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DESCRIPTION AND SAFETY ANALYSIS OF THE POOL CRITICAL ASSEMBLY

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ABSTRACT

The Pool Critical Assembly (PCA) is basically similar and in most respects identical to the natural-convection cooled, 1-Mw version of the Bulk Shielding Reactor (BSR). The description and analysis in this report, therefore, consists largely of material that was adapted from existing reports, ORNL-TM-2231, Description and Safety Analysis of Significant Change of the BSR for 2-Mw Operation, F. T. Binford, T. P. Hamrick, and L. E. Stanford, in particular.

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2. INTRODUCTION

2.1. Historical Background and Motivation

Prior to 1957, certain research personnel in the Neutron Physics Division and the Operations Division saw the need and utility of a versatile, simply designed critical facility that could be operated in the already-constructed Bulk Shielding Reactor (BSR) pool. The interest of the Neutron Physics personnel was motivated primarily by the advantages of this type of reactor in pursuing their various research projects. Since the Oak Ridge Research Reactor (ORR) had not been completed at that time, but was under construction, Operations Division personnel were interested in such a facility as a training device and for the purpose of studying such things as future core configurations for the ORR, shielding problems, new instrumentation, and reactor physics.

The Pool Critical Assembly (PCA) was completed in 1958 and has since proved its value as a training and research tool.

Prior to 1964, the PCA was operated by personnel of the Neutron Physics Division; in 1964, the Operations Division assumed responsibility for its operation.

2.2. Brief Description of the PCA

The Pool Critical Assembly is a light-water moderated and cooled, pool-type facility. The core, usually reflected on all sides by light water, is of the heterogeneous type and uses enriched uranium fuel in the form of aluminum-clad, aluminum-uranium alloy fuel plates.

The PCA is located near the northwest corner of the same pool in which the 2-Mw Bulk Shielding Reactor,[1] also operated by the Operations Division, is located; however, unlike the movable BSR, the support structure for the PCA core and control chamber guides is mounted on a plate anchored to the floor of the pool.

A loading platform is located above the core to facilitate access to certain instrumentation and the core. The PCA control room is located at the northwest side of the reactor bay (see Figures 1 and 2).

In addition to the lack of complexity of this reactor system, one of the unique features of the PCA is the versatility of the core's design,

