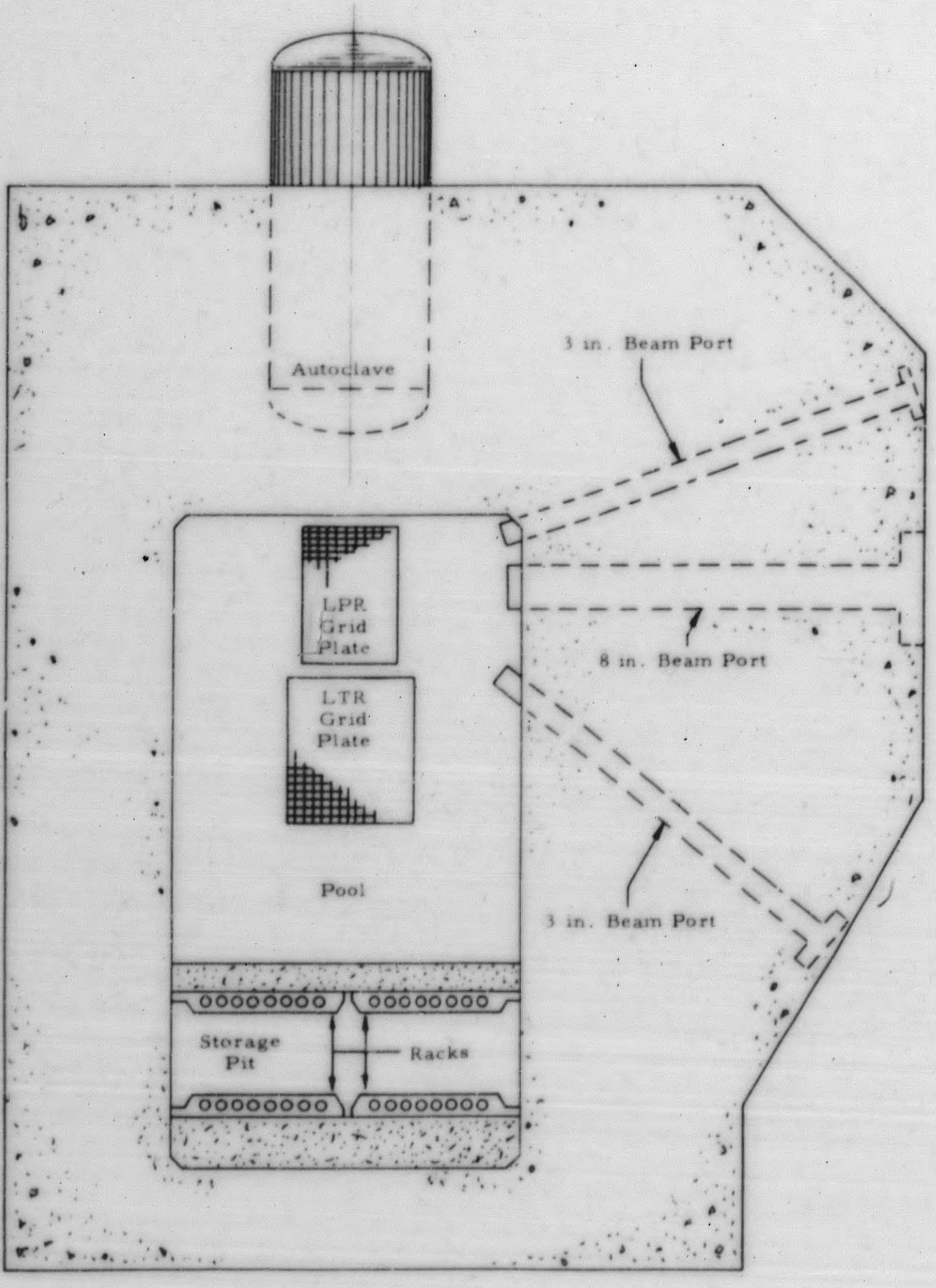
Figure 3 is a diagram of the pool showing both grid plates. The pool, 13 ft. long, 7 ft. wide, and 18 ft. deep, has poured concrete walls ranging in thickness from 6 to 8 ft at the basement (core) level and 6 ft at the first floor level. The water level is about 13 ft above the top of the LTR core. At one end of the pool is a storage pit, 2 1/2 ft wide by 5 1/2 ft deep below the pool's bottom level. The pit contains racks for fuel element storage, and a concrete shield cover to permit drainage of the pool during maintenance and repair operations.

There are three beam ports: two 3-in. ports which do not extend into the pool, and one 8-in. port which had a tube extending to the edge of the LPR grid plate. This extension has been removed for the reference experiments, thus no beam ports or tubes extend to within 24 in. of the LTR or LPR grid plates. The beam ports will be plugged with concrete blocks to provide biological shielding.

A major penetration in the shield wall, adjacent to the LPR grid plate, accommodates an autoclave presently being used for hot exponential experiments. The autoclave is separated from the pool water by a 1-in.-thick aluminum pressure plate, boral plate, four inches of lead bricks, and two shield tanks containing water.

Additional sand shielding is used in the space between the autoclave and the hole in the shield wall, and a thick layer of concrete blocks has been erected around the autoclave for shadow shielding. This arrangement — necessary for LPR operation at 200 kw — provides more than

FIG. 3: SECTION THROUGH LPR POOL



enough biological shielding for the LTR critical experiments which will be conducted at a much lower power level and at a greater distance from the autoclave. Additional shielding is provided by the autoclave facility which will not be used during these experiments; interlocks require the autoclave to be filled with water during LTR operation, even though this shielding is not required for normal operations.

The grid plate will be installed about 6 in. from the LPR grid plate, as shown in Figure 3, and rigidly mounted on the pool floor in a steel frame. The 5-in.-thick aluminum grid plate — about 30 by 35 in. — has a matrix of drilled holes which hold the components of the core in the arrangement shown in Figure 2. The grid plate is similar to the LPR grid plate, having 2-7/16-in. holes for the base of the fuel elements, 1-in.-diameter holes for the base of the beryllium and aluminum elements and the shim blades, and 7/8-in.-diameter cooling holes between the fuel element locations.

The bridge over the pool — which supports the rod drives and other auxiliary equipment — will be repositioned to provide access to the new grid location.

B. FUEL ELEMENTS AND SHIMS

The MTR-type fuel elements in the LPR core will be used in these experiments. Of the 24 elements available, eighteen are full elements with 10 fuel plates. The sandwich-type plates contain a 0.020-in.-thick core of U-Al alloy with 30 w/o highly enriched uranium (greater than 90% U-235). The sides and edges of the plates are clad with 0.015 in. aluminum. Each plate contains about 19 gm U-235, 190 gm U-235 per full fuel element. The curved-plate fuel elements, which fit in a rectangle 2.996 in. by 3.222 in., have an active length of 23-1/2 in. The water gaps between plates are 0.276 in.

In addition to the 18 full elements, two fuel elements with five fuel plates and five aluminum plates may be used for shimming. Four other fuel elements have only six outer fuel plates each, the inner region being used for the control rods. Three of these elements contain safety rods, the fourth contains the regulating rod. Each of the four control rods contains 115 gm U-235.

To cover the range of reactivities needed to perform the experimental program, the effective loading of the fuel elements can be changed by placing poison shims or fuel shims in the water gaps between the fuel plates of each fuel element except the rod containing fuel elements. The poison shims are 0.050-in.-thick stainless steel plates, about 2.25 in. wide by 24 in. long (active). The fuel shims are lucite plates, approximately 1/8 in. thick, 2.25 in. wide, and 24 in. long (active). U-Al foils on loan from the LMFRE critical experiment will be cemented to one side of each fuel shim. These foils, 0.010 in. thick, 1.6 in. wide, and 12 in. long, contain 18 w/o highly enriched uranium (greater than 90%) in aluminum alloy. The U-235 content per foil is 1.6 gm. Ten foils will be cemented to each lucite strip and covered with a layer of waterproof plastic tape.

The reactivity worth of these shims is discussed in Appendix E. None of the assemblies should require the addition of more than three or four poison shims or five fuel shims per fuel element. The limitation and the method of handling shims is discussed in Chapter V. The shims will never be mixed in any one assembly; i.e., the assembly will contain either poison or fuel shims — not a combination.

Since the Operating Procedures stipulate that shims shall not be added to or subtracted from the fuel elements in the core, mechanical interlocks will be attached to each element. These interlocks are right-angle brackets, hinged along the top edge of each fuel element. When the fuel element is in the core, it is adjacent to a beryllium element or a shim rod, either of which extend several inches above the top of the fuel element; however, the fuel element stands alone when removed from the core. In the core position, the hinged bracket covers part of the top of the fuel element by the presence of the adjacent element, but, when removed from the core the bracket can be swung back to expose the top of the fuel element and permit access to the shims. In the core position the bracket does not materially impede the flow of water through the fuel element, but it does prevent the removal of a poison or fuel shim.

C. BERYLLIUM AND ALUMINUM ELEMENTS

High purity, solid beryllium blocks (2.875 ± 0.002 in. sq) will be borrowed from Oak Ridge National Laboratory in 1/4-, 1-, and 4-in. lengths. Each block is pierced by a No. 8 hole through the center so that beryllium elements (about 24 in. active length) can be assembled by stacking the blocks and holding them together rigidly with a 3/16 in. aluminum rod. The rod will be screwed into an aluminum base plate and locked with lock nuts. The base plates, similar to those used on the fuel elements, have a nominal 1-in.-diameter pin which fits into the grid plate. An eyebolt is located at the top of each element to facilitate remote handling. The aluminum elements to be inserted in the four test hole locations in most of the assemblies are solid pieces, approximately 3 in. sq by the same active length as the beryllium, fastened rigidly to aluminum base plates similar to those used for the beryllium elements. A special feature of these base plates — 1/8-in.-thick aluminum webs that extend into the area under the adjacent fuel elements — allow the fuel elements to rest on an integral part of the aluminum elements. This prevents the accidental removal of the aluminum elements before the fuel is removed. An eyebolt will be fastened to the top of the elements to facilitate remote handling.

D. SHIM RODS

One plate and four cruciform shim rods are used in each assembly. The blades are nominally 6 in. from tip to tip (3 in. for central plate) and 0.250 in. thick by 25 in. active length. Three types of rods will be used to simulate the "rod out", "rod in", and "rod partially inserted" conditions. One solid-aluminum set will mockup the "rod out" condition. To mockup actual LTR shim rods, a second set — a sandwich of 0.020-in. cadmium sheet surrounded by aluminum plates — will give a total thickness of 0.250 in. The third set, consisting of cadmium and stainless steel sections, will mockup a partially inserted blade.

The composite and cadmium shim rods will be fabricated from four aluminum plates bent at right angles and cemented together (the cadmium between) with high-strength epoxy resin or double face tape. The technique for fabricating strong and durable rods of this type has been successfully used in the laboratory for several years. The blades will be welded to an aluminum base plate with a 1-in.-diameter pin to fit into the grid plate, and a 1/8-in.-thick web extending under the adjacent fuel elements to prevent accidental removal of shim rods before fuel element removal, and insure that shim rods are inserted into the grid plate before the fuel elements are loaded. An eyebolt will be fastened to a small aluminum plate atop each shim rod to permit remote handling.

E. TEST HOLE INSERTS

Various test hole inserts will be fabricated to simulate test loops which may be used in the reactor. These inserts, approximately 6 in. sq, fit in the space created by removing the four solid aluminum elements. The inserts are clad in aluminum, mounted on an aluminum base place with 1-in.-diameter pins to fit in the grid plate holes, and 1/8-in.-thick webs which extend under the adjacent fuel elements to prevent accidental test insert removal before the fuel elements are removed.

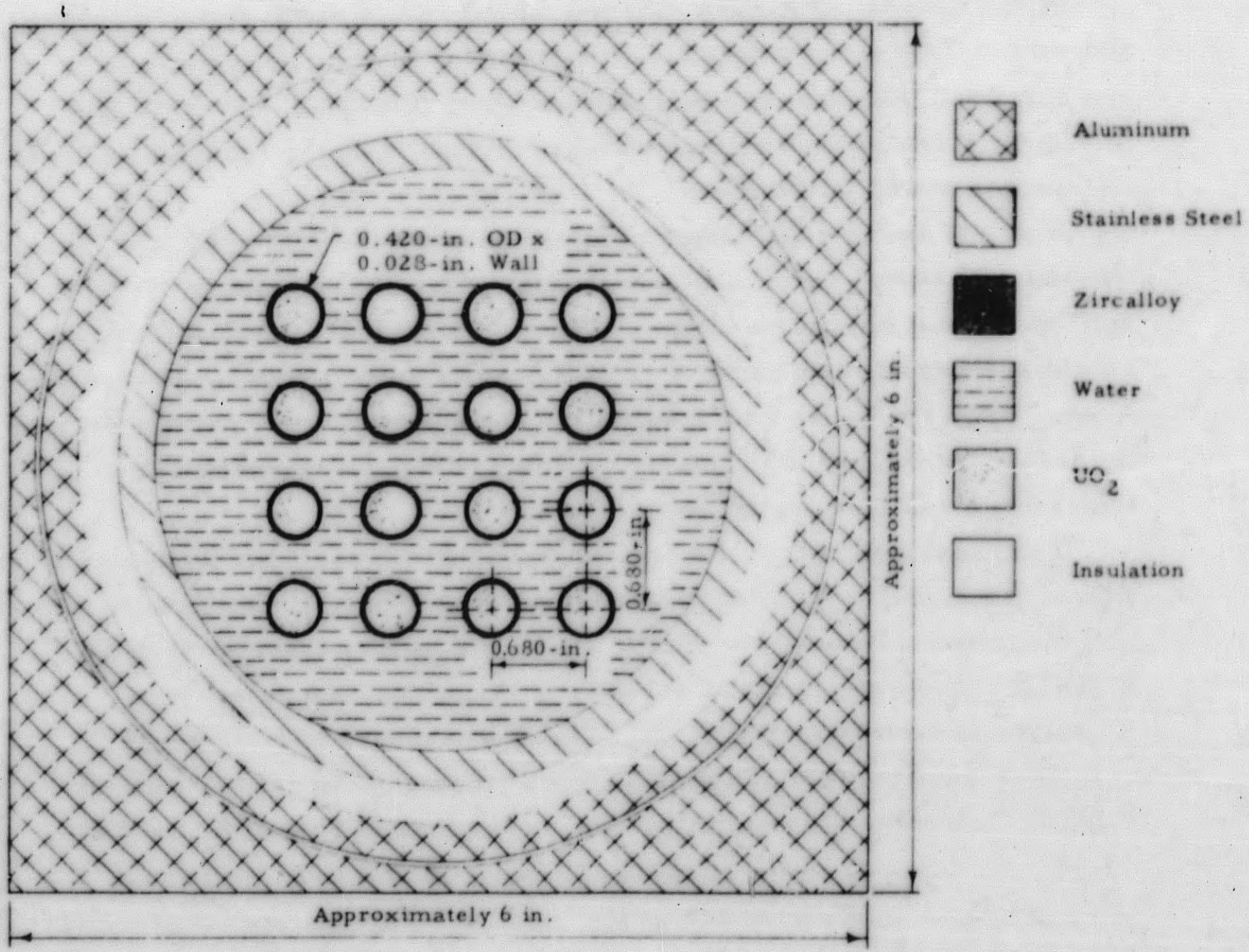
Figure 4 shows a typical test hole insert, consisting of sixteen 0.420-in. OD by 0.028-in. wall Zircalloy tubes filled with UO_2 of 7.0% enrichment to a bulk density of 9.42 g/cm^3. The tubes, on a square pitch of 0.680 in., are mounted rigidly in a water-filled stainless steel pipe, 4.50-in. OD by 0.237-in. wall. This assembly is fixed in a 5.047-in.-diameter hole milled in a 6-in.-sq aluminum block. The 0.273 in. annulus will be filled with Styrofoam or equvilent material to simulate insulation. Other test inserts may contain an ETR-type fuel element (unirradiated) rather than the lattice shown in Figure 4, or dummy elements with no uranimum. No simulated gas loops with large voids will be used.

The reactivities associated with these test loops are discussed in Appendix E, and limitations on allowable types of test inserts are stated in Chapter V.

F. CONTROL RODS AND DRIVES

Three safety rods and one regulating rod will be used for operation and safety. The rods, oval in cross-section, are approximately 2 1/4-in. by 0.89-in. over all and move vertically in the central region of the four speical 6-plate fuel elements. The active length is about 26 inches. The safety rods are aluminum cans, electroplated on the inside with a 0.020 to 0.050-in.-cadmium layer and filled with tamped boron carbide powder. The regulating rod is a hollow can with a composite wall of 0.049-in.-aluminum, 0.065-in.-304 stainless steel, and 0.065-in.

FIG. 4: CROSS SECTION OF TYPICAL TEST LOOP INSERT



aluminum on the inside. The can has holes in the bottom so it is always filled with water.

In the present LPR core, the regulating rod is worth 50 to 70 cents, and each safety rod is worth 4 to 6 percent in reactivity, depending on rod position. Based on these measurements, the safety rod locations in Figure 2 should have enough worth to comply with the rod worth limitations stated in Section V.

The rod drives, duplicates of units operated for years at the Bulk Shielding Facility, have been operated for more than a year in the LPR facility without failure on scram. The safety rods are coupled through an electromagnet to an acme-threaded nut and shaft driven by an electric motor. The regulating rod drive is the same except for the electromagnet. On scram signal or power failure, current to the electromagnets is interrupted and the rods fall by gravity.