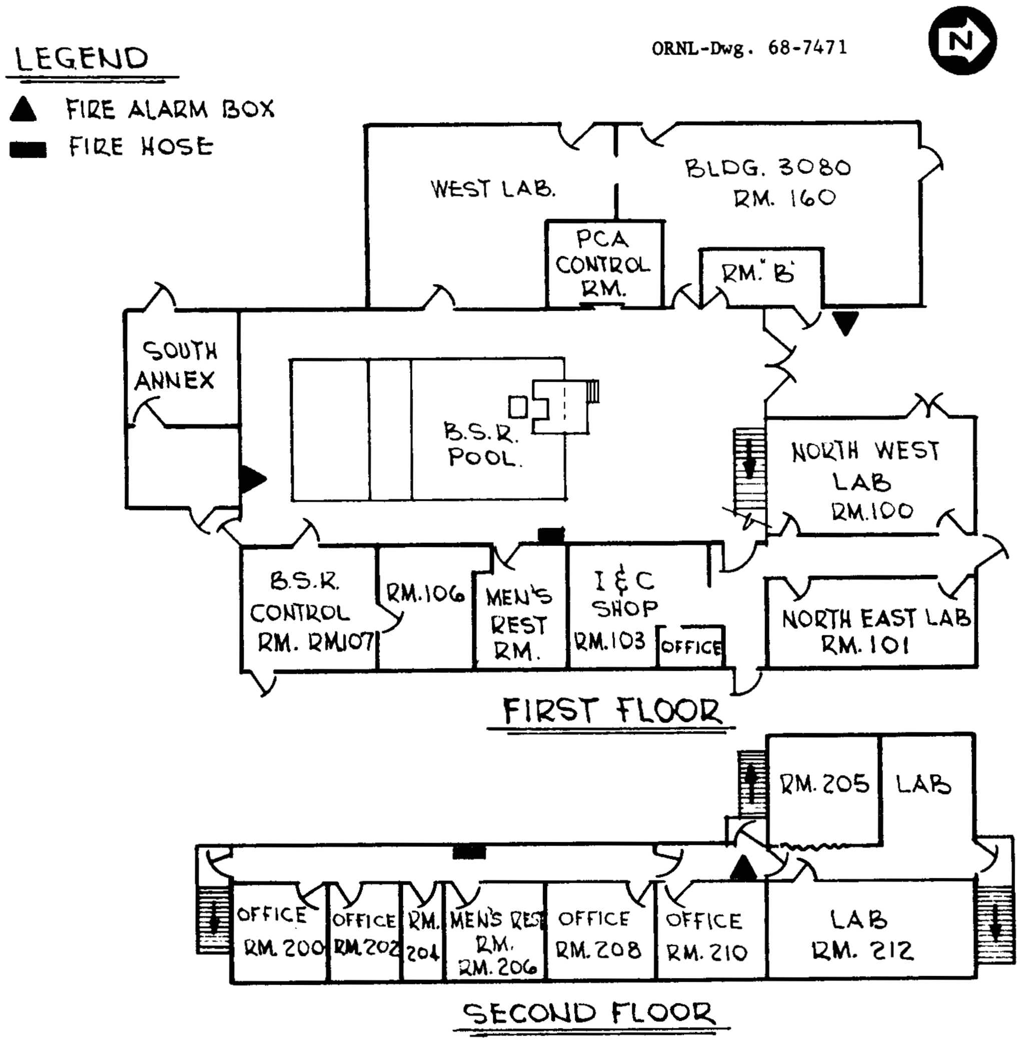

Fig. 1. BSF and Associated Experimenters' Laboratory Space

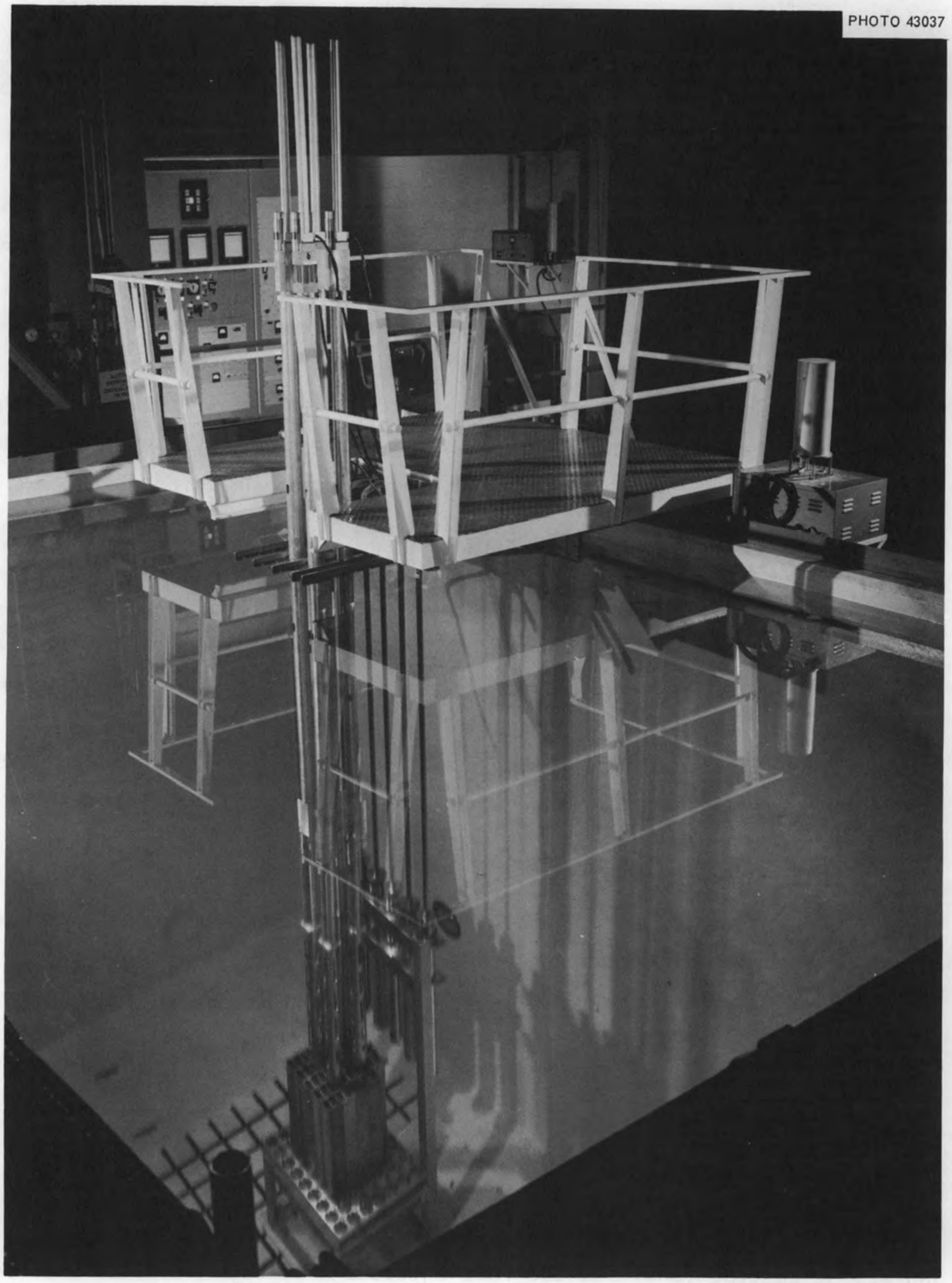

Fig. 2. The Pool Critical Facility with the Control Room Shown in the Background

which permits it to accept either the 18-plate BSR or 19-plate ORR fuel elements. (This is accomplished by means of "stacked" grid plates; the BSR-type grid plate, which has round holes to accommodate the BSR end boxes, may be aligned on top of the ORR-type grid plate, which has square holes to accommodate the ORR end boxes.)

Control of a core (consisting of either BSR or ORR fuel elements) is normally accomplished by means of three shim-safety rods and a regulating rod and the associated drive mechanisms. The rods are moved in the vertical direction within special control-rod fuel elements. The three shim-safety rods contain boron carbide as the neutron absorbing material, whereas the regulating rod is a shell of stainless steel (type 347). The three shim-safety rods are supported by electromagnets; the extension tubes on the magnets are, in turn, attached to lead screws each of which is equipped with an in-line drive motor. The regulating rod is attached directly to its drive tube (no clutch) and may be positioned either by manual operation or by the servo mechanism. The four special fuel elements through which these rods travel are identical, each containing about 70 g of ^{235}U in nine fuel-bearing plates. The special control-rod elements are positioned in the ORR-type grid plate by the use of adapters which support the rod elements at the correct height for the ORR fuel elements; these same rod elements are used without adapters in the BSR-type grid plate.

The PCA is operated at relatively low power levels (i.e., at or below 10 kw); consequently, the natural convection of the pool water is adequate to satisfy the cooling requirements. However, since the pool water is demineralized and circulated for the BSR requirements, a brief description of the water systems will be given in Section 4.

3. DESCRIPTION OF THE BULK SHIELDING FACILITY

3.1. Facility Site

The Bulk Shielding Facility (BSF), which is composed of the Bulk Shielding Reactor (BSR) and the Pool Critical Assembly (PCA), is located in Building 3010 in the north-central area of the main portion of the Oak Ridge National Laboratory (ORNL). Figure 3 shows its approximate location with respect to other Laboratory facilities and also presents a general plan of the Laboratory itself. As can be seen, Building 3010 is in the immediate vicinity of Building 3042, which houses the Oak Ridge Research Reactor (ORR), and Building 3005, which contains the Low-Intensity Testing Reactor (LITR).* The location of ORNL within the East Tennessee region is shown in Figures 4 and 5.

Data concerning the geophysics, geography, meteorology, and population distribution at the ORNL site are available elsewhere, notably in ORO 99, A Meteorological Survey of the Oak Ridge Area, published by the Weather Bureau, USAEC, and in ORNL 3572, The High Flux Isotope Reactor, A Functional Description, F. T. Binford and E. N. Cramer, editors.

3.2. Building and Pool

The PCA is located in the same building and in the same pool as the Bulk Shielding Reactor (BSR). These facilities are maintained as a normal requirement for the BSR; and, since the normal conditions are completely adequate for the PCA, requirements on these systems are not included in the PCA Procedures.

3.2.1. Building

The facility is housed in a steel frame building with Q (corrugated metal) siding and is shown in Figure 6. The building is 77 ft long x 51 ft wide, over-all, and houses the pool and reactor in a bay 77 ft long x 32 ft wide x 35 ft high. The remainder of the building contains offices, instrument rooms, experiment rooms, and a small shop. The floor plan is shown in Figure 1.

3.2.2. Pool

The reactor pool is of reinforced ordinary concrete construction resting, at least in part, on bedrock. Figure 7 shows plan and sectional views of the pool with the concrete filler blocks in place. The inner surfaces of the pool are coated with $\sim$0.030 in. of thermosetting plastic paint to improve water tightness, to aid in cleaning, to enhance visibility,

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*Deactivated in October, 1968.

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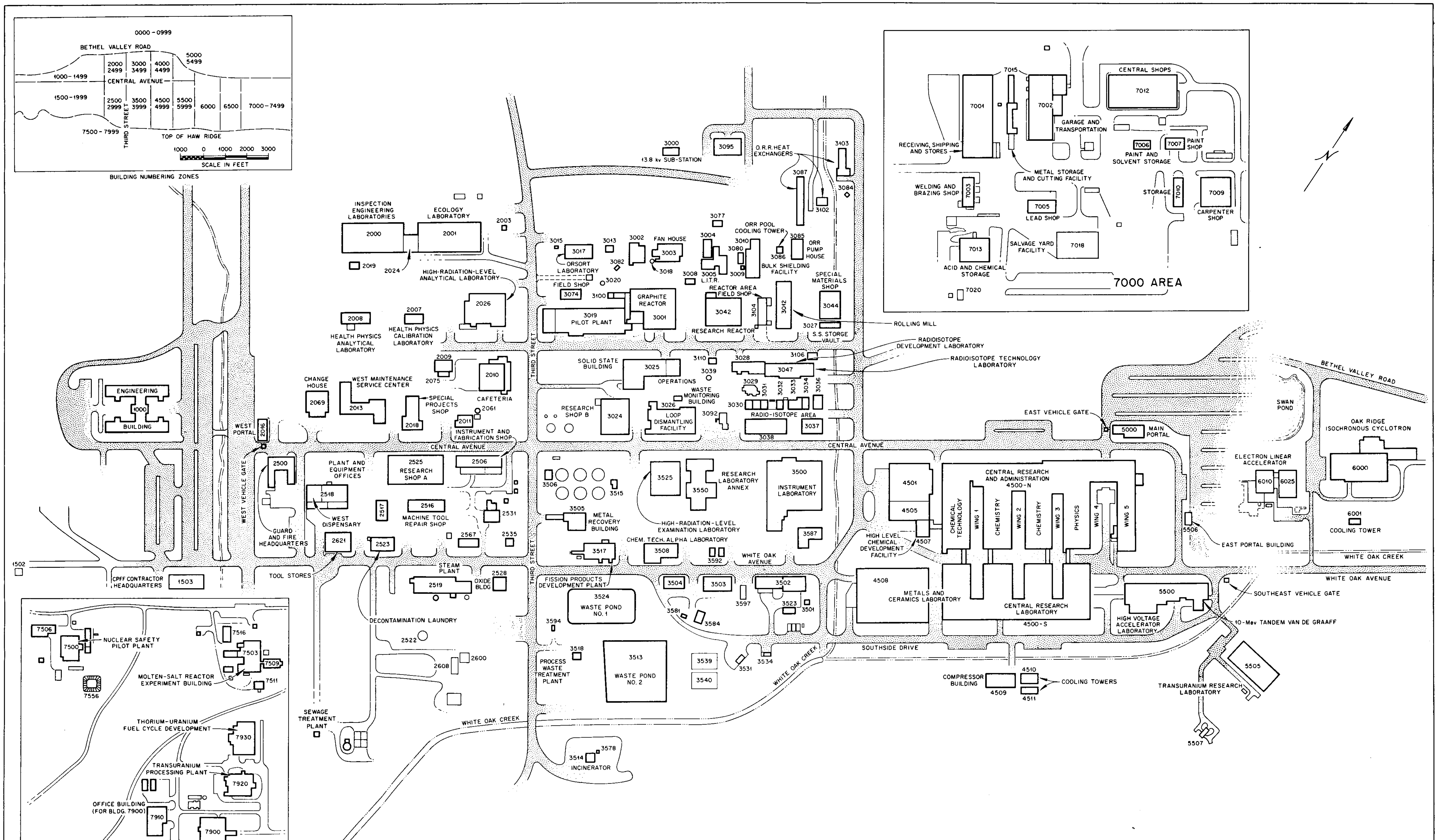

Fig. 3. Oak Ridge National Laboratory Permanent Facilities Plan

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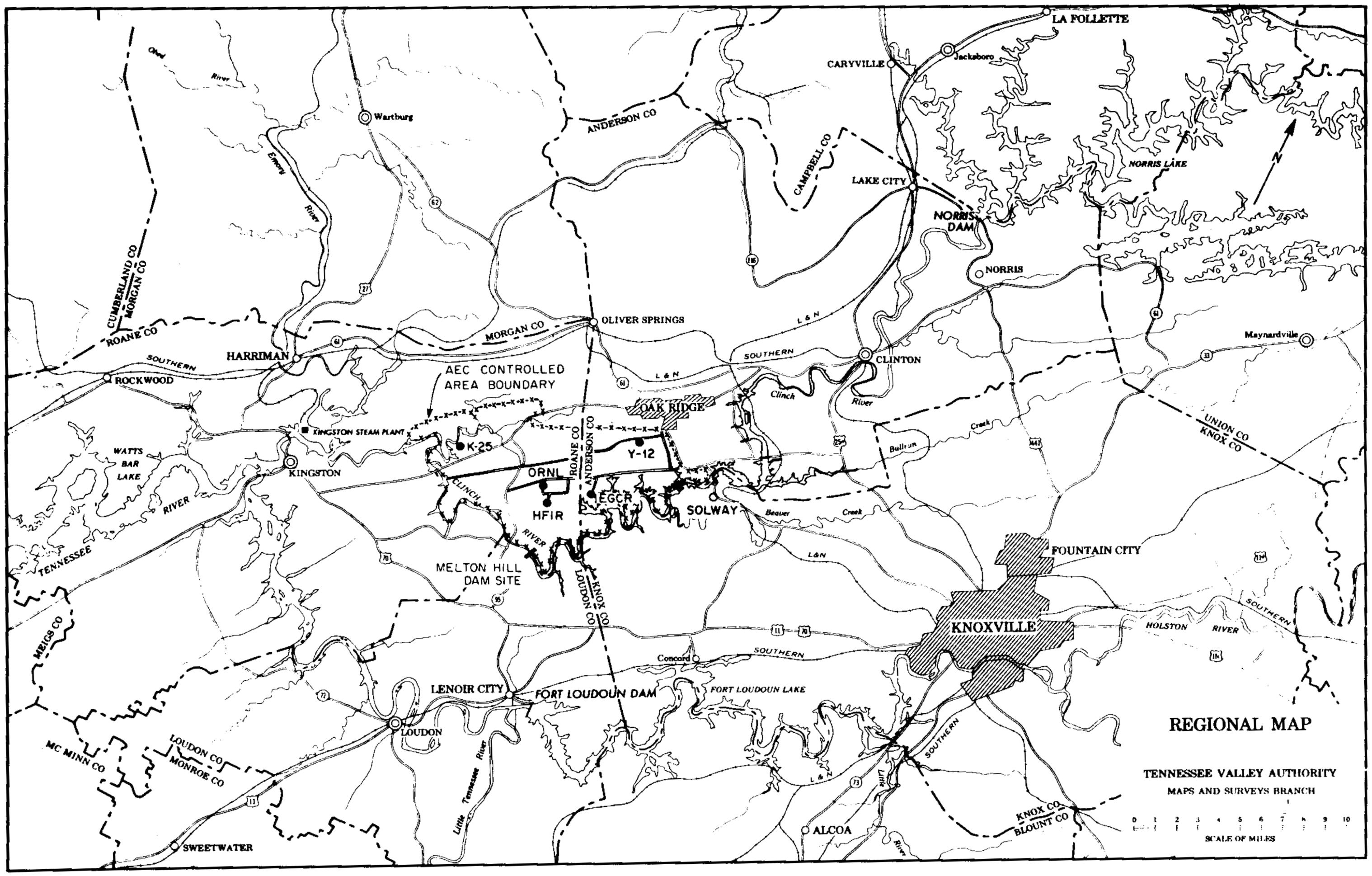