The measured scram time in water is 500 milliseconds from initiation of the signal to full rod insertion, and about 350 milliseconds for half insertion. The delay time for actual rod motion is about 30 milliseconds. The maximum rod speed is 3.74-ipm for the safety rods, 37.4-ipm for the regulating rods. This corresponds to a maximum reactivity addition rate of about $0.00020 \Delta k$/sec for the safety rod and $0.00025 \Delta k$/sec for the regulating rod, in their most sensitive positions.

The rod movement is not ganged, each rod is raised separately according to a selector switch at the console. The rod control switches are spring loaded to return to the "off" position. Rod positions are displayed at the console by separate meters and digital indicators accurate to ± 0.01-in. An automatic control system is available but will not be used during these experiments.

G. NEUTRON SOURCE

The startup neutron source will be encapsulated and attached to a length of cable so it can be handled from the bridge above the core. The LPR source of Po-Be contains 0.4 curies of Po-210 (November 2, 1959) with an emission of about 1.0×10^6 n/sec. If this source strength is insufficient, a 10 curie source will be used. When not in use, the source will be lowered into an aluminum tube in the fuel element storage pit.

H. REACTOR INSTRUMENTATION

The LTR critical experiment will be operated and controlled by several neutron detectors located, in aluminum capsules, around the edge of the grid plate. A schematic diagram of the instrumentation is shown in Figure 5.

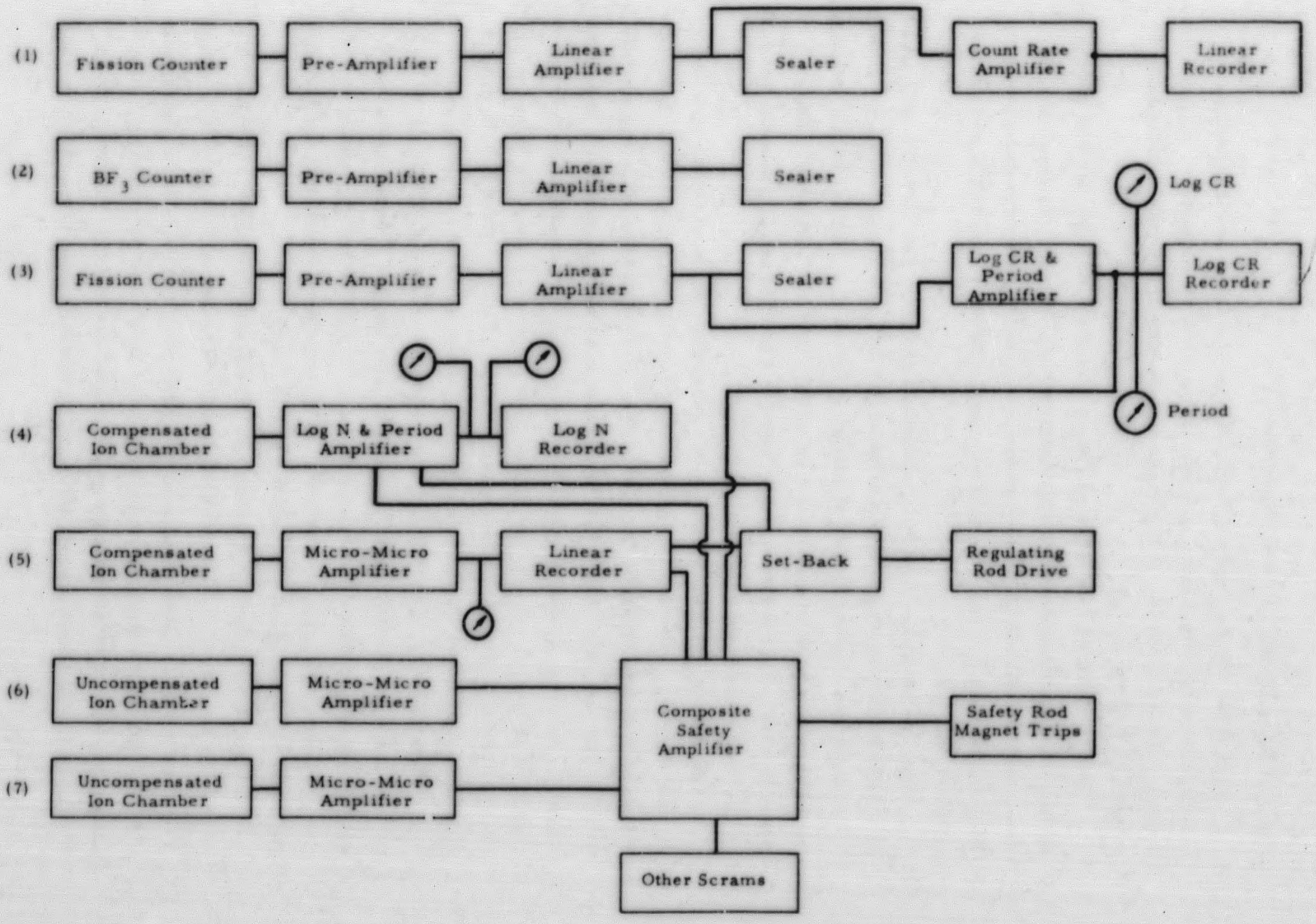
The first three channels shown in Figure 5 will be used to obtain multiplication data when new assemblies are being built to criticality. They employ standard fission chambers and BF_3-filled pulse counters. Additional data for startup will be supplied by the gamma insensitive channel (5).

For operational purposes, channels (1), (3), (4), and (5) will yield two linear and two logarithmic signal indications at the console. Channels (4) and (5) are driven by boron lined, electrically compensated ionization chambers with a sensitivity of about 4×10^{-14} amp/nv. Only gamma insensitive detectors are used in the operating channels because of the fuel elements' moderate activity.

The safety circuits effect both setback on the regulating rod and scram of the safety rods. The setback circuit is actuated by channel (5) at 85% of full scale on the recorder, and by channel (4) at a period of less than 20 sec. On setback, the regulating rod is driven in at normal speed. The trip circuits operate through the composite Safety Amplifier. Channel (4) yields two period trips, one fixed at 3 sec, the other adjustable from 20 to 3 sec, and normally set at 10 sec. Channel (3) also yields an adjustable period trip normally set at 10 sec. Level trips are obtained from channels (5), (6), and (7). Channel (5) trips at 95% of full scale on the recorder, and channels (6) and (7) trip at 100% of full scale on the amplifiers. Because of the expected gamma activity of the fuel elements, the power level at which these latter two level trips will start to operate is estimated at 100 watts.

Other scram signals feeding into the Composite Safety Amplifier are: (1) a low level scram from the log CR output of channel (3), which requires a signal of at least 2 cps. before the rods can be withdrawn at startup; (2) high voltage failure at the ion chamber power supply; (3) loss of water from the autoclave; (4) high radiation in channels (3) or (4) of the LPR Remote Area Monitoring System (see discussion in next

FIG. 5: REACTOR INSTRUMENTATION DIAGRAM



section); and (5) movement of boral shutter. Channel (3) trips at 4 mr/hr and channel (4) at 10 mr/hr. A schedule of the various trips is listed in Table I. Annuciator lights on the console indicate a scram and the source of the scram to the operator.

TABLE I
SCHEDULE OF TRIPS

Channel	Action	Initiation	
		Period	Level
3	scram	-	<2 cps
3	scram	10 sec, adj.	-
4	set-back	20 sec	-
4	scram	3 sec, fixed	-
4	scram	10 sec., adj.	-
5	set-back	-	85% full scale
5	scram	-	95% full scale
6	scram	-	100% above ≈ 100 watts
7	scram	-	100% above ≈ 100 watts
-	scram	-	High voltage failure
-	scram	-	Loss of water in autoclave
RAMS - 3	scram	-	Radiation level above 5 mr/hr
RAMS - 4	scram	-	Radiation level above 5 mr/hr
-	scram	-	Boral shutter movement

I. REMOTE AREA MONITORING SYSTEM

The entire laboratory is served with a multichannel remote area monitoring system consisting of a series of gamma-sensitive ionization chambers. The system is shown in Appendix C along with operating ranges.

Five channels are located in and around the LPR area:

1. At the top of the pool underneath the bridge
2. Office wall adjacent to control console
3. Front of 8-in. beam port
4. Above autoclave
5. Front of autoclave.

The channels yield an audiable and visual alarm at the console. The trip points (adjustable) will be set at approximately 5 mr/hr for all channels except channel (2) which will sound the building evacuation alarms at 100 mr/hr.

V. OPERATING PROCEDURES

A. ADMINISTRATION

Each experiment will be reviewed in detail by these members of B&W's Safeguard Review Committee: D. V. P. Williams, Chairman; M. C. Edlund, W. M. Breazeale, R. E. Wascher, and the Group Supervisor assigned to the LTR critical experiment program.

The Group Supervisor and members of the group will be assigned by the Laboratory Supervisor. The Group Supervisor will be a senior AEC-licensed physicist with extensive experience in critical experiments. Other members of the group include another senior physicist, a junior physicist and two technicians. Group Supervisor's duties include review and approval of each experiment, assignment of personnel for each experiment, control of key to control console, and authorization of all loading operations.

Before the experimental program begins, a complete manual of operating procedures, of which Chapter VI is an abridgement, will be written. All members of the experimental group must be thoroughly familiar with these procedures. Changes in the operating procedures can be authorized only by the Laboratory Chief or the Group Supervisor. Temporary changes will be recorded in the operations log book, but permanent changes require a written amendment to the Operating Procedures.

B. FUEL STORAGE AND HANDLING

The fuel elements will be stored in the shielded storage pit at the far end of the pool (Fig. 3) until the critical experiment program is started. This storage configuration is safe for the maximum fuel element loadings to be used in these experiments, and at least up to 300 gmp U-235 per element, including fuel shims. (See Appendix E.) The fuel shims will be stored in Subassembly Room No. 2, in rigidly mounted compartments on an edge-to-edge spacing of 1 ft, sized to accept no more than 20 fuel shims or 320 gm U-235 per compartment.

Before loading each new assembly, fuel elements will be transferred in sets from the pool storage pit to the LPR grid plate, adjacent to the LTR grid plate, where they will be accessable for the addition of shims. The fuel shims will be transferred to the pool area, as needed, and immediately loaded into the fuel elements. After removal from the fuel elements, the shims will be immediately returned to the subassembly room storage area. Thus all fuel shims will be accounted for either in the fuel elements or the subassembly room.

The number of fuel elements located on the LPR grid plate at any one time will be restricted to eight including the rod-containing fuel elements. These elements will be located on the end of the LPR grid plate farthest from the LTR core, with a minimum distance of 25 in. closest approach. The elements will be spaced on alternate grid locations so they are never closer than 3 in. edge-to-edge spacing. No other configuration will be possible because an aluminum plate will be securely fastened on top of the LPR grid plate with holes in the above stated positions only. The safety of this configuration is discussed in Appendix E.

After the shimming has been completed, the shimmed elements will be returned to the fuel storage pit or inserted in the LTR core, and a new set of elements will be transferred to the LTR grid plate for shimming.

Changes in the shimming of the fuel elements will be made only with the element in the LPR grid plate, never in the LTR core itself. Violation of this rule is impossible because of the mechanical interlock on the fuel element.

When not in the LTR core, the beryllium and aluminum elements and various shim blades will be removed from the pool or stored in a rack in the corner at the bottom of the pool, at least 12 in. from either the LTR or LPR grid plates. The fuel-bearing test hole inserts will be stored in the subassembly room in a safe configuration when not in use. The fuel loading of the test inserts is not expected to exceed 300 gm U-235.

C. CHECK LISTS

Before each experimental run, the following check list will be completed to insure that all equipment is in the proper location and in working order. A more detailed check list will be prepared and entered into the "Operating Procedures Manual" before the experimental program begins.

1. A visual inspection of pool's core, drive mechanisms, and experimental facilities around and inside the pool, will be made. Special attention will be paid to insure that the beam tube plugs and covers are in place, that autoclave and shield tanks are filled with water, that the boral shutter is locked in the closed position, and that a portable survey meter is on hand and operational. A check will be made to note the physical location of any components in the pool that could affect the reactivity of the core, although none are anticipated since the LTR grid plate is at least 24 in. from the pool walls. The general core configuration in relation to the particular experiment will be noted.

2. The location of all fuel elements and shims will be determined, i.e., core, storage pit, LPR grid plate, and inventories will be checked

3. The console instrumentation will be turned on and allowed to warm up at least 15 min. Scram and setback will be checked with gamma or neutron sources as required for the particular trip. Period checks will be made by moving a source rapidly toward the detector. All scram checks will be made with safety rod magnets energized and rods raised approximately 2 in. Setback checks will be made with the regulating rod raised enough to give positive indication of rundown on setback signal. The source will be inserted for channel level readings.

These items will be checked and recorded:

a. Jordan Monitor System (RAMS)
    Recorders (ink & paper)
    Alarm and scram settings (channels 2, 3, 4)

b. Channel 1
    Recorder
    Zero adjust

c. Channel 2
    - Pulse height setting
    - Gain setting

d. Channel 3
    - Recorder
    - 60 cycle check
    - Period check, setting and scram
    - Source level, cps

e. Channel 4
    - Recorder
    - Period check, setting, setback, scram

f. Channel 5
    - Recorder
    - Zero adjust
    - Source level
    - Setback and scram check

g. Channels 6 and 7
    - Zero adjust
    - Source level
    - Meter scram check

4. The following items will also be checked and recorded.

    a. Channel 3 fission chamber position, pool
    b. Rod positions (safety up, regulating down) pool and console
    c. Startup approved
    d. Startup announced on building intercom
    e. Close bay doors and turn warning light on
    f. Cock at least two safety rods and make sure regulating rod is fully in
    g. Insert neutron source and make sure recorders are on scale and respond

D. INITIAL APPROACH TO CRITICAL

The initial approach to criticality will follow the procedure described below. This procedure will be followed whenever an appreciable change in the reactor loading is made, i.e., a change greater than about 10% in reactivity.

Starting with the empty LTR grid plate, the five shim rod locations will be filled with the aluminum shim rods. Next, the four test hole positions will be filled with aluminum elements. Then the fuel elements containing the safety and regulating rods will be inserted and their rod drives mounted in place. At this point the check-list previously discussed will be completed, two safety rods cocked, the third safety rod and the regulating rod fully inserted, and the neutron source introduced. The beryllium elements will then be added, one at a time, until all 38 beryllium spaces have been filled.

The fuel elements will then be loaded incrementally, the first loading being less than 1/3 the predicted number required for criticality, including the rod containing elements. Since the full elements will have been shimmed so that the predicted number of fuel elements for criticality will always be 20, the first loading will be three elements. The counting rate will be taken under three conditions: one safety and one regulating rod in; one regulating rod in; and no rods in. The reciprocal multiplication will be plotted for at least three channels, and the predicted critical mass will be determined for these three conditions from extrapolation of the multiplication data predicting the lowest mass. The size of the next loading will be less than 50% of the remaining extrapolated critical mass.

This procedure will be followed for subsequent loadings until the multiplication curve is well defined. The loadings will be made in order of decreasing importance, i.e., building from the center of the core outward. The final loading increment will be such that the reactor will be critical on the regulating rod, with the three safety rods fully withdrawn. In general, at least five incremental loadings will be made. Because of the moderate activity of the fuel elements, there may be a substantial increase in neutron activity from the (γ,n) reaction on beryllium, which would tend to increase the apparent multiplication as fuel elements are added. This affect may be checked periodically by removing the neutron source and remeasuring the background, although the contribution of these neutrons will make the predicted mass conservative, as determined by extrapolation of the reciprocal multiplication data.

As the neutron level reaches the operating range, the source will be removed and the regulating rod adjusted to maintain the flux level. To verify criticality the regulating rod will be withdrawn slightly to put the reactor on a long positive period.

If the critical point is reached before the 20 fuel elements are added, or if the assembly does not go critical after the 20 are added, the fuel elements must be re-shimmed. This will be accomplished by removing the fuel elements from the core to the LPR grid, changing the shimming uniformly, and repeating the loading cycle. If the full configuration after the second shimming is sub-critical or critical with fewer than 20 elements, a third shimming may be necessary. Depending on the particular experiment and the importance of a uniform loading, these final shimmings may be done on several fuel elements only rather than on the full complement, although all shimming will still be done with the fuel element in the LPR grid.

On completion of each experiment the reactor will be shut down and all control rods fully inserted. A shutdown check list will be prepared and entered into the Operating Procedures Manual.

E. CONTROL ROD CALIBRATION

For safe operation the approximate worth of the safety and regulating rods must be known. During the approach to critical, some information on these rod worths will be obtained from the extrapolated reciprocal multiplication data. As an added check that the safety rods have sufficient hold-down power, rod worth will be measured by the rod-drop technique. Since this method may be subject to error because of the (γ,n) reaction in beryllium, the data must be compared with the multiplication data. If a consideration of these two methods raises a serious question about the approximate safety rod worth, it will be checked further by inserting a safety rod, disconnecting the rod from its drive, and reshimming the entire assembly to determine the new critical mass. If the safety rod worth is less than the equivalent of about 9% in reactivity for the three safety rods, they will be relocated to more sensitive regions.

The regulating rod will be calibrated to ascertain that it is worth less than about 0.7% in reactivity and to aid in the measurement of small reactivity changes. The calibration will be done using the rod-bump-period method, by removing the rod in small increments, measuring periods and leveling with one of the safety rods. This technique also implies accurate knowledge of the in-hour relation which is in question for beryllium reactors. For comparison, the worth will be measured by rod drops and, if necessary, by reshimming the reactor. If the regulating rod is in too sensitive a position, it will be relocated to correspond to the operating limitations set forth.