Fig. 4. Regional Map

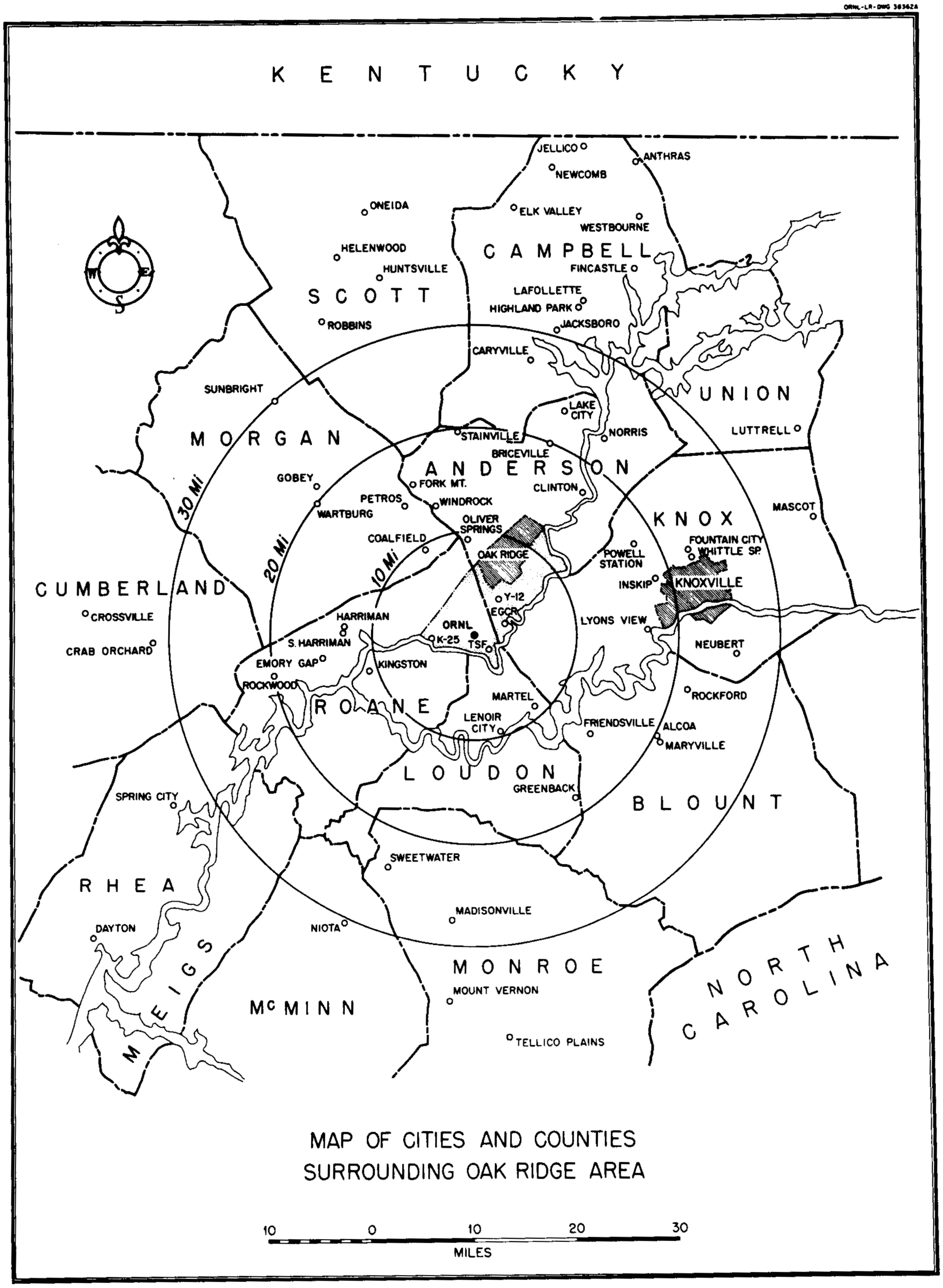

Fig. 5. Map of Cities and Counties Surrounding the Oak Ridge Area

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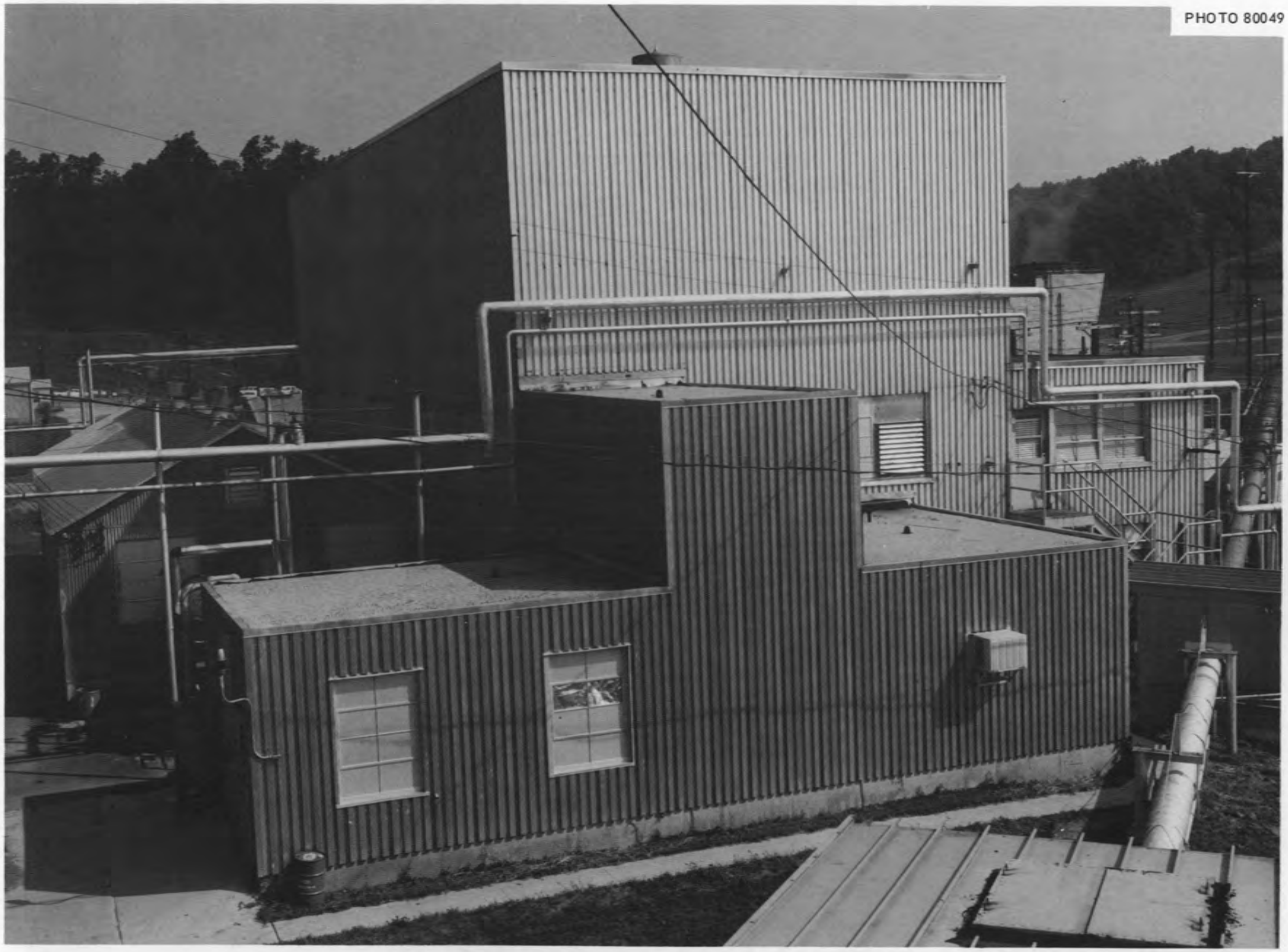

Fig. 6. The Bulk Shielding Facility

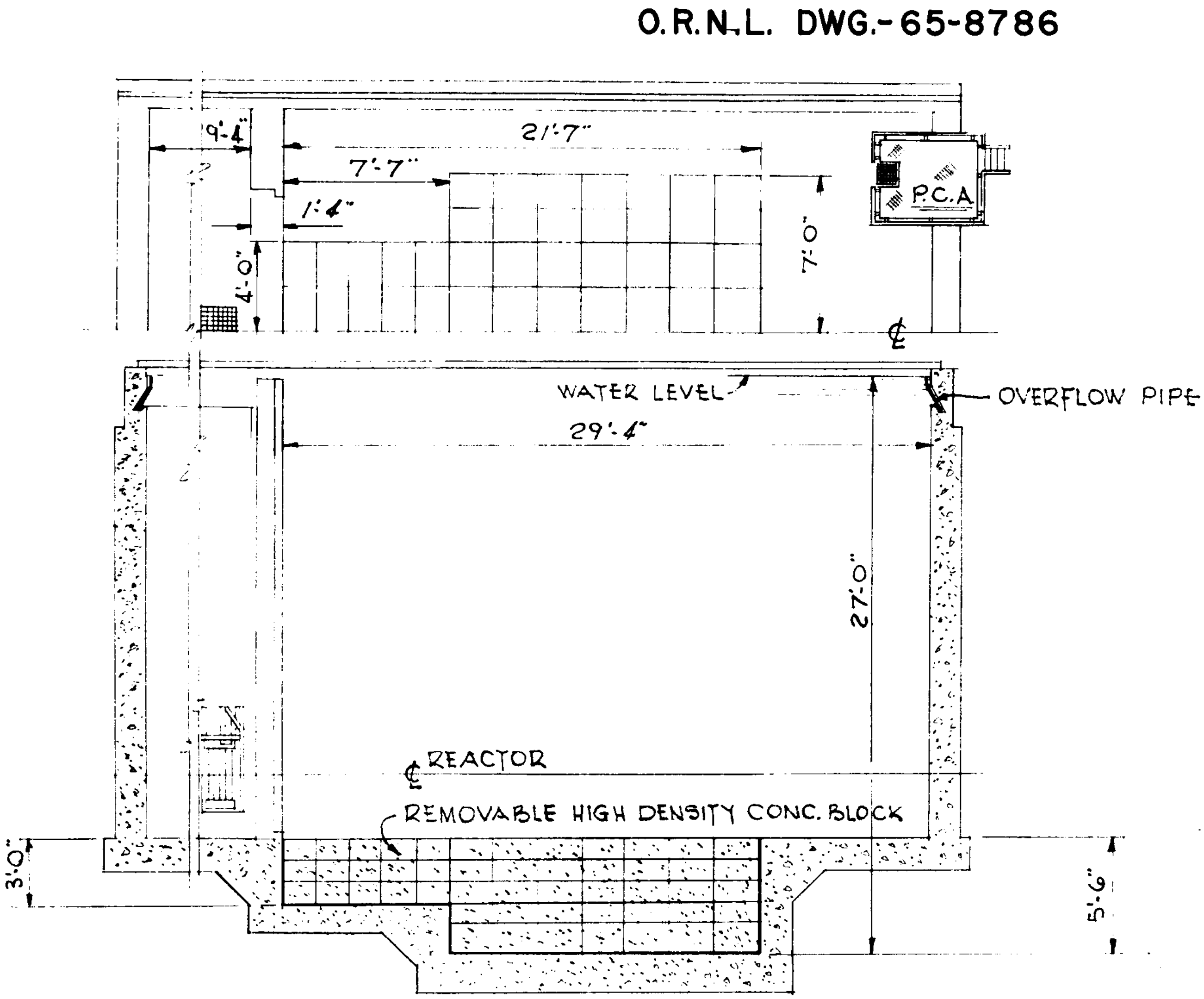

Fig. 7. Longitudinal Section of the BSR-PCA Pool

and to minimize corrosion of the reactor components by isolating the pool water from corrosive agents present in the concrete. The pool is 40 ft long x 20 ft wide and has a nominal depth of 21 1/2 ft; however, the pool floor contains a stepped pit so that a maximum water depth of 27 ft can be obtained (see Figure 7). The pool may be divided into two sections (a small section 9 ft 4 in. long and a large section 29 ft 4 in. long) by a removable dam. The stepped pit is in the large section and consists of a section having dimensions of 14 ft x 14 ft x 5 1/2 ft deep and a second section having dimensions of 7 ft 7 in. x 8 ft x 3 ft deep (depth measurements referred to the floor level at the 21 1/2 ft depth). The 14-ft-sq section may be filled with special concrete blocks to the 3-ft depth to form a single-level well, or both wells may be filled with special concrete blocks to provide a single-level floor when desired (see Figure 7). With no filler blocks in the pool, the pool volume is ∿138,000 gal (∿128,000 with all blocks in place). Steel rails are mounted on the east and west walls of the pool to accommodate a wheel-mounted bridge that spans the width of the pool to support the BSR. An additional similar bridge, called the instrument bridge, is used to provide a working platform and space for special equipment.