F. CRITICAL LOADING OF OTHER ASSEMBLIES

New assemblies will be built following the approach outlined in the "Initial Approach to Critical". The previous assembly will first be unloaded in the reverse order of the loading, or according to one of the procedures listed below.

1. Remove fuel elements in sets (of eight or less) to the LPR grid where the shimming is changed consistent with the new assembly, then transfer the elements to the pool storage pit. Continue until all non-rod-containing fuel elements are removed from the core.
2. Remove as many beryllium elements as needed and transfer them to beryllium storage area.
3. Remove fuel elements containing control rods and transfer them to the LPR grid.
4. Remove aluminum elements from the test hole and transfer them to the beryllium storage area, or remove them from the pool.
5. Remove aluminum shim blades and transfer them to the beryllium storage area, or remove them from pool.

The new assembly will then be constructed in the sequence described previously.

The loading and unloading sequences will be rigidly enforced to prevent the accidental removal or addition of any core component that might make the reactor critical. The calculations of the reactivity worth of each component are discussed in Appendix E. The removal of any fuel element and replacement of the void by water will decrease reactivity, so the fuel elements are always added last and removed first. The removal of any beryllium element and replacement of the void by water will decrease reactivity, so the beryllium will always be added before the fuel. Removal of the shim rods always increases reactivity, so they are added before the fuel and removed after the fuel. This may not be true of the test hole inserts depending on their composition; however, to insure safe and consistent loading practices, the test inserts will always be added before and removed after the fuel.

To insure that the shim rods and test inserts are always added before the fuel elements and removed after the full elements, they are equipped with webs which extend underneath the adjacent fuel elements. This presents a mechanical back up to a possible human error in the loading procedures.

G. OPERATING LIMITATIONS

The following operating limitations will be adhered to.

1. Power Level

The operating power level will be kept as low as possible, consistent with the requirements of the experiment. Normally, this will be less than one watt for reactivity measurements. For short times, the power level will be increased to the 100 watt range for foil irradiations. For a very few special irradiations, it may be necessary to go to a power level of approximately 1000 watts.

2. Reactivity Addition Rat

The maximum reactiv[illegible] addition rate will be restricted to 0.00025 Δk/sec, the maximum reactivity addition rate of any control rod in its most sensitive position.

3. Maximum Excess Reactivity

The available excess reactivity will generally be less than the worth of the regulating rod, and never greater than 1.5% in reactivity. The maximum "locked-in" excess reactivity, which would be held down by the fixed shim blades and test inserts, is expected to be less than 15% in reactivity.

4. Control Rods

At least two safety rods will be cocked and available for shutdown at all times. Normally, and especially with new assemblies, the number will be three, but in some special power-mapping experiments, one of the three safety rods may have to be replaced with a normal fuel element. Each safety rod will be worth more than the regulating rod, and at least 9% shut down reactivity will be available in the three safety rods. The regulating rod worth will normally be less than 75 cents, and in no case greater than one dollar.

5. Maximum Number of Fuel Element Shims

The maximum number of poison shims per element will be that required to hold down the basic assembly (Fig. 2). It is estimated that two to three shims per element may be required. The limit on the number of full shims per fuel element will be 80 gm per fuel element extra or 10% in reactivity, whichever is greater. (See Appendix E.)

6. Test Inserts

The range of reactivities possible from various test inserts will be from -3% to + 3% (see Appendix E). No experiments will be undertaken in which there would be floodable voids large enough to yield a reactivity increase of 1% or more if the void became filled with water. In all cases this potentially floodable void would be included in the operating margin of 1.5% excess reactivity requested, so that if 1% in reactivity were potentially available the reactor would be shimmed so that at most only 0.5% would be available in movable control rods.

The potential hazard of floodable voids is well recognized; therefore, the preceeding operating limitation was specified and will be strictly enforced. If any experiments are undertaken with floodable voids, they will be done late in the experimental program, after experience has been gained in the operation and characteristics of the reactor. If there is any question about the reactivity worth of a flooded void for a particular experiment, that experiment will be preceeded by carefully controlled experiments on the importance of small voids. The resulting data can then be extrapolated to the actual test inserts.

VI. NORMAL OPERATING HAZARDS AND HEALTH PHYSICS PROCEDURES

A. DIRECT RADIATION

The LTR critical assembly will be covered at all times with approximately 13 ft of water. (The concrete shielding walls at the elevation of the core are from 6 to 8 ft thick.) Since the LTR core will lie in approximately the same position as the LPR, the extensive health physics surveys made during LPR operations may be used. Scaling down the LPR data at 10 kw operation to the maximum LTR power of 1000 watts, there will be no position above and around the pool where the combined gamma and neutron doses are greater than one-tenth of tolerance. This includes the N-16 activity at the surface of the pool.

B. ACTIVATION OF MATERIALS

The LPR fuel elements to be used in these experiments are moderately radioactive, reading approximately 1 r/hr per element at 15 ft in air. From past operating records, the burnup of the fuel elements is conservatively estimated to be less than 1 MW-day. These elements, not to be removed from the pool, will be stored in the shielded storage pit if the pool is drained. If a fuel element should be inadvertently raised toward the surface of the pool, the dose rate 4 ft from the surface would be in the range of 10 to 100 mr/hr. Compared with the residual activity of the fuel elements, the activity added by the LTR critical experiments is negligible.

The other components of the assembly — beryllium and aluminum elements, fuel and poison shims, test inserts, and shim blades — may be slightly activated in the higher power runs but can be left in the pool, if necessary, to decay to acceptable levels. In any event, any item removed from the pool is continuously monitored by a portable beta-gamma survey meter and governed by the laboratory health physics regulations.

C. HEALTH PHYSICS

1. Administration

The Health Physics Supervisor is responsible directly to the Atomic Energy Division management. All health physics procedures have been established to meet the requirements of Part 20, Title 10, "US Code of Federal Regulations", as well as applicable state and local regulations. A set of operating rules and procedures for the laboratory Health Physics Program has been published for use by the personnel concerned. These rules are enforced by the Health Physics Supervisor and Group Supervisors, without exception. The instruments needed to carry out the laboratory Health Physics Program are described in Appendix C.

2. Personnel Monitoring

Contamination on clothing or person will be monitored by the individual when changing shifts or as needed. Monitoring instruments are suitably located. General exposure will be monitored by film badges, film rings, and pocket dosimeters. The dosimeters are read at the end of each day, and the films are processed and stored by an outside contractor. The Health Physics Supervisor maintains all records and investigates all cases of unusual exposures.

3. Building Survey

Floors and working surfaces will be surveyed periodically by health physics personnel who will report the degree of area contamination. Air within the building will be sampled periodically with a high volume air sampler; in certain areas air will be sampled continuously using a portable air sampler, detector, and recorder.

General beta-gamma radiation level monitors operate continuously in all areas of the laboratory. Various channels operate recorders, sound alarms, and scram reactors, as shown in Appendix C. There is also a continuously operating air monitor above the LPR pool.

At the beginning of and during the LTR critical experiment program, health physics personnel will monitor the radiation level of fast, intermediate, and thermal neutrons, and also beta and gamma radiation throughout the working area. Should an experiment require special health physics rules and precautions, these will be published.

4. Site Survey

Surveys for contamination and general monitoring on the Company site and the surrounding area will be scheduled by the Health Physics Supervisor. Unusual weather conditions requiring changes or special operating precautions will be published by the Health Physics Supervisor as needed.

D. DISPOSAL OF RADIOACTIVE MATERIALS

Since the LTR critical experiments will be done at low power no high level activities will be induced. At the conclusion of the experiments the fuel elements will be returned to the LPR for future use. The beryllium elements are not expected to be appreciably radioactive, but they will be surveyed and returned to Oak Ridge National Laboratory in accordance with AEC regulations. The fuel shims will be returned to the AEC for ultimate disposal when the LMFRE critical experiment equipment is dismantled.

Only solid materials will be used in these experiments. Solid and liquid wastes will be handled as described in Section III. A of the "Application for License for the Fuel Element Fabrication Plant and Supplement".[1] Should the pool water become contaminated as a result of a fuel element failure, the procedure outlined in Chapter VII will be followed.

VII. UNUSUAL CONDITIONS

A. EMERGENCY PROCEDURES

The emergency procedures in use at the Critical Experiment Laboratory are described in Appendix D.

B. FIRE

The materials as well as the laboratory are fire resistant, and will contain all source and fissionable materials in the event of fire. If a fire occurs during operation, the experiment will be shut down until safe working conditions are restored.

C. EARTHQUAKE

The history of earthquakes in the Lynchburg area is summarized in Exhibit 3 of the "Critical Experiment Hazards Evaluations".[8] It is very unlikely that an earthquake in this area could bring about conditions that could not be handled by the present emergency plan.

D. STORM

The building will withstand any weather conditions likely to be encountered at this site.

E. FLOOD

The laboratory is 139 ft above the level of the James River and 20 ft above the run-off level. This provides adequate protection against damage by any conceivable flood. (These measurements were taken in the autumn of 1955.)

F. STRIKES, RIOTS, AND OTHER VIOLENCE

In the event of strike, riot, or other violence that might interfere with normal operation, the operators will lock reactor controls off and lock all security doors leading to source or fissionable materials.

This precaution, together with the plant security force, will assure protection of these materials.

G. SABOTAGE

Access by a saboteur is restricted in several ways. The laboratory is surrounded by a chain-link fence, located at least 50 ft from the building. The grounds are patrolled by the plant security police. All laboratory employees are required to have AEC clearance, and visitors are registered and escorted while in the building. When not in use the operating console is locked by a key-switch; after working hours, the laboratory, bays, and subassembly rooms are locked and inspected periodically by the plant security police.

If a saboteur were to gain unrestricted access to the LPR area, the most serious damage would result from his rearranging the core in a supercritical configuration. To do this, the saboteur would have to unload the core because the mechanical interlocks require the removal of fuel elements before the other components can be removed. The reloaded configuration might or might not be supercritical with the safety rods in, depending on the reactivity worth of the shim rods removed. However, this loading and unloading sequence would be time-consuming and would require experience and skill. Furthermore, fuel elements have to be loaded one at a time, so it would be virtually impossible for the saboteur to continue to add more fuel elements during the self-limiting BORAX-type excursion that would occur.

The saboteur might try to damage the reactor by raising all control rods and producing a power excursion, but success would be very unlikely for these reasons:

1. The console would be locked.
2. The scram circuits would have to be rewired, requiring considerable time and skill if it were to be done without scramming the reactor in the process.
3. Even if all rods were removed, the operating rules limit the amount of excess reactivity to the worth of the regulating rod, in no case more than 1.5%. This is less than that needed to produce a BORAX-type excursion that would destroy the reactor.

H. POWER FAILURE

All controls and equipment are designed so that a power failure will scram the reactor safely. When power is restored, the complete startup procedure will be followed.

I. LOSS OF POOL WATER

The LPR pool, when filled to normal operating level, contains 11,700 gal of water. The pool fill pump has a 10 gpm capacity, requiring 19.5 hr to fill the pool. About 1.2 hr are required to drain the pool when the drain is fully open. The drain valve will remain locked shut when fuel elements are in the pool. A complete opening of the 8-in. beam tube would drain the pool in 4 min; however, this is considered impossible since the tube is sealed on the water side by a bolted pressure plate and metal O-ring, and on the experimental area side by a bolted access door. The 3-in. beam ports are sealed by welded aluminum pressure plates on the water side and screw caps on the experimental side. Loss of water through the autoclave opening is prevented by a 1-in.-thick aluminum pressure plate, reinforced by vertical and horizontal metal ribs.

Only one leak — in a pool wall light — has been discovered since LPR operation began. The four pool lights are connected to 3/4-in. drain lines and to a 3/4-in. electrical conduit. The two lights on each side of the pool are connected to the same drain and power conduit. The lower light on either side is at the top of the LPR Core, 5 ft above the pool floor. If a leak in any light were large enough to fill the drain line, 7.4 hr would be required to reduce the water level to 5 ft. That such a leak could develop and remain unnoticed for any length of time is very unlikely.

If all of the water were drained from the pool, the direct radiation level at the top of the pool would be under 20 r/hr, and much lower in the area of the console and other spaces normally occupied. Long before this, however, the audible alarm of the Remote Area Monitoring System would announce the loss of water. The moderate fuel element activity and the lower-power operation planned for the reference experiments would preclude the possibility of fuel element meltdown.

J. FUEL ELEMENT RUPTURE

Either abrasion or a pin hole defect could cause a fuel element plate to rupture, and a defect in the waterproof tape covering a fuel shim could cause it to corrode. However, such occurences are unlikely in that these fuel elements have been operated for over a year without failure. In addition, the U-Al fuel shims are fairly resistant to oxidation, and the waterproof tape covering will be inspected periodically.

If a rupture should occur, a small quantity of fission products would escape into the water which would become slightly contaminated. This would not constitute a major hazard to personnel, and no significant amount of gaseous activity would escape to the air. The foregoing statement is conservative because past history on fission breaks at the MTR indicates that the release of activity is very small. In addition, the radioactivity of the fuel will be moderate, and the activity of the fuel shims will be negligible. Contaminated pool water will be passed through the ion exchange beds in the water purification system, and the contaminated resin will be handled as waste. Any residual activity adsorbed on the concrete walls and other pool components will be cleaned up by approved techniques. A 5000-gal storage tank is available to facilitate water cleanup.

VIII. EVALUATION OF MAXIMUM CREDIBLE ACCIDENT

A. TYPES OF ACCIDENTS

1. Mechanical Failure Alone

The experiment has been designed so that mechanical failures alone cannot cause a serious accident. Loss of water, power failure, and fuel element failure have been discussed in Chapter VII.

There are no unfilled void regions in the core. All core components are held rigidly in their appropriate holes in the grid plate. The accidental expulsion of poisonous shims from the fuel elements is prohibited by their weight and the mechanical guard that prevents their removal while in the core. The expulsion of fuel shims is prevented by the mechanical guard. The construction of the U-Al foils in the fuel shim will prevent them from slipping into a more reactive position. To obtain a 1% increase in reactivity, the five upper U-Al foils in each of 16 fuel shims would have to fall exactly 6 in.

No experiments will be done in which there would be floodable voids large enough to yield a reactivity increase of 1% or more if the void became flooded. Normally, no configurations will be constructed with floodable voids, but in the few experiments where they are necessary, the reactor will be shimmed so that the 1% reactivity potentially available by flooding will be included in the 1.5% maximum available excess reactivity limitation; thus only 0.5% would be available in movable control rods.

2. Human Error With and Without Mechanical Failure

a. Continuous Control Rod Withdrawal

If the operator fails to start up the reactor properly, a power over-shoot could occur. The startup accident analyzed in Appendix F, shows that under the worst conditions and conservative assumptions, the maximum excursion power would be less than 3000 watts. Over-shoot is limited because of the slow rate of control rod withdrawal and the low operating power and trip points. In this excursion neither the LPR fuel plates nor the U-Al fuel shims will reach excessive temperatures. The analysis assumes no inherent shutdown mechanism and assumes that the excursion is terminated by two safety rods. If the safety rods should fail, the self-regulating features of water-moderated reactors would come into play.

b. Removal of Core Components in Violation of Operating Procedures

The operating procedures require this loading and unloading sequence: (1) shim rods, (2) test holes, (3) control rod containing fuel elements, (4) beryllium elements, and (5) fuel elements. Furthermore, mechanical interlocks, described in Chapter IV, are rigidly fastened to the shim rods and test inserts to prevent their being removed before fuel elements are removed. Therefore, the sudden increase in reactivity caused by removing shim rods or test holes from a critical reactor is not considered credible.

c. Removal of Poison Shims or Addition of Fuel Shims

The operating procedure states that fuel element shims may be changed only when the fuel element is outside the core. Furthermore, mechanical interlocks, described in Chapter IV, prevent the addition or removal of shims while the fuel element is in the core. Even if the mechanical interlock could be violated, about six poison shims would have to be removed , or about 10 fuel shims added, to change reactivity by 1% Therefore, the sudden increase in reactivity caused by removing poison shims or adding fuel shims is not considered credible.

3. Conclusions

There are no credible mechanisms by which an excess reactivity of greater than 1% could be added — except for the continuous rod withdrawal and simultaneous failure of the safety rods to scram the reactor. However, since the reactor may have an available excess reactivity of 1.5% under certain circumstances, it is assumed that, by some unknown mechanism, the addition of 1.5% excess reactivity is credible.

B. MAXIMUM CREDIBLE ACCIDENT

Although the excess reactivity available in the reactor will never exceed 1.5%, maximum credible accident is defined as one in which 2% excess reactivity is instantaneously added to the critical reactor. A discussion of the resulting excursion, in terms of data from the SPERT and BORAX experiments, is presented in Appendix G. It may be concluded that a 2% excursion would not destroy the reactor, although some of the fuel plates in the high flux region of the core would be close to the melting point.

To determine the hazard to operating personnel and general public, it is assumed that, coincident with the excursion, reduced heat transfer will occur in some of the coolant channels so that 10% of the fuel plates and 50% of the U-Al foils in the fuel shims melt. The consequences of this accident, fully discussed in Appendix H, are summarized here.

Based on the maximum quantity of U-Al fuel shims expected, the quantity of fission products in the melted fuel will correspond to an excursion of approximately 10 MW-sec. In addition some longer-lived fission products from earlier LPR operation will be contained in the elements. The estimated pool water activity, assuming that all fission products in the molten fuel escape into the water, is given in Appendix H. The activity ranges from 23.7 mc/cm^3, 10 sec after the excursion, to about 0.02 mc/cm^3 10 hr later. The radioactive water does not present a personnel hazard, and it may be disposed of as indicated in Chapter VII. The dose rate from the water activity and residual core is below tolerance through the concrete shielding wall, but ranges from 1.5×10^5 r/hr at 10 sec, to 100 r/hr after 10 hr, directly above the top of the pool. The dose rate in the operating area due to air scattering is difficult to compute, but should be 10^{-3} to 10^{-4} times that at the surface of the pool, providing reasonable time for evacuation.