4. REACTORS

4.1. Brief Description of BSR

From 1950 to 1963 the Oak Ridge National Laboratory's Bulk Shielding Reactor (BSR) was used primarily as a neutron and gamma-ray source in radiation shielding studies. The 1-Mw natural-convection-cooled facility has been described in considerable detail previously (see references 1, 2, and 3). It consisted of a small enriched-fuel reactor (a nominal 5 x 6 array of MTR-type fuel elements) suspended in an open pool of light water, the pool and water system, and appropriate instruments and controls. The reactor structure was supported by a wheel-mounted carriage which, in turn, was supported by a wheel-mounted bridge that spanned the reactor pool. The carriage was movable along the length of the bridge, and the bridge was movable along the pool length.

In 1963 the use of the BSR was diverted to basic research, particularly in solid state physics; and in 1966 the Bulk Shielding Reactor and its ancillary facilities were revised to permit continuous operation at a power level of 2 Mw. This required the installation of a forced-convection cooling system containing a heat exchanger, cooling tower, decay tank, filter, and demineralizer units. Other revisions included increasing the worth of the control-rod complement and upgrading the containment system. The revised system still permits the degree of mobility of the reactor described above. Positive stops welded to the rails upon which the bridge moves ensures adequate minimum spacing between the BSR and PCA cores.

Operation of the BSR at power levels up to 1 Mw with natural convection or up to 2 Mw with forced-convection cooling is permitted by use of a hand-operated valve beneath the grid plate. When open, this valve permits natural convection of pool water upward through the core (see Figure 8). Under these conditions, the reactor may be operated in essentially the same manner as with the old 1-Mw system. When closed, it permits the establishment of full forced-convection flow (∿1,000 gpm) downward through the core, thus the reactor may be operated at power levels up to 2 Mw on a continuous basis if required. The filter and demineralizer system are continuously in operation, even if the BSR is shut down, in order to maintain the pool water purity within acceptable limits.

The BSR core is usually arranged to best accommodate the experiments and, therefore, may be varied considerably. It usually consists of a nominal 5 x 6 array of 18-plate MTR-type fuel elements and is light-water reflected on four sides and heavy-water reflected on two sides. (The D_2O is a consequence of experiment requirements and not for the express purpose of a reactor reflector. The control-rod complement may contain from four to six control rods depending upon the particular requirements. Fuel elements differ from those normally used in the PCA only in that they contain more fuel (∿190 gm ^{235}U in new fuel elements and ∿90 gm ^{235}U in control rod elements).

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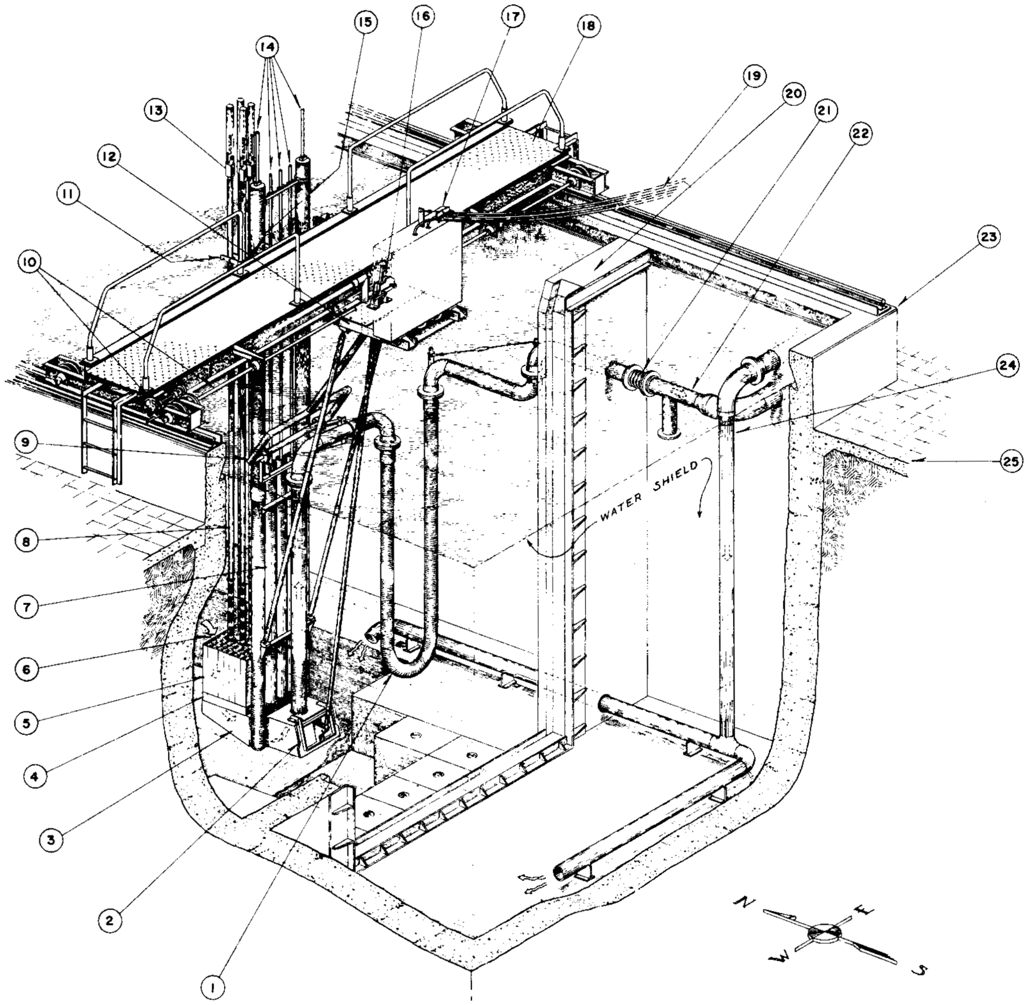

ISOMETRIC VIEW OF THE BSR Showing The Valve (item 2) Closed For Forced Convection Cooling. Arrows Indicate The Direction And Flow Path Of The Cooling Water.

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— LEGEND —

1. FLEXIBLE COOLING-WATER LINE
2. NATURAL-CONVECTION-COOLING INLET VALVE
3. WATER MANIFOLD AND REACTOR GRID SUPPORT
4. FUEL-ELEMENT GRID PLATE
5. FUEL ELEMENTS (REACTOR CORE)
6. REACTOR COOLING-WATER INLET
7. IONIZATION CHAMBERS AND GUIDES
8. SHIM-ROD-DRIVE GUIDE TUBES
9. WATER SPRAY SYSTEM FOR NITROGEN-16 DISPERSION
10. NORTH-SOUTH TRAVERSING DRIVE
11. SHIM-ROD-DRIVE LATCH BAR
12. REACTOR SUPPORT CARRIAGE
13. SHIM-ROD DRIVE MOTOR AND GEAR BOX
14. IONIZATION-CHAMBER SUPPORT TUBES
15. REACTOR SUPPORT TUBES
16. INLET-VALVE OPERATOR
17. INSTRUMENT AND CONTROLS JUNCTION BOX
18. REACTOR SUPPORT BRIDGE (MOBILE)
19. REACTOR INSTRUMENT AND CONTROL CABLES
20. DAM WITH GATE REMOVED
21. EXPANSION JOINT
22. REACTOR COOLING-WATER LINE TO HEAT EXCHANGER
23. POOL PARAPET
24. REACTOR COOLING-WATER RETURN LINE FROM HEAT EXCH.
25. FLOOR LEVEL OF BUILDING

Fig. 8. Isometric Drawing of the 2-Mw Forced-Convection-Cooled BSR

4.2. Pool Critical Assembly (PCA)

The PCA is a low-power critical assembly and is very similar in construction to the BSR. It is located in a fixed position in the northwest corner of the pool as shown in Figures 1 and 2. The principal difference between the facility and the former 1-Mw BSR design are that the grid plate is arranged to accommodate both ORR and BSR fuel elements and that the PCA is fixed in position rather than being mobile. The assembly is used primarily for performing special tests which do not require any appreciable power level and for training. The assembly is administratively limited to a maximum power of 10 kw. It is designed to duplicate the nuclear characteristics of the BSR and to provide properties similar to the ORR when the latter operates under conditions of low power and natural-convection water cooling. Positive stops (welded to the rails upon which the BSR bridge travels) prevent the BSR bridge from being moved close enough to the PCA to bring the two facility cores within range of significant interaction[4] even if the instrument bridge were removed.

The PCA is light-water cooled by natural convection, water moderated, and, usually, water reflected. It is fueled with aluminum-clad uranium-aluminum-alloy fuel elements similar to the MTR type. As shown in Figure 9, each standard element consists of 18 curved, composite-type fuel plates, each 3-in. wide, 24-in. long, and 0.060-in. thick. The complete standard assemblies currently used contain $\sim$140 g of ^{235}U although other weights are available for special tests. Taking into account the structural members, the metal-to-water ratio in the fuel region is about 0.7. The PCA normally uses BSR-type fuel elements which are equipped with single cylindrical end-boxes that fit into the grid plate and have no upper end boxes (see Figure 9).

The control-rod-receiving fuel elements are similar to the standard element except for slots provided for the entry of the control rods. These slots are obtained by omitting half of the fuel plates and inserting a rectangular tube with internal dimensions 1 1/8 x 2 5/8 in. The control rods have cross sectional dimensions of about 7/8 x 2 1/4 in. and are moved inside the channels provided within the special fuel elements (see Figure 10). The details of the rods are discussed in Section 4.3.

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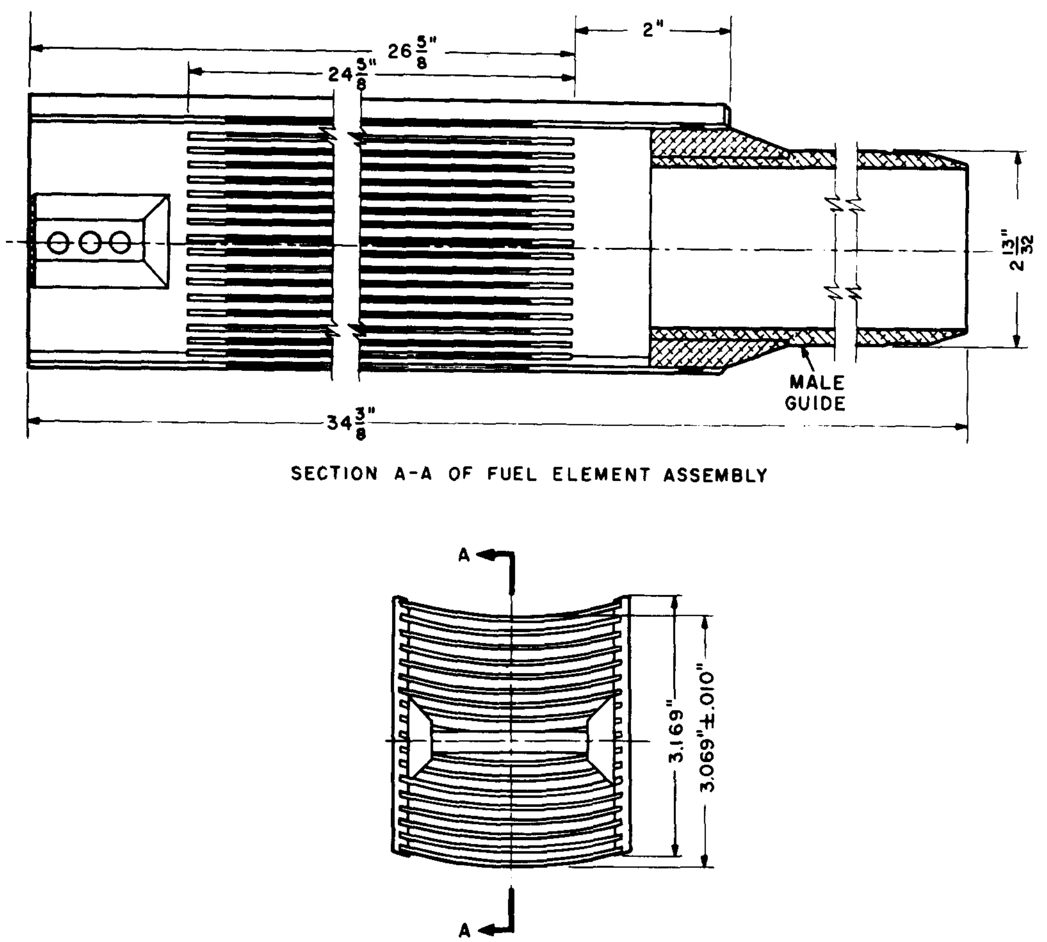

Fig. 9. PCA Fuel Element

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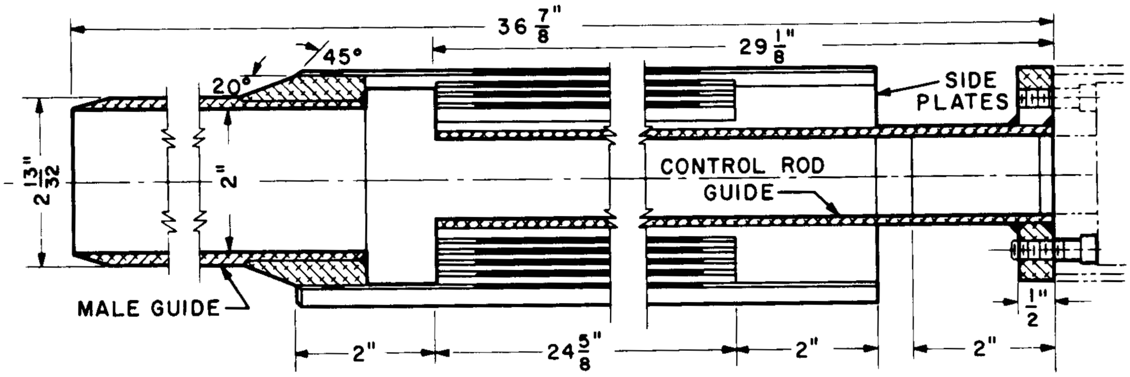
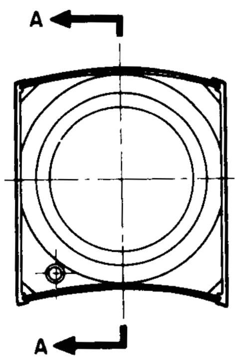
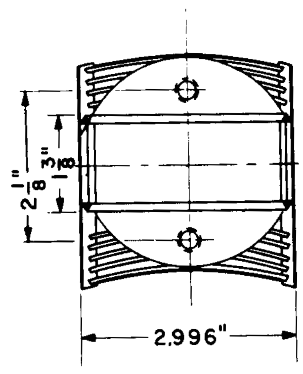