The only fission product activities escaping to the air are Xe and Kr isotopes. If all the Xe and Kr is released instantly, the concentration in the LPR wing would be 0.087 micro-curies/cm^3 of air, immediately after the excursion; if the iodine isotopes are included the concentration is 0.31 micro-curies/cm^3 of air. If this activity were uniformly dispersed, anyone located in the center of the LPR area would receive a dose 3.8 r/hr from Xe and Kr, and 13.5 r/hr from Xe, Kr, and I. These levels are low enough to permit evacuation before any substantial fraction of a lethal dose is incurred. The values are also very conservative because the volatile fission product activity will be transferred rather slowly from the molten fuel to water, and then to the air. Since iodine will probably dissolve in the water, it will not be released to the air. If the activity is contained in the building, the dose rate 50 ft from the laboratory would be less than 3 mr/hr, thus evacuation of the building would eliminate any direct radiation hazard to personnel.

C. ENVIRONMENTAL HAZARDS

Thus far the activity released in the excursion is assumed to be contained in the building, so no environmental hazard exists to the general public. The calculations in Appendix H give the radiation dose outside the building, assuming that all volatile activity is instantly released from the building and diffused downwind. The maximum integrated dose from the fission product cloud, assuming inversion conditions, is only 89 mr 50 ft from the laboratory. Hazards resulting from fallout and washout were not computed because of the low activity released.

The most serious environmental hazard is the inhalation of I-131. If the iodine is assumed to be released from the pool water, the dose delivered to the thyroid from I-131 inhalation during the passage of the cloud is below the 1 yr tolerance level at all points offsite, and it is above the first week tolerance level only under inversion conditions to a distance of about 7500 ft. On site the thyroid dose is only significant under inversion conditions within about 440 ft of the laboratory. Several moderating circumstances make these dose levels highly conservative:

1. The quantity of I-131 release is soluble in a few liters of water.
2. The volatile activity cannot be released instantly from the building since the ventilating system will seal the building after an excursion.

3. Most of the I-131 is contained in the LPR fuel elements from the previous operation — very little is in the fuel shims. The assumption that 10% of the LPR fuel plates melt is very conservative and, if the accident occurs after the first week of LTR operation, the I-131 activity in the fuel will have been reduced by a factor of at least two.
4. The thyroid is not an indispensable organ in man.

In conclusion, the maximum credible accident would not endanger local residents off the company site, and there would be no serious hazard to employees outside the laboratory building. Laboratory employees in the LPR wing would be exposed to a dangerous condition only if all of the many conservative assumptions should prevail at the same time.

IX. APPENDICES

APPENDIX A
SITE LOCATION, POPULATION DENSITY, AND METEROLOGICAL DATA

1. Site Location

The Critical Experiment Laboratory is located in Campbell County, Virginia, approximately 4.5 air miles east of downtown Lynchburg, the largest center of population in the area. The site and surrounding counties are shown in Figure 6. To the east and north, the direction of prevailing winds, the population is predominantly rural. The largest concentration is the town of Farmville, approximately 40 miles east.

The Laboratory is situated on a 550-acre plateau, bounded on three sides by the James River. The B&W Nuclear Facilities Plant and associated buildings are located about 1200 ft SE of the Laboratory. The tracks of the James River Branch of the Chesapeake and Ohio Railroad follow the river around the site. At their closest approach the tracks pass about 440 ft SW of the Laboratory, with 1.6 miles of track within 0.5 miles or less of the Laboratory. Freight traffic averages one train per hour; there is no passenger service. The Laboratory Guardhouse and security fence are about 50 ft from the Laboratory.

2. Population Density

The 1950 population figues for surrounding counties and the city of Lynchburg (1958 census) are shown in Table II [2]. In the immediate vicinity, the nearest dwelling to the west is 2500 ft away, and the nearest to the east is 5500 ft from the Laboratory. U. S. Highway No. 460 passes within about 2 mi of the Laboratory. Location of population in the immediate vicinity of the Laboratory is shown in Table III.

3. Meterological Data

Pertinent meterological data were extracted from a report by Col. J. M. Morgan, Jr., [2] which has been published as Exhibit H of

TABLE II

POPULATION IN NEARBY COUNTIES

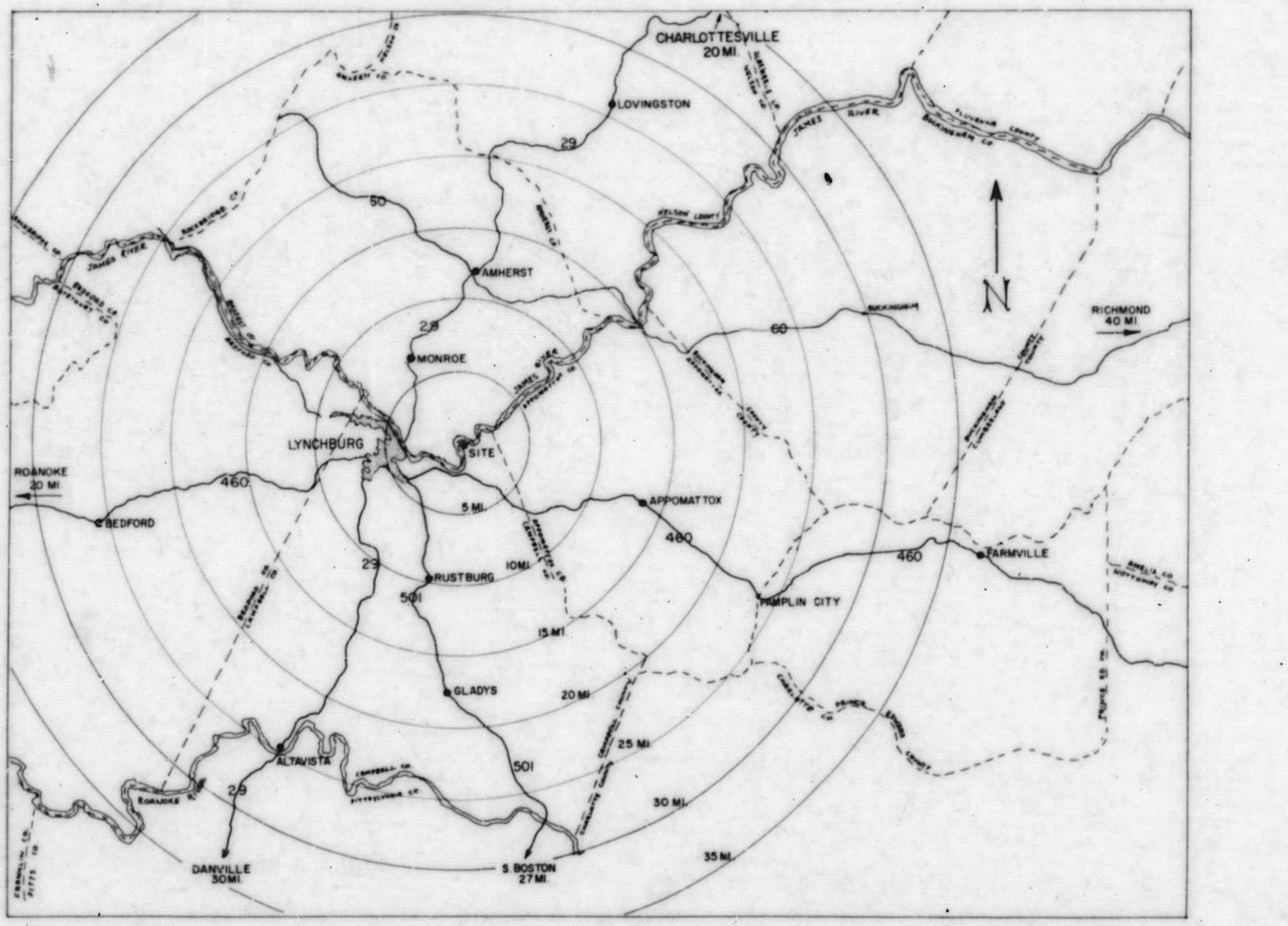
County	Area, sq mi	Population Urban	Population Rural	Total	Rural Population Density, persons/ sq mi
Amherst	476	1038	22294	23332	48
Appomattox	343	1464	7300	8764	21
Buckingham	576	264	11964	12228	21
Campbell	530	51942	24662	76604	46
Charlotte	478	1787	12270	14057	26
Nelson	468	-	14042	14042	30
Prince Edward	357	4375	11023	15398	31
City of Lynchburg	13	53161	0	53161	4089

TABLE III

LOCATION OF POPULATION IN IMMEDIATE VICINITY OF THE CRITICAL EXPERIMENT LABORATORY

Building	Distance from CEL	Population, or Normal Working Force
Critical Experiment Lab	0 - 5 ft	30
Nuclear Facilities Plant	1200 ft	585
Off Site	0 - 0.5 mile	2
	0.5 - 1.0 mile	7
	1.0 - 1.5 mile	15

FIG. 6: MAP OF AREA AROUND THE CRITICAL EXPERIMENT LABORATORY



reference[8]. Data on wind conditions were taken from U. S. Weather Bureau records[10] at the Preston Glenn Airport and from observations at the Laboratory. Wind roses constructed from these data are shown in Figure 7. The U. S. Weather Bureau data are a composite, covering the years 1914 to 1933 at downtown Lynchburg, and 1936 to 1950 at the Preston Glenn Airport, 6.5 miles SSW of downtown Lynchburg. Because the intervening terrain is moderately hilly, these data are not directly applicable to the Laboratory site. Data taken at the site (covering the period July, 1958 to September, 1959 – 692 observations) are shown in Figure 7 for comparison, and confirm the low probability of winds from the west.

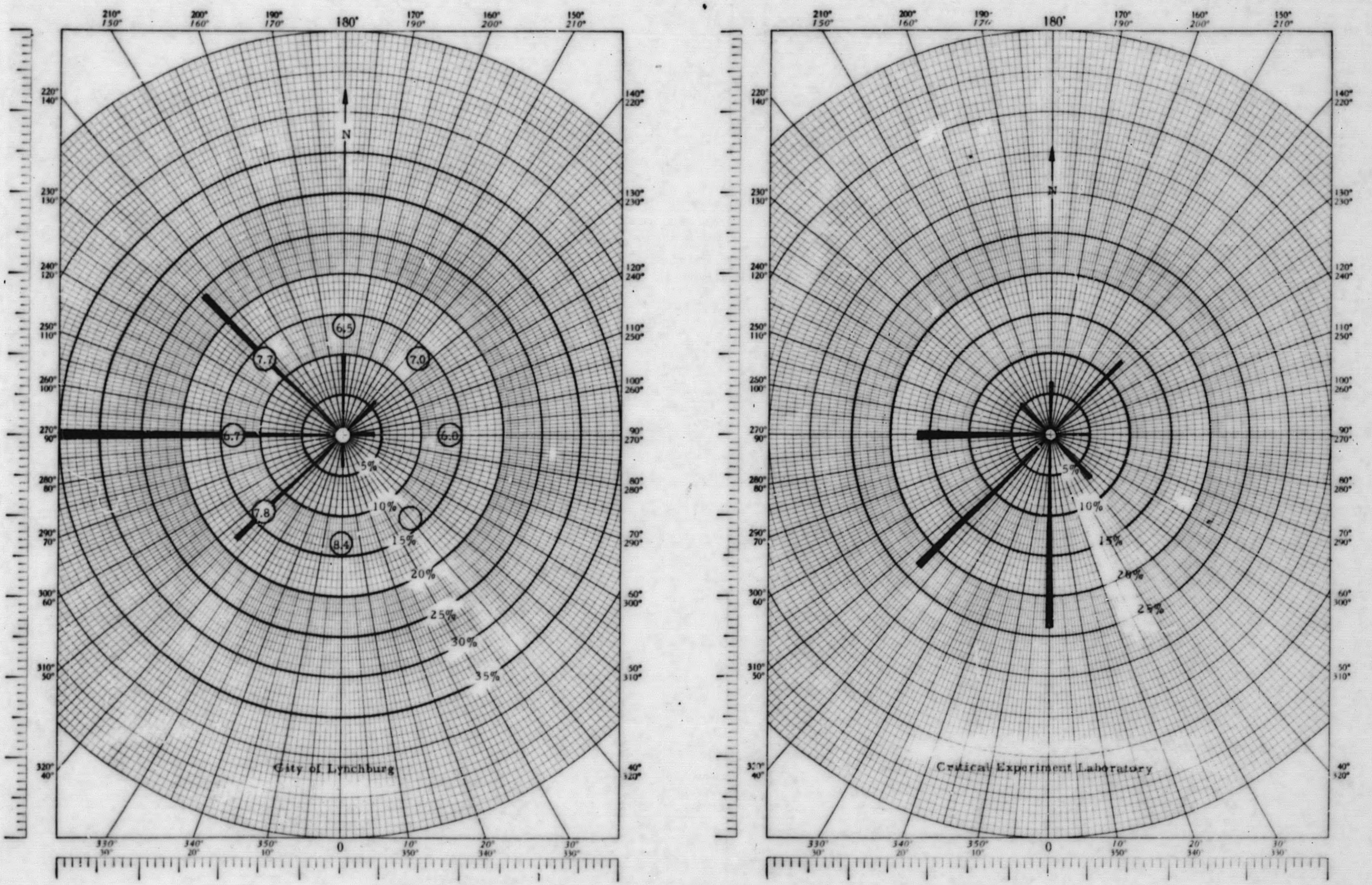
Data on inversion frequency are not available in Lynchburg. The nearest Weather Bureau office that collects lapse rate data is in Greensboro, North Carolina. Therefore reasonable meterlogical parameters have been assumed for both average and inversion conditions using data from the U. S. Weather Bureau.[11]

TABLE IV

ASSUMED METEROLOGICAL PARAMETERS

Parameter	Average Condition	Large Inversion
h, meter	0.	0
n	0.25	0.50
$\bar{u}$, m/sec	3.0	3.0
c^2	0.04	0.005
c	0.20	0.07

FIG. 7: WIND ROSES FOR THE CITY OF LYNCHBURG AND THE CRITICAL EXPERIMENT LABORATORY



NOTE: Length of Bar Represents Probability of Wind Blowing From That Direction. Numbers in Circles Represent Average Wind Velocity in mph.

APPENDIX B
CRITICAL EXPERIMENT LABORATORY BUILDING

Figure 8 is a photograph of the Critical Experiment Laboratory; basement and first floor plans are shown in Figure 9 and 10. The Laboratory has two shielded bays and a reactor wing, plus control rooms, sub-assembly rooms, a counting room, health physics laboratory, chemical laboratory, physics laboratory, electronics shop, machine shop, change rooms, and offices. The building is completely air-conditioned to control temperature and humidity.

The wing housing the Lynchburg Pool Reactor (LPR) is about 40 ft long and 36 ft wide on the ground floor. It is 22 ft high in front and 35 ft high in the rear, where an overhead service crane is installed. A balcony about 10 ft above the ground floor extends to the edge of the reactor pool; part of the balcony area is walled off for office space. The LPR control console is on the balcony.

The wing is constructed of standard structural steel members, cinder blocks, and commercial brick, 8-in. thick. Plan and elevation drawings of this wing are shown in Figures 11 and 12.

The LPR wing is ventilated by three heat pumps, two of which draw air from the outside into the LPR wing. The air is vented through a 2-ft-square port in the wall above the pool surface. The particulate activity above the pool is monitored by a G-M tube, assisted by a pump and micron-size filter. On a high radiation signal, the inlets to the heat pumps and the shutter to the ventilating port are automatically closed, in less than 2 sec. This action is accompanied by visible and audible alarm. Since all doors to the wing are closed, (part of the pre-critical check list), this effectively seals off the area.

FIG. 8: CRITICAL EXPERIMENT LABORATORY – PHOTOGRAPH

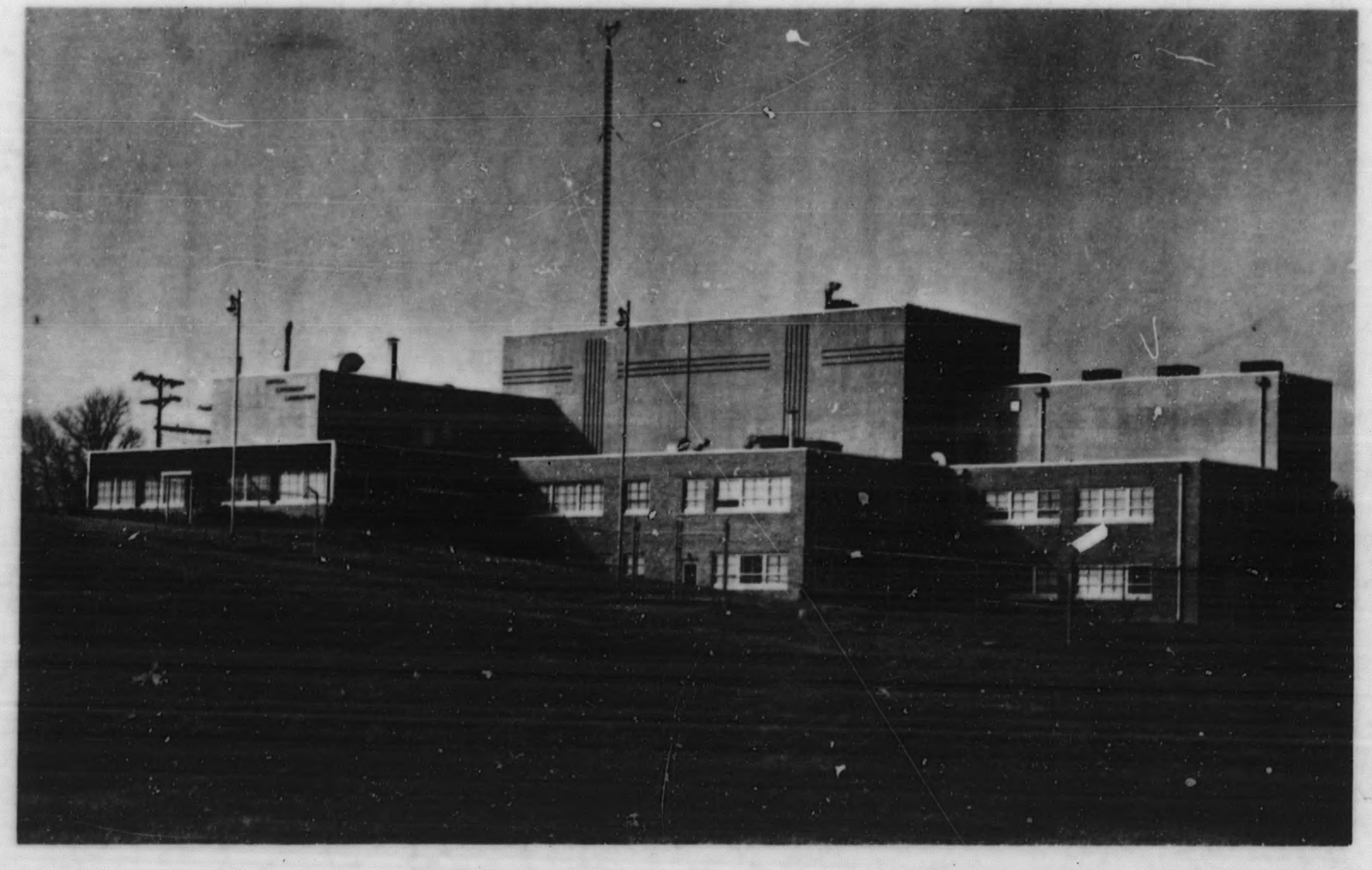
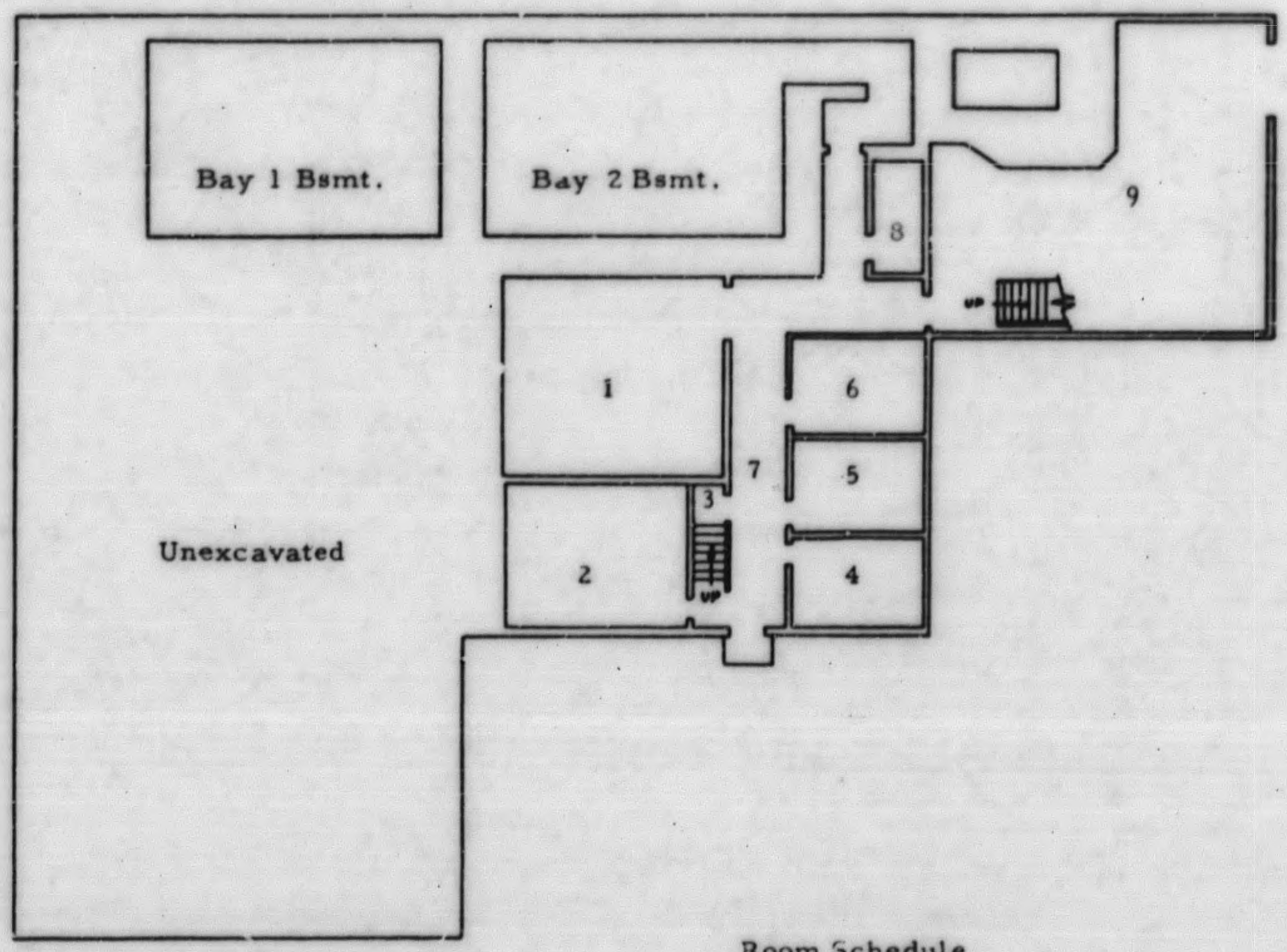


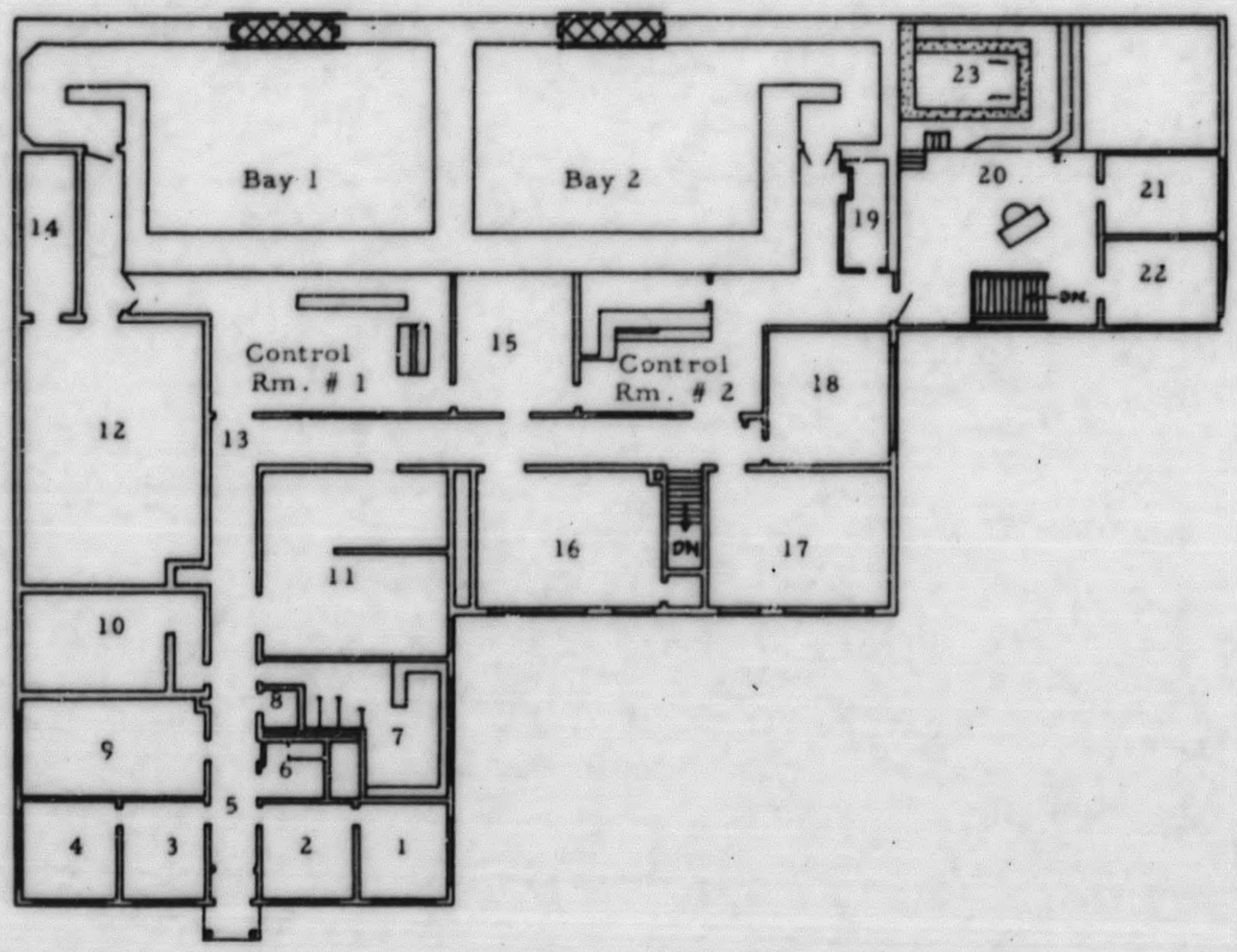
FIG. 9: CRITICAL EXPERIMENT LABORATORY – BASEMENT FLOOR PLAN



Room Schedule

1 - Sub-Assembly Rm. No. 2
2 - Physics Laboratory
3 - Janitor's Closet
4 - Office No. 8
5 - Office No. 9
6 - Office No. 10
7 - Hall
8 - Men
9 - Reactor Basement Area

FIG. 10: CRITICAL EXPERIMENT LABORATORY – FIRST-FLOOR PLAN



Room Schedule

1 - Office No. 1
2 - Office No. 2
3 - Office No. 3
4 - Office No. 4
5 - Entry Hall
6 - Women
7 - Men
8 - Janitor
9 - Health Physics Laboratory
10 - Counting Room
11 - Physics Shop
12 - Sub-Assembly Room No. 1
13 - Hall
14 - Storage
15 - Operations Office
16 - Electronic Shop
17 - Chemistry Laboratory
18 - Office No. 5
19 - Storage
20 - Control Room No. 3
21 - Office No. 6
22 - Office No. 7
23 - Pool Reactor

FIG. 11: FIRST-FLOOR PLAN OF THE LPR WING

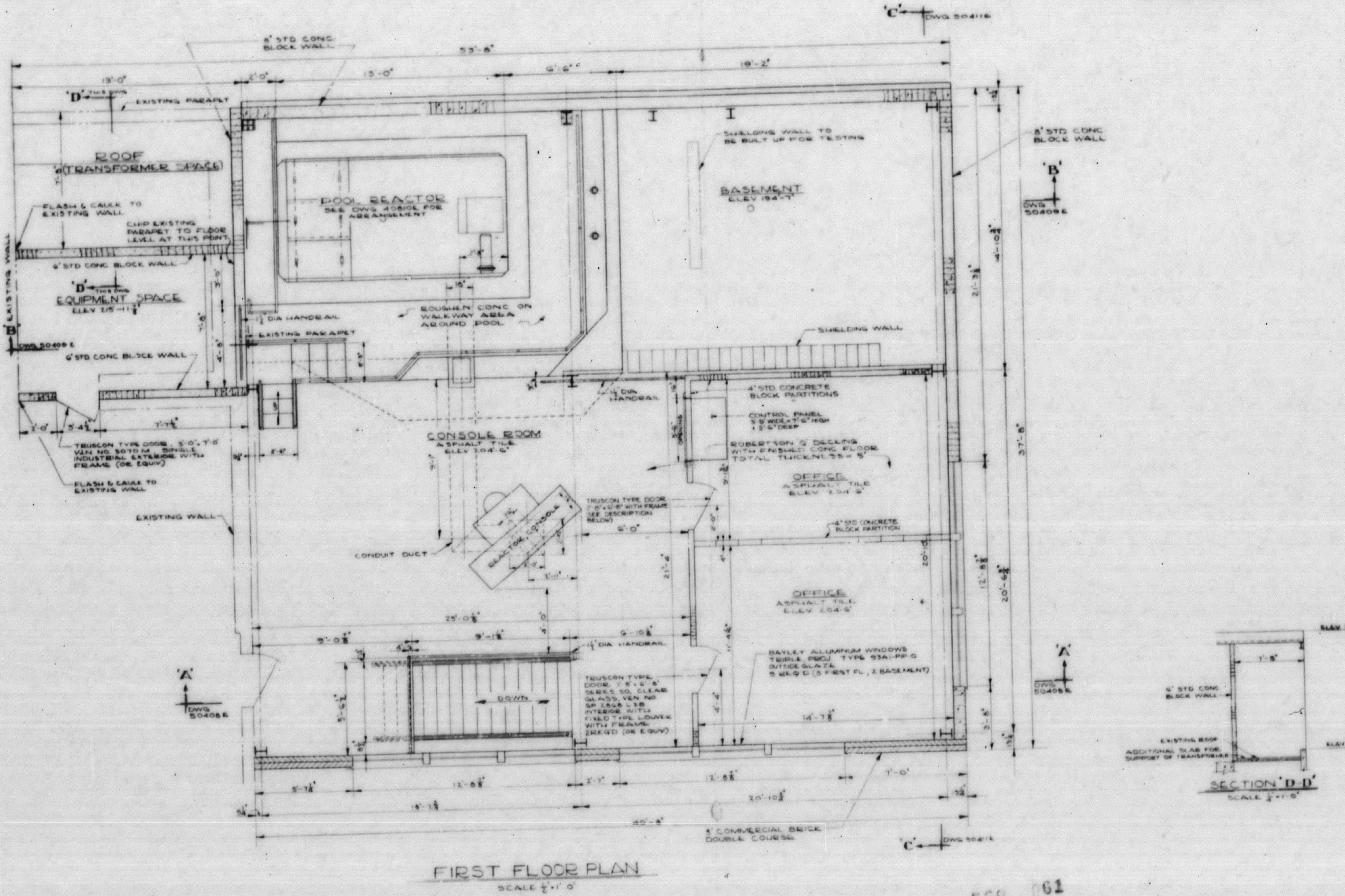
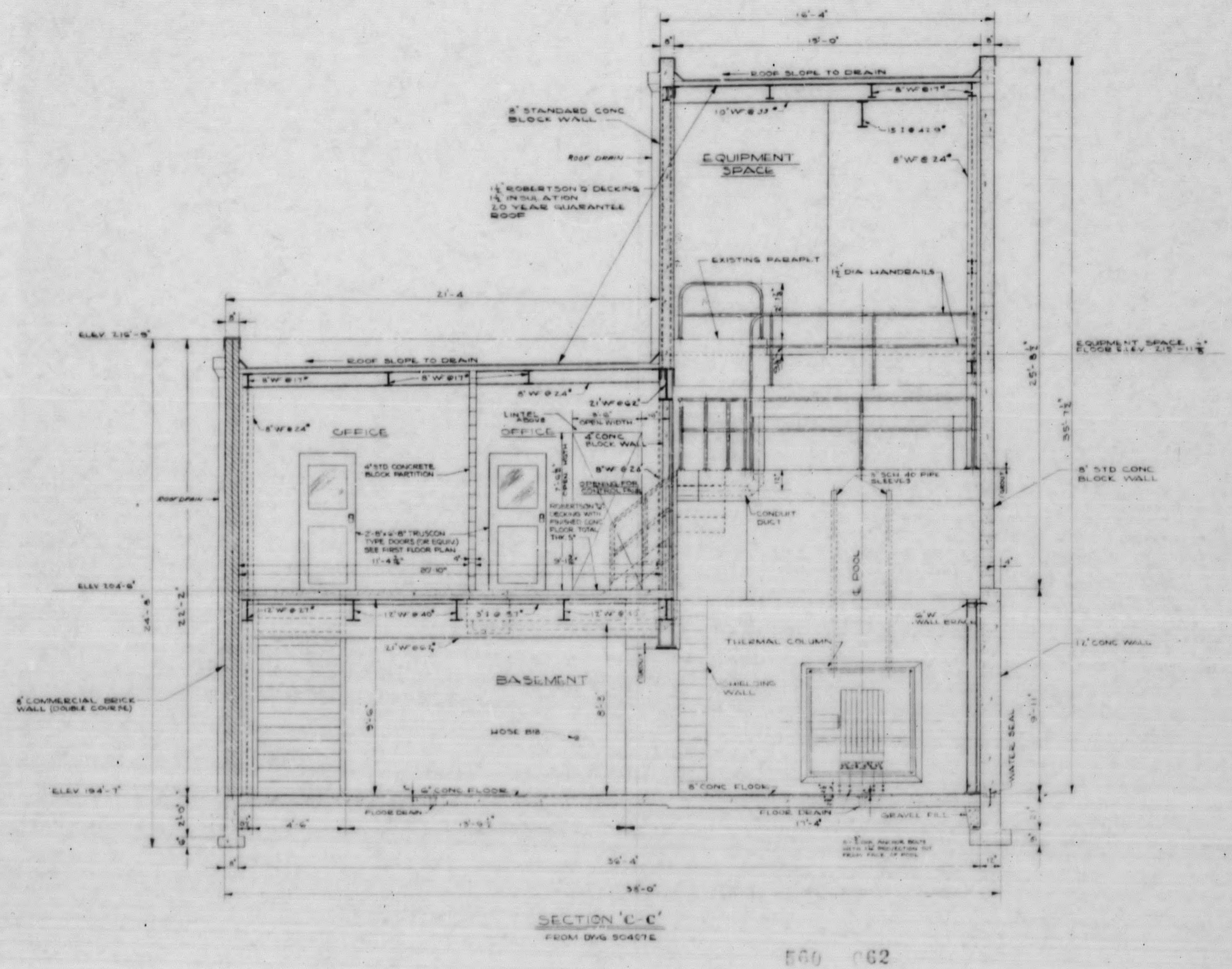


FIG. 12: ELEVATION OF THE LPR WING



APPENDIX C
HEALTH PHYSICS EQUIPMENT

Table V lists the available health physics equipment, which monitors the air constantly for radioactive particulate matter, and monitors the entire area for gamma radiation from operating reactors or fission products that escape into the reactor bays and other areas.

The entire laboratory is checked by a multi-channel remote area monitoring system, consisting of a series of gamma-sensitive ionization chambers, three master station units on the reactor control panels, and a recorder in the health physics office. This system is shown in Figure 13.

A constant air monitor — located above the LPR pool — contains an air suction pump, to collect particulate matter on a filter paper, a radiation detector, and a count-rate meter to measure activity. These data are continuously recorded in the Health Physics Office and over-signal is indicated audibly and visibly at the LPR control console. Two other air monitoring systems are available if needed.

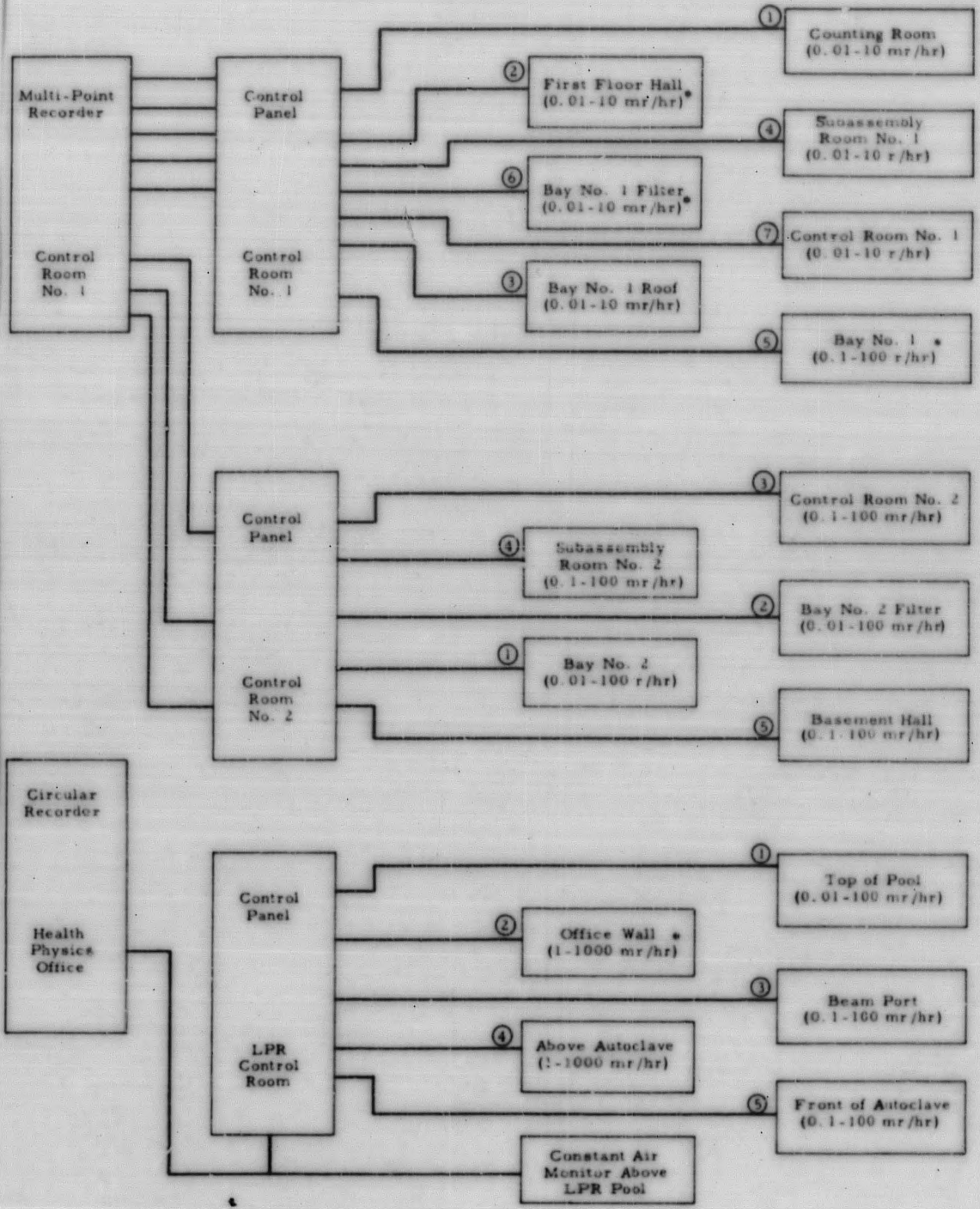
TABLE V
LIST OF HEALTH PHYSICS INSTRUMENTS

Instrument	Number
Count rate meter — G-M probe survey instrument	2
Complete end-window gas flow proportional counting unit	3
Beta-gamma survey meter — portable, indicating count rate	2
Alpha survey meter — portable, thin-window, ion chamber type	2

TABLE V (CON'T)

Instrument	Number
Neutron survey meter — portable, all energies	1
Beta-gamma survey meter — portable, (0-2500 mr/hr), ion chamber type	4
Beta-gamma survey meter — portable, (0.1-250 r/hr), ion chamber type	1
Fast neutron detector — removable, a-c operated	1
Background monitor (Monitron) — fixed, a-c operated, ion chamber type	1
Background monitors — a-c operated, counter type	3
Continuous air monitor — a-c operated	1
High volume air sampler — a-c operated	3
Film badge — gamma and neutron (per person)	1
Pocket dosimeter — gamma and neutron (per person)	2
Remote area monitoring system (Figure 13)	1

FIG. 13: REMOTE AREA MONITORING SYSTEM



*Sound Building Evacuation Alarm

APPENDIX D
EMERGENCY PLAN

1. Introduction

The experiments to be conducted in the Critical Experiment Laboratory present potential hazards common to reactor operations. Although the possibility of a disaster is remote, because of careful planning and design of the experiment, equipment, controls, and operating procedures, this emergency plan has been adapted to protect the health and safety of Company employees and area residents, in the event of a major disaster or accident at the laboratory.

The Critical Experiment Laboratory and the Nuclear Facilities Plant are located on a 550-acre site 4.5 air miles east of downtown Lynchburg, Virginia, about 2 miles from U. S. Highway No. 460. The plant employs nearly 600 people who work in a modern one- and two-story industrial structure. The laboratory normally employs about 30 people, although as many as 40 people occasionally occupy the building, located about 1200 ft from the plant.

This emergency plan is part of a comprehensive disaster plan for the entire site, but those portions relating to incidents at the Nuclear Facilities Plant have been submitted previously as the Nuclear Facilities Plant License Application, dated August 11, 1955, and are incorporated by reference.

2. Emergency Organization

The Laboratory Emergency Officer — the supervisor of the Critical Experiment Laboratory — or his appointed alternate, directs the action of all laboratory personnel in the emergencies covered by this plan. He integrates his actions with the Site Emergency Officer, who is responsible for the execution of the emergency plan over the entire site. The Site Emergency Officer also serves as the emergency officer for the

Nuclear Facilities Plant. The Health Physics Supervisor and the Plant Security Officer, or their appointed alternates, report directly to the Site Emergency Officer.

3. Equipment

Standard fire fighting and first aid equipment are available at the laboratory. The health physics instruments listed in Appendix C are kept at the laboratory, in operating condition. Standard operating procedures also require that a health physics instrument be kept in operating condition in the control room when an experiment is in operation. Additional fire fighting equipment, first aid and medical supplies, and health physics equipment are available on instant call from the Nuclear Facilities Plant.

4. Alarm System

If the radiation level in the laboratory or the LPR wing becomes too high, the building evacuation alarm will sound. This alarm can also be actuated manually, from either of the three control consoles (or from the front of the building) if the operator believes that a dangerous condition has arisen. The alarm is given by several loud bells located in strategic places around the laboratory. Each bell produces a 75-db sound level at 10 ft.

5. LABORATORY EMERGENCY PROCEDURE

When the alarm sounds, all people in the building leave from the nearest exit and assemble just inside the front door with all available health physics equipment. If the reactor has not scrammed automatically, the operator scrams it manually before leaving his post.

The Laboratory Emergency Officer is in complete charge. The Secretary immediately checks the list of occupants to determine whether everyone is accounted for. The Health Physicist obtains enough radiation data to make a proper safety evaluation of the building's occupants. Persons working under his direction make continuous surveys of the building and the surrounding area. Additional information is obtained from the Remote Area Monitoring System. After the Health Physicist and Laboratory Emergency Officer determine that the radiation level at the front of the building is negligible, or after the occupants have been evacuated to a safe area, the counting room technicians read pocket dosimeters to determine individual radiation exposures.

The Operations Supervisor endeavors to determine the cause of the alarm. A person designated by the Laboratory Emergency Officer immediately contacts the Guard House at the Nuclear Facilities Plant, using the regular phone lines or the special guardhouse telephone system. The Guard Shift Supervisor is informed of the situation, and the phone lines are kept open to supply additional information as it becomes available. Information on wind velocity and direction is obtained from the aerovane on top of the laboratory, and from the radiation surveys conducted outside the building.

6. Plant Emergency Procedure

Upon being advised of an emergency at the laboratory, the Guard Shift Supervisor summons the Site Emergency Officer, the Plant Health Physics Supervisor and the Plant Security Officer. The Site Emergency Officer is responsible for carrying out the procedures necessary to protect the health and safety of everyone at the plant site and in the surrounding territory. The Health Physics Supervisor immediately starts radiation surveys inside and outside the plant. The Plant Security Officer details men to keep people inside buildings and to keep traffic from entering the area until the incident has been fully evaluated.

The Site Emergency Officer summons police, fire, and ambulance service when required. He is also responsible for ordering evacuation of any building or of the entire site if necessary. Private automobiles will be used for evacuation. Since the site is surrounded by the James River on three sides, exit is possible in only one direction. Injured personnel requiring first aid or medical attention are removed from the scene of the incident by personnel familiar with safety equipment procedures.

7. Post Emergency Procedure

The preceeding emergency procedures are in force until allpersons are moved out of the danger area. Radiation and contamination levels, progressing back toward the location of the incident, are determined with portable survey instruments. Detailed surveys of surface and airborne contamination will localize and delimit areas of safe occupancy. Suitable barriers are erected to confine contamination and to prevent accidental radiation exposure.

A group of local physicians will provide emergency medical care. Background blood counts, already taken for all employees at the laboratory, will be repeated periodically. Nose smears, blood counts, and urine analyses are performed after the incident and medical treatment is begun if needed.

Decontamination will be initiated as soon as possible; the area will be rendered safe, within acceptable tolerance levels, before personnel are permitted to work in the vicinity of the incident. The USAEC is notified of the incident immediately, in compliance with Title 10, Part 20 of the Code of Federal Regulations, and applicable amendments, entitled "Standards for Protection Against Radiation". After a thorough investigation of the incident, a report will be prepared describing the nature of the experiment, the material involved, the damage to equipment and building, the medical and radiation history of the personnel affected, the decontamination procedures followed and the extent of contamination, the recovery of radioactive materials, the cause of the incident, and the means of preventing future incidents.

8. Arrangements With Outside Agencies

The Lynchburg Police Department and the Virginia State Police will be available on call to help the Plant Security Force handle an emergency.

A serious hazard to the people living around the site is very unlikely. This area is very sparsely populated, especially in the down-wind direction. If it ever becomes necessary to evacuate any of these people, the the State Police will assist.

Lynchburg city officials and members of the Civil Defense organization have been informed of the potential hazards. The Civil Defense organizations will carry out any action necessary to assure the safety of Lynchburg and area residents.

9. Indoctrination

All laboratory personnel have copies of the emergency plan, and conferences will be held to ensure that everyone is thoroughly acquainted with the plan and associated individual duties. Before a new core is loaded, several drills will be conducted to supply additional training and to test the plan's effectiveness.

APPENDIX E
SUMMARY OF NUCLEAR CALCULATIONS

1. Estimated Reactivity of Initial Assembly

The estimated reactivity of the basic assembly, with aluminum shim rods and aluminum in the test holes is

$$k_{eff} = 1.07 \pm 0.06.$$

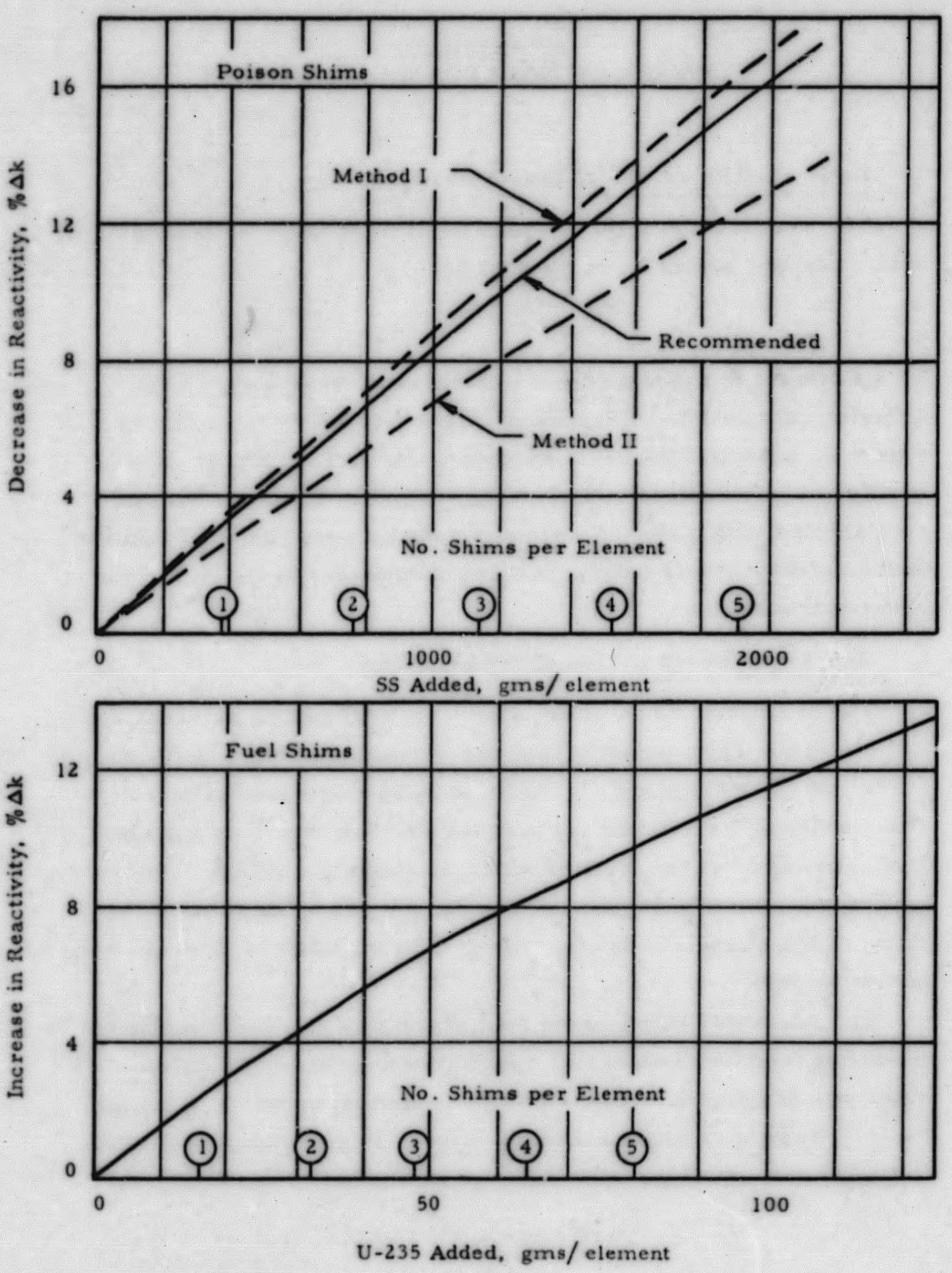
This estimate is based on a two-dimensional, four-group PDQ diffusion calculation configuration slightly different from that of Figure 2, but extrapolated to the conditions of Figure 2 by several simple one-dimensional relative calculations. Figure 2 was not recalculated with a two-dimensional code because the uncertainties in the validity of this calculation are comparable to the uncertainty in the extrapolation.

2. Reactivity Effect of Fuel Element Shims

The reactivity effect of the stainless steel poison shims in the fuel elements was calculated in a one-dimensional core model for two cases, the beryllium and aluminum being homogenized differently. This calculation takes into account the addition of poison and the displacement of water, both of which decrease reactivity. The results are shown in Figure 14, for 382 gm of stainless steel per shim. Each shim is 0.050 in. thick, 2.50 in. wide, and 24 in. long (active length).

The reactivity effect of the fuel shims was calculated using the results of the two-dimensional calculations, assuming that no water was displaced. This condition is approximated by mounting the U-Al foils on a lucite backing. Figure 14 gives the results for 10 U-Al foils, or 16 gm of U-235 per shim.

FIG. 14: REACTIVITY WORTH OF POISON AND FUEL SHIMS



3. Shim Rod Worth

The estimated reactivity worth of the four cruciform shim rods and the central blade is 25±10% Δk, by extrapolating two-demensional calculations. This worth breaks down to 8±5% Δk for each inner cruciform rod, 3±2% Δk for each outer cruciform rod, and 3±2% Δk for the central blade.

4. Reactivity Worth of Various Components

The reactivity worth of various components in the assembly has been estimated in order to assess the potential hazard should these components be removed and the void filled with water. A variety of one-dimensional calculational methods were used, with the core homogenized to maximize the effects.

a. Central Shim Rod Blade

If an aluminum central blade is removed, and the void is filled with water, reactivity will increase by 1.0±0.5% Δk.

b. Beryllium Reflector

If the entire beryllium reflector is removed, and the void is filled with water, reactivity will decrease by about 6±2% Δk.

c. Internal Beryllium Elements

If one internal beryllium element is removed, and the void is filled with water, reactivity will decrease by about 1.3% Δk.

d. External Aluminum Reflector Element

If the beryllium reflector is completely surrounded by a 3-in. aluminum reflector, removal of the aluminum and replacement of the void with water will increase reactivity by only 0.12% Δk.

e. Fuel Elements

The removal of any fuel element and replacement of the void with water will decrease reactivity.

f. Aluminum Test Insert

The removal of one 6-in.-square aluminum inserts and replacement of the void with water will decrease reactivity by 1.2%.

g. Test Inserts

If one typical test loop insert (Fig. 4) is substituted for aluminum, reactivity will decrease by 0.4%. The dummy loop containing Al_2O_3 instead of UO_2, will decrease reactivity by about 2%. The most highly loaded test insert anticipated (ETR fuel element) will increase reactivity by about 2%. The range of reactivity contributions from various test loop inserts is expected to be within ±3%.

5. Reactor Parameters

a. Temperature Coefficient of Reactivity

The calculated total temperature coefficient of reactivity is -2.9×10^{-4}/°C at 20 C. The calculated temperature coefficient due to fuel heating alone is -0.6 to -1.8×10^{-6}/°C. These calculations compare favorably with those done for the ETR critical experiment,[12] -3.8×10^{-4}/°C, and -1.8×10^{-6}/°C, but they are not in agreement with ETRC measurements which report -5.6×10^{-5}/°C for the total temperature coefficient over the range of 70 to 120 F.[13] The cause of this disagreement is not known, but it may be due to the lower metal-to-water ratio in the LPR fuel elements. In any event, the temperature coefficient is probably not less than about -6×10^{-5}/°C.

b. Void Coefficient of Reactivity

The calculated void coefficient of reactivity is in the range of -2.4 to -4.7×10^{-3} Δk/% void in fuel element, to -1.7 to -3.3×10^{-3} Δk/% void in the water in the fuel element. This compares with the calculated ETR value of -2.5×10^{-3} Δk/% void in the water. These void coefficients are over-all averages. No point-by-point calculations were made; instead the results of calculations for ETR critical experiment are reported, since they are quite similar in nuclear respects to the LTR. The ETR calculations showed that the void coefficient for large voids is never positive anywhere in the reactor, and that, as a function of void size, the coefficient may be slightly positive, up to about 1.5-cm void diameter. Later measurements made in the ETRC and the ETR have shown that the void coefficient is negative for all void sizes and fuel positions.[13]

c. Neutron Lifetime

The calculated prompt neutron lifetime in the basic assembly is 9±4 x 10^{-5} sec, which compares favorably with the ETR calculation of 8.1 x 10^{-5} sec. The expected lifetime of experiments in the test holes was not calculated, although the minimum lifetime is assumed to be 7 x 10^{-5} sec on the basis of ETR calculations.[12]

6. Nuclear Safety Calculations

The criticality of the 24 LPR fuel elements stored in the storage pit was computed using a two-group, one-dimensional nuclear safety code, applied to an infinite lattice. Each fuel element was assumed to contain 300 gm U-235 (including shims) with a 20-in. spacing between rows and a 2-in. edge-to-edge spacing between elements in a row. The calculation gave a k_{eff} of 0.74±0.05 for the flooded condition.

The criticality of eight LPR fuel elements, each containing 300 gm of U-235, on a pitch of 6.07 in. between rows and 6.39 in. between elements, was calculated by the same code using cylindrical geometry with infinite water reflector. A k_{eff} of 0.68±0.05 was obtained.

The maximum possible interaction between the LTR critical core and four excess LPR fuel elements (with a 300-gm loading of U-235 each and separated by 28 in. of water) was estimated to be less than 0.01% Δk.

APPENDIX F
REACTOR POWER CALCULATIONS

1. Maximum Operating Power Level

The reactor — cooled by natural circulation — will operate at a maximum power level of 1000 watts. These conditions should not create any hazard since many of the MTR-type elements in the LPR have been operated for long periods of time at power levels above 100 kw. There is some uncertainty about the temperature distribution in the U-Al fuel shims that may be added to some of the LPR fuel elements.

At 1000 watts, the maximum thermal neutron flux will be about 1×10^{10} n/cm^2-sec. If the U-Al fuel shims were perfectly insulated, transferring no heat to the water, the maximum temperature rise would be about 4.7 °C/min. This could be tolerated since only a few 1000-watt runs are contemplated, none of which would last more than 30 min.

Under more realistic but still conservative assumptions, the steady state temperature distribution may be calculated assuming a 1/8-in. lucite plate on each side of the U-Al fuel shims. In slab geometry, the temperature differences are

$$T_0 - T_1 = \frac{Q}{2k_1} a^2 \qquad \text{and} \qquad T_1 - T_2 = Qa\left(\frac{b-a}{k_2} + \frac{1}{h}\right),$$

where T_0 = temperature at center of U-Al fuel,
T_1 = temperature at U-Al — lucite interface,
T_2 = bulk temperature of water = 80 F,
Q = volume heat generation = 1.85×10^4 Btu/hr - ft^3,
a = half-thickness of U-Al section, 0.00208 ft,
b = thickness of lucite, 0.0125 ft,
k_1 = thermal conductivity of aluminum, 118 Btu/hr - ft - °F,
k_2 = thermal conductivity of lucite, 0.05 Btu/hr - ft - °F, and
h = heat transfer coefficient of water, 7000 Btu/hr - ft^2 - °F.

Under these assumptions, at 1000 watts, the temperatures are T_2 = 80F,

T_1 = 88 F, and T_0 = 88 F. Actually there will be some thermal resistance from possible air gaps between the U-Al foils and between the foils and the lucite. Assuming that air gaps between each foil are equal in thickness to each foil (0.01 in.), the effective thermal conductivity of the U-Al section is reduced to k_1 = 0.03 Btu/hr - ft - °F, which will increase $T_0 - T_1$ from 3×10^{-4} F to 1.3 F — or on an absolute basis, T_0 = 89 F. Therefore, the temperatures in the fuel shims will be very moderate during sustained 1000-watt operation.

2. Startup Accident

It is assumed that the reactor is critical and that all rods are withdrawn at their maximum speeds. However, ganged rod operation is impossible, so the worst situation is the withdrawal of the regulating rod which adds reactivity at the rate of 0.00025 Δk/ sec.

A Newson-type analysis is used to determine the minimum period resulting from this accident:[16]

$$T > \left[\frac{L}{2R \ln(P_1 / P_0)}\right]^{\frac{1}{2}}$$

where T = minimum period, sec

L = prompt neutron lifetime, 7×10^{-5} sec,

P_0 = initial power, and

P_1 = final power.

This conservative analysis neglects delayed neutrons.

Two more cases are considered: in the first, the reactor is critical at source power, $P_0 = 10^{-3}$ watts; in the second, the reactor is critical at the maximum operating power, $P_0 = 10^3$ watts. P_1 is the power level where safety rod trip occurs. For case one it is conservatively considered that the trip does not occur until the uncompensated ion chambers come on scale, P_1 = 100 watts. In case two the trip point will be 1.5 times full scale, P_1 = 1500 watts. Therefore, the minimum reactor period at which trip occurs is 0.111 sec for case one and 0.587 sec for case two.

Starting with t = 0 at the trip point, there will be a 0.030-sec delay before the rods begin to fall in (measured delay time). It is further assumed that the rods fall in at an acceleration (a) 500 cm/sec^2, and that only two safety rods, worth 3% each, are effective. If W is the total worth of the safety rods divided by their active lengths, $W = 0.06/61 = 0.001\ \Delta k/\text{cm}$, then reactivity is reduced $1/2\ aWt^2$ by the safety rods, and the maximum power is

$$P_{max} = P_1 \exp\left[\frac{0.030}{T} + \frac{(t-0.030)}{T} - \frac{1}{6}\frac{aW}{L}(t-0.030)^3\right].$$

The maximum power level is reached at the time,

$$t_{max} = \left(\frac{2L}{aWT}\right)^{\frac{1}{2}} + 0.030.$$

For case one, $t_{max} = 0.080$ sec, and $P_{max} = 1.77\ P_1 = 177$ watts.
For case two, $t_{max} = 0.052$ sec, and $P_{max} = 1.25\ P_1 = 1875$ watts.

The excursion was recalculated using the more conservative assumption that the power increased during the calculated period for 0.350 sec, the measured delay time to half rod insertion. Under this assumption, $P_{max} = 2360$ watts for case one, 2730 watts for case two.

In conclusion, the maximum power level reached in a startup accident is definitely less than 3000 watts, well under the heat flux limit for MTR-type fuel elements. The U-Al fuel shims will not be damaged either, since a power level increase by a factor of three will increase the calculated temperatures to $T_0 = 108$ F, $T_1 = 104$ F, and $T_2 = 80$ F.

3. Step Reactivity Addition

For step reactivity additions the previous procedure can be followed, but the prompt period is given by

$$T = \frac{L}{k_{ex} - 0.0075}.$$

Again no inherent shutdown mechanism is postulated, and the excursion is terminated by the scram of two safety rods. For a 1.5% reactivity step, T = 9.3 ms, and for a 1.0% reactivity step T = 28 ms.

The worst case will be steady operation at 1000 watts, with a trip at 1500 watts, and a 30-ms delay before the rods begin to fall in. This calculation gives a maximum power of 930 MW for the 1.5% step, 47 kw for the 1.0% step.

The 1.5% step will almost certainly melt the aluminum in the LPR fuel elements and the fuel shims. The 1.0% step will not damage the LPR fuel elements. The U-Al fuel shims will reach a central temperature of about 550 F, using the earlier conservative assumptions of lucite on both sides and insulating air gaps. This is well under the melting point of aluminum, 1200 F; however, the temperature inside the lucite will approach 480 F which is above the softening point. Therefore, some distortion of the lucite is expected.

APPENDIX G
DISCUSSION OF SPERT AND BORAX EXPERIMENTS

The self-regulating characteristics of water moderated reactors using MTR- and ETR-type fuel elements have been extensively investigated in the BORAX and SPERT experiments. The applicable SPERT and BORAX data presented here can be used to predict the maximum reactivity additions that can be safely self-regulated.

The SPERT experiments included a study of instantaneous reactivity additions, up to 1.4% (asymptotic period of 7 ms), at an initial temperature of 20 C and a power level of 5 watts.[15] The power excursion is characterized by a large peak followed by a gradual "tailing off" for periods exceeding about 35 ms. For an excursion yielding an asymptotic period less than 35 ms the power level becomes oscillatory, with amplitudes increasing as the periods decrease. No divergent oscillations were observed in this series of experiments.

The data can be correlated on a log-log plot as shown in Figures 15 and 16 (taken directly from references 15 and 16. An extrapolation of the fuel temperature data suggests that the fuel would melt when an asymptotic period less than 4 to 5 ms (>2% Δk) is introduced. For an excursion of this type the peak power level would be about 3000 MW, and the energy release would be about 20 to 50 MW-sec.

Other SPERT experiments were done with reactivity added at rates from 0.0008 Δk/sec to 2.25% Δk with the water at 20 C. When the water was initially at saturation temperature, excess reactivity of 1.75% was added.[12] Instabilities in the SPERT experiments were observed[17] when ramp reactivities of 2% (or greater) excess reactivity were introduced. These instabilities, similar to the "chugging" observed in the BORAX experiments, are large power peaks with very short effective periods at approximately 1-sec intervals. Total reactivity additions up to 2.75% were introduced, lasting for 2 to 3 min before the reactor was deliberately scrammed.

FIG. 15: MAXIMUM POWER LEVEL — SPERT EXPERIMENTS

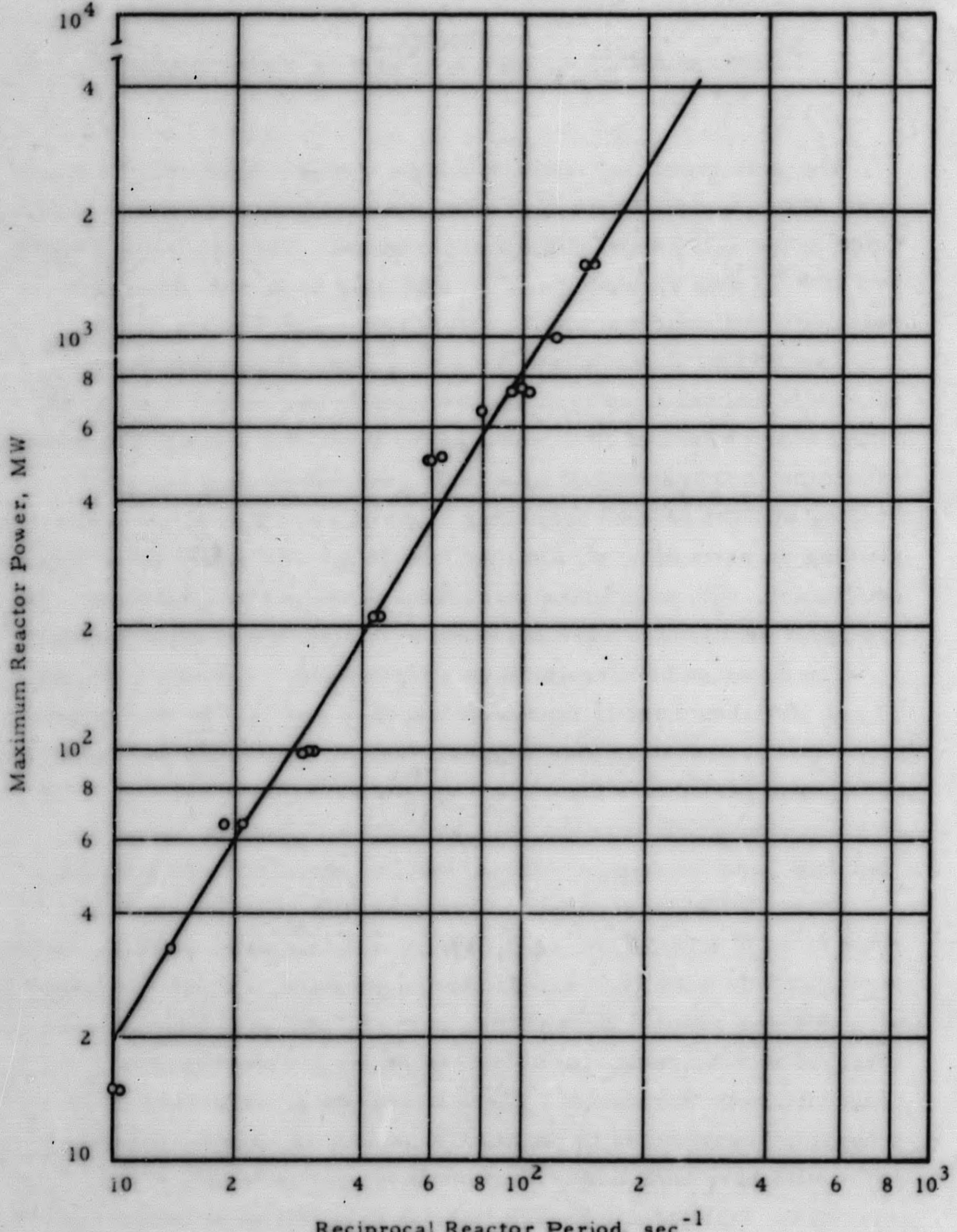
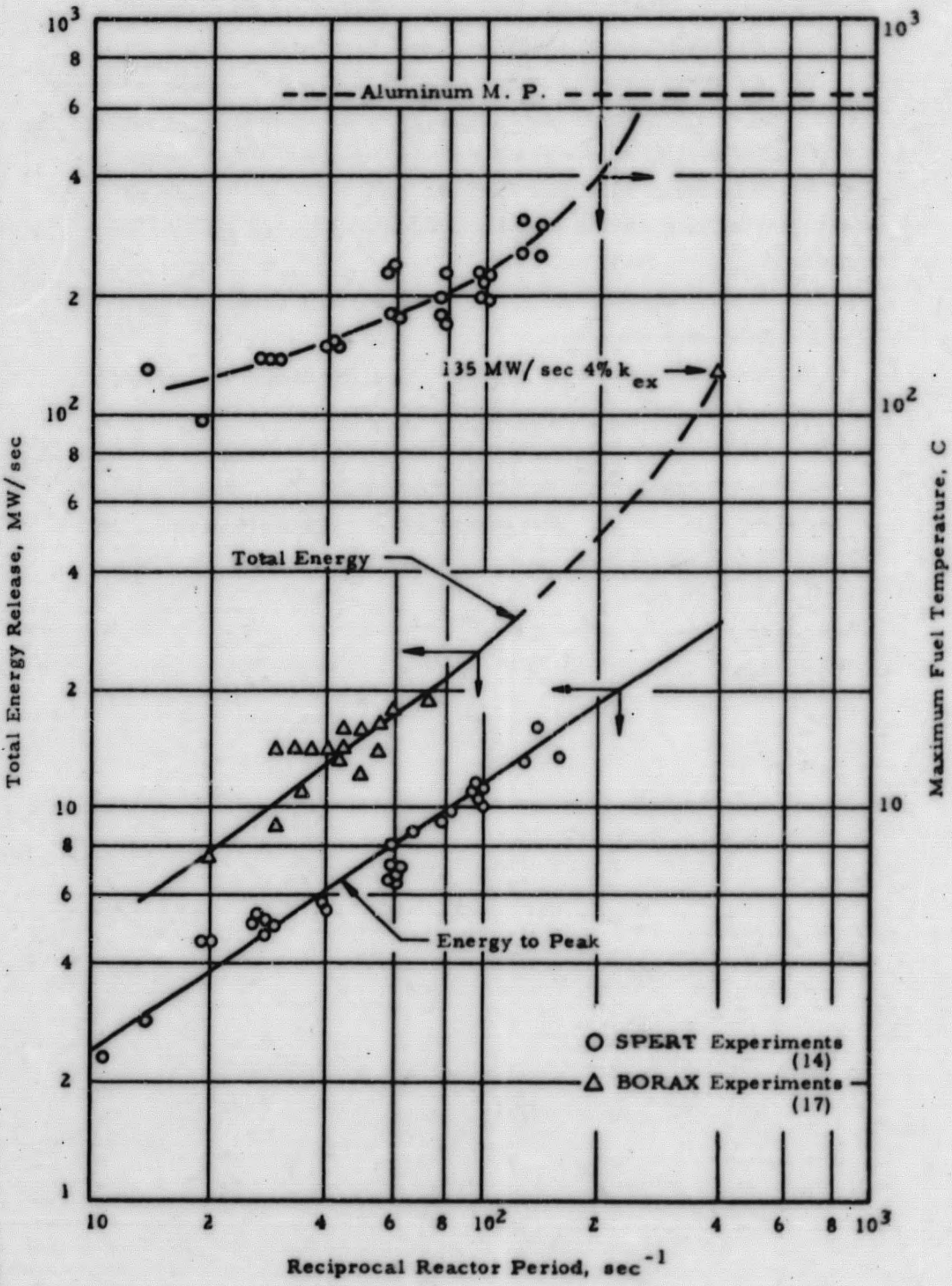


FIG. 16: TOTAL ENERGY RELEASE AND FUEL TEMPERATURE



In the BORAX tests the maximum step reactivity addition was 2.1% when the water was initially at saturation temperature.[16] The fuel did not reach its melting point, but some plates were deformed. Therefore, no experiments with water initially at 20 C were done beyond step additions of 1.25%

The self-regulatory properties of this type of reactor were not calculated by B&W because of the many assumptions required and the unknown reliability of the results.[18] As reported in reference 12, burnout in an MTR-type fuel element with 0.015-in. clad will occur for an asymptotic period of about 0.0025 sec, or an excess reactivity of nearly 3%.

Table VI compares the SPERT and BORAX experiments to the LTR critical experiments.

In conclusion, the LTR critical experiment will be self regulating for step and ramp reactivity additions in the range of 1.5% to 2.0%. Excursions from a reactivity increase of 2.0 to 2.5%, or higher, probably won't melt the fuel elements although power oscillations may occur. The maximum energy release for a 2% excursion will be on the order of 20 to 50 MW-sec.

TABLE VI

IMPORTANT PARAMETERS FOR SPERT, BORAX, AND LTR CRITICAL EXPERIMENT CORES

Parameter	BORAX Core I	BORAX Core II	SPERT	LTR Critical Expt
Prompt neutron lifetime, sec	6.5×10^{-5}	7.5×10^{-5}	5×10^{-5}	$\approx 7 \times 10^{-5}$
Void coefficient, Δk/% void	-2.4×10^{-3} *	-1.0×10^{-3} *	-1.9×10^{-3}	-1.7 to 3.3×10^{-3}
Temp coefficient, $\Delta k/°C$	-1.2×10^{-4} **	-6.9×10^{-5} **	-9×10^{-5}	-6 to 30×10^{-5}
Cross section of fuel element, in.	3 x 3	3 x 3	3 x 3	3 x 3
Plates per element	18	10	17	10
Plate thickness, in.	0.060	0.060	0.060	0.050
U-235 per element, gm	139	93 or 157	168	190
Metal-water ratio	0.626	0.422	~20.6	~20.4

*Based on measured replacement of 10% of core water by steam at 200 F.

**Average from 27 to 93 C.

APPENDIX H
EFFECTS OF MAXIMUM CREDIBLE ACCIDENT

1. Maximum Credible Accident

To set an upper limit on the environmental hazard created by the maximum credible accident, it is postulated that some unknown mechanism instantaneously adds 2% excess reactivity to the just-critical reactor. During the excursion, 50% of the maximum amount of U-Al alloy in the fuel shims and 10% of the fuel plates in the LPR fuel elements are assumed to melt because of coincident plugging of some fuel element coolant channels. SPERT and BORAX data indicate that the fuel elements should not melt during a 2% excursion; but an appreciable number of the U-Al fuel shims may melt because of poorer heat transfer. The maximum fraction of total fuel in the U-Al shims is 0.26, or 80 gm U-235 extra per fuel element.

Based on these assumptions, there are two sources of activity freed from the fuel elements: that generated more or less instantaneously in the excursion, and the residual activity built up in the LPR due to the high power operation of the LPR fuel elements before the LTR critical experiments began. In comparison with the LTR critical experiment, the activity generated in the course of the LTR critical experiment program may be neglected. Based on experimental data shown in Figure 15, a 2% excursion will generate up to 50 MW-sec of energy. Therefore the effective excursion in terms of activity release is

$$W = 50[0.50 \times 0.26 + 0.10 \times 0.74] = 10 \text{ MW-sec.}$$

In addition to the 10 MW-sec effective burst, 10% of the activity previously built up in the LPR fuel elements will be released. The integrated power of these fuel elements will be less than 1 MW-day, with a two week cooling time before the LTR critical experiments begin. During the cooling time all of the volatile fission products will have decayed except I-131 and Xe-133, which have relatively long halflives. To more accurately compute the quantity of I-131 and Xe-133 present in the fuel elements before the excursion, this equation is used:

$$N_{F.P.} = 3.1 \times 10^{10} \frac{Py}{\lambda} \left(\frac{1 - e^{-\lambda t_1}}{1 - e^{-\lambda T}} \right) e^{-\lambda t_2} (1 - e^{-\lambda t}).$$

In this equation for cyclical operation, the maximum operating history of the LPR in the month preceeding the LTR critical experiments is expected to be: 1 run per day at 200 kw, lasting 0.5 hr, then $P = 2 \times 10^5$ watts, $t_1 = 0.5$ hr, $t_2 = 23.5$ hr, $T = 24$ hr, and $t = 30$ days. "y" is the fission product yield, and λ is its decay constant. Applying a further attenuation of $e^{-\lambda t}$ where t is 2 weeks between LPR and LTR operation, 1.02×10^{18} I-131 atoms and 8.70×10^{17} Xe-133 atoms would be present at the time of the excursion .

2. Gamma Activity in the Pool

Some of the activity in the pool will be dissolved in the water and some will remain in the fuel elements. In computing the maximum pool activity no loss of activity from the pool itself is assumed. The gamma activity can be computed from the equations[19]

$$Q_a = 4.8 \times 10^{16}\, t^{-1.21}\, W, \text{ and}$$
$$Q_b = 2.3 \times 10^{14}\, P\, [\, T^{-0.21} - (t_0 + T)^{-0.21}],$$

where Q_a is the gamma activity generated in the burst (Mev/ sec) and Q_b is the activity from previous LPR operation. To obtain an upper limit on the gross activity, Q_b, P is assumed to be 1000 kw, t_0 is 1 day, and T is the time after LPR operation ceases, or $T = t + 2$ weeks. Being interested in past excursion times much shorter than two weeks, t is neglected and this contribution is assumed constant during the period immediately following the excursion, so that $T = 2$ weeks.

Therefore the activity release to the water consists of two components, Q_a with $W = 10$ MW-sec, and 10% of Q_b, or

$$Q_{water} = 4.8 \times 10^{17}\, t^{-1.21} + 1.9 \times 10^{13}\, .$$

The activity remaining in the core is

$$Q_{core} = 1.92 \times 10^{18} t^{-1.21} + 1.7 \times 10^{14},$$

where $W = 50 - 10 = 40$ MW-sec.

The total activity, for various times after the excursion, assumed to be uniformly dissolved in the water, is listed in Table VII. The dose rates through the side wall concrete shielding and 3 ft above the top of the pool are given. The former is higher by a factor of about five.

TABLE VII

POOL WATER ACTIVITY AND DOSE RATES

Time After Excursion	Total Mev/sec in Water	mc/cm³ in Water	Dose Rate, r/hr thru concrete	Dose Rate, r/hr top of pool
10 sec	2.96×10^{16}	23.7	0.0036	1.45×10^{5}
1 min	3.43×10^{15}	2.9	0.0004	1.68×10^{4}
10 min	2.24×10^{14}	0.19		1.10×10^{3}
1 hr	4.25×10^{13}	0.035		2.09×10^{2}
10 hr	2.05×10^{13}	0.017		1.01×10^{2}

Due to the residual activity in the core, the dose rate above the pool will not be measurably increased because of the depth of water above the core.

3. Volatile Fission Product Activity

The volatile fission products — Kr, Xe, and possibly Br and I — might escape from the pool water. Table VIII lists the activities created during the excursion as well as the activities of T- 131 and Xe-133. Only the gamma emitting isotopes with halflives longer than 1 min were considered, on the basis of decay schemes and nuclear properties listed in reference[20]

Of these volatile fission products, Kr and Xe are assumed to be instantaneously transferred to air. The Br and I probably will not transfer because of their small quantity in terms of mass, and their

TABLE VIII

VOLATILE FISSION PRODUCT ACTIVITY IN WATER

Time After Excursion	Total Curies in Water Br	Kr	I	I-131	Xe	Total	Total μc/ cm³ water
0	27.3	51.5	223	28.0	35.5	341	7.3
1 min	26.8	51.2	222	28.0	35.5	335	7.2
10 min	22.4	48.2	204	28.0	35.5	310	6.7
1 hr	7.4	34.3	134	28.0	44.8	220	4.7
10 hr	3×10^5	1.9	38	27.2	35.2	78	1.7

TABLE IX

DOSE RATE AND VOLATILE FISSION PRODUCT ACTIVITY IN AIR

Time After Excursion	μc/cm³ in Air		Dose Rate in Bldg, r/hr		Dose Rate 50 ft outside Bldg, mr/hr	
	Kr+Xe	Kr+Xe+I	Kr+Xe	Kr+Xe+I	Kr+Xe	Kr+Xe+I
0	0.087	0.310	3.8	13.5	0.84	2.98
1 min	0.087	0.308	3.8	13.4	0.84	2.96
10 min	0.084	0.288	3.7	12.5	0.80	2.76
1 hr	0.079	0.203	3.4	8.8	0.76	1.95
10 hr	0.037	0.075	1.6	3.3	0.36	0.72

solubility in water. However, because of the particular hazard of I-131, two cases are assumed: the release of Kr and Xe to the air, and the release of Kr, Xe, and I to the air. This activity will quickly diffuse into the volume of the LPR wing. The concentration in air, assuming uniform mixing, is given in Table IX along with the dose to a poison at the center of a sphere equal in volume to the LPR wing.[21]

4. Environmental Hazard

To determine the hazards to the general public, the worst instantaneous release of all the volatile fission products in the building is calculated. The diffusion of the radioactive cloud is governed by Sutton's equations and the atmospheric diffusion parameters listed in Appendix A.[19] The cloud dose for inversion conditions is shown in Table X.

TABLE X

GAMMA CLOUD DOSE FROM VOLATILE FISSION PRODUCTS (INVERSION CONDITIONS)

	Total Dose, mr	
Distance	Kr + Xe	Kr + Xe + I
50 ft	24.7	88.7
440 ft	9.6	34.2
1200 ft	6.2	20.7

The only environmental hazard of consequence is the inhalation of I-131 — assuming it can escape from the pool water. The critical organ for I-131 is the thyroid gland where the maximum permissible short-term inhalation dose is 0.70, 17, and 170 μc.[11] This will be caused by an exposure of 0.3 rem the first week, or 15.7 rem the first year, or 150 rem in 70 years. The number of microcuries inhaled from the radioactive cloud is (average) $I_1 = 7.5 \times 10^{-4}\ Q/x^{1.75}$, and

(inversion) $I_2 = 6.0 \times 10^{-3}\ Q/x^{1.50}$,

where Q is the number of microcuries of I-131 instantaneously released from the building and x is the distance from the building to the observer. (The constants in the integration of Sutton's equation have been included.) A breathing rate of 17 l/min was used. Table XI lists the thyroid doses resulting from the release of all the I-131 in the water (28 curies) to the air.

TABLE XI

DOSE DELIVERED TO THYROID BY I-131, rem

	Normal Conditions				Inversion Conditions			
Distance, ft	μc inhaled	first week	first yr	70 yr	μc inhaled	first week	first yr	70 yr
50	178	76.5	165	165	2800	1210	2610	2610
440	3.95	1.7	3.63	3.63	108	46.0	102	102
1200	0.70	0.29	0.63	0.63	24.2	10.2	22.3	22.3
2500	0.19	0.08	0.18	0.18	8.3	3.4	7.7	7.7
5500	0.05	0.02	0.05	0.05	2.5	1.1	2.3	2.3

REFERENCES

1 To Supply Irradiation Services, Proposal EA-438 and Supplements, The Babcock and Wilcox Company, Lynchburg, January 13, 1959.

2 Application for Amendment AEC License CX-1 to Cover the Operation of the Nuclear Merchant Ship Critical Experiment in Bay No. 1, BAW-103, The Babcock and Wilcox Company, Lynchburg, July 1959.

3 Hazards Summary Report for Extension of Light Water Moderated, Thorium-Uranium Fueled Critical Experiments, BAW-91, The Babcock and Wilcox Company, Lynchburg, February 1959.

4 Liquid Metal Fuel Reactor Experiment - Critical Experiment - Final Hazards Evaluation, BAW-1049-1, The Babcock and Wilcox Company, Lynchburg, June, 1958.

5 Pool Test Reactor Hazards Evaluation - Final Report, BAW-74 and supplements, The Babcock and Wilcox Company, Lynchburg, May, June and July 1958.

6 To Permit Two Megawatt Operation of the Lynchburg Pool Reactor, BAW-74, Supplement 3, The Babcock and Wilcox Company, Lynchburg, October 16, 1959.

7 Application for License for Fuel Element Plant, Docket No. N-27 and supplement, The Babcock and Wilcox Company, Lynchburg, August 1955.

8 Critical Experiment Facility Hazards Evaluation, The Babcock and Wilcox Company, Lynchburg, February 1956.

9 Report on Hydrology, Meterology, and Waste Disposal of an Industrial Site in Campbell County, Virginia, Prepared for the Babcock and Wilcox Company, February 9, 1955.
(Morgan, Col. J.M., Jr.,)

10 Local Climatological Data for Lynchburg, Virginia, United States Weather Bureau, 1958.

11 Meterology and Atomic Energy, AECU-3066, United States Deptment of Commerce, Weather Bureau.

12 de Boisblanc, D.R., et al., Engineering Test Reactor Critical Facility Hazards Summary Report, IDO-16332, Phillips Petroleum Company, Idaho Falls, March 27, 1957.

13 de Boisblanc, D.R., et al., Engineering Test Reactor Critical Facility Hazards Summary Report - Supplement I, Phillips Petroleum Company, Idaho Falls, January 24, 1958.

14 Newson, H.W., Control Problems in Piles Capable of Very Short Periods, ORNL-1857, Union Carbide Nuclear Corporation, Oak Ridge, April 1947.

15 Schroeder, F., et al., "Experimental Study of Transient Behavior in a Sub-Cooled, Water-Moderated Reactor", Nuclear Science and Engineering, 2, 96, (1957).

16 Drebuch, J.R., Experimental Determination of Self-Regulation and Safety of Operating Water-Moderated Reactors, Geneva Paper p/481, 1955.

17 Forbes, S.G., Schroeder, F., and Nyer, W.E., "Instability in SPERT-I", Nucleonics 15, No. 1, 41 (1957).

18 NRL-4495 and NDA-154-1 (CLASSIFIED)

19 Burdick, E.E. and Henscheid, J.W., Reactivity Measurements in the ETR Critical Facility, IDO-16454, Phillips Petroleum Company, Idaho Falls, July 10, 1958.

20 Blomeke, J.O., Nuclear Properties of U-235 Fission Products, ORNL-1783, Union Carbide Nuclear Corporation, Oak Ridge

21 Nyer, W.E., et al., "Transient Experiments with the SPERT-I Reactor, Nucleonics 14, No. 6, 44 (1956).

22 Rockwell, T., Reactor Shielding Design Manual, McGraw-Hill, New York, 1956.

23 Morgan, K.Z., et al., Maximum Permissible Concentration of Radioisotopes in Air and Water for Short Period Exposure, Geneva Paper p/79, 1955.

END