SECTION A-A OF SPECIAL FUEL ELEMENT ASSEMBLY



BOTTOM VIEW



TOP VIEW

Fig. 10. Special Fuel Element for PCA Control Rod

Fuel and other elements rest in an aluminum grid with dimensions of 28 x 25 x 5 in. deep. A total of 63 holes permits loading of any array up to 9 elements by 7 elements. Actually, criticality is usually attained using considerably fewer elements, but this grid design permits flexibility in the loading pattern. Figure 11 shows details of the grid structure and the general structural arrangement of the reactor.

With the aid of special tools, fuel elements and other core pieces can be installed or removed from the grid by manipulation from the working platform. The fuel elements can be handled under water at all times, thus protecting personnel from radiation if irradiated BSR or ORR elements are used. When not in use, the fuel elements are stored in a "criticality safe" fuel rack located under water or in the fuel storage vault.

The reflector normally consists of pool water adjacent to the core; however, other simple arrangements are possible. For example, dummy fuel elements can be filled with graphite, beryllium oxide, or other material, as desired, and inserted into the grid around or within the core proper. Or, as was done during initial operation of the BSR, a large slab of reflector material can be placed along one or more sides of the core assembly.

4.3. PCA Control Rods and Drives

The PCA is normally operated with three shim-safety rods, each of which consists of boron carbide canned in aluminum and a hollow stainless steel regulating rod which has the same external dimensions as the shim rods. The rods are oval shaped, having dimensions of 0.875 in. thick x 2.250 in. wide x 24 in. long. They have a 24-in. vertical travel and move within special fuel elements. Details of the special fuel elements are shown in Figure 10. These elements are flanged to guide tubes that extend several feet above the working platform and are bracketed to that structure (see Figure 11). The guide tubes serve a dual purpose in that, in addition to guiding the control rods and control-rod-drive tubes, they hold the special fuel elements in place while withdrawing the control rods. An in-line drive unit consisting of an electric motor, position transmitters, limit switches, etc., is attached to the top of the guide tube of each rod. Details of the rod-drive mechanism are shown in Figure 12.

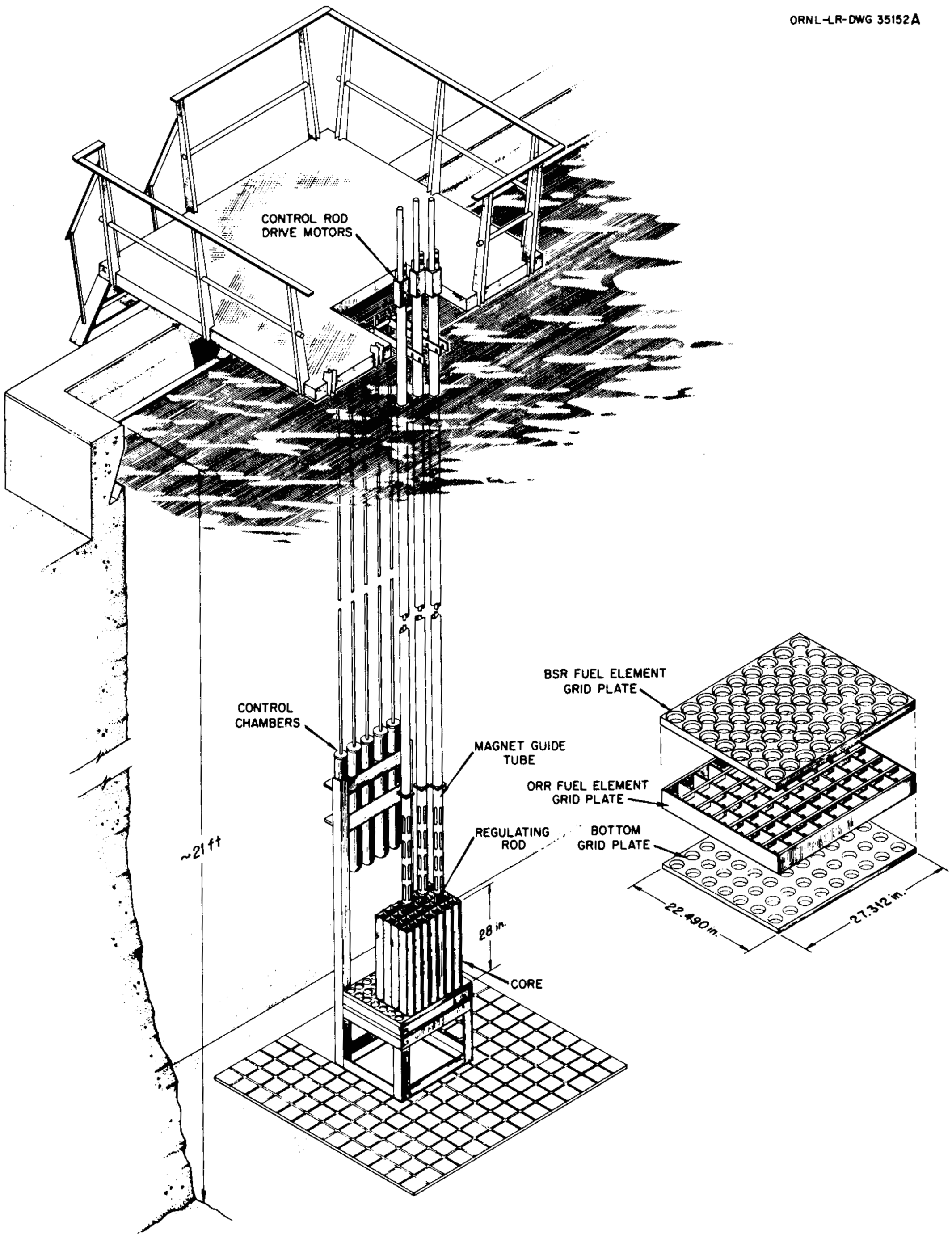


Fig. 11. Structural Arrangement of the Pool Critical Assembly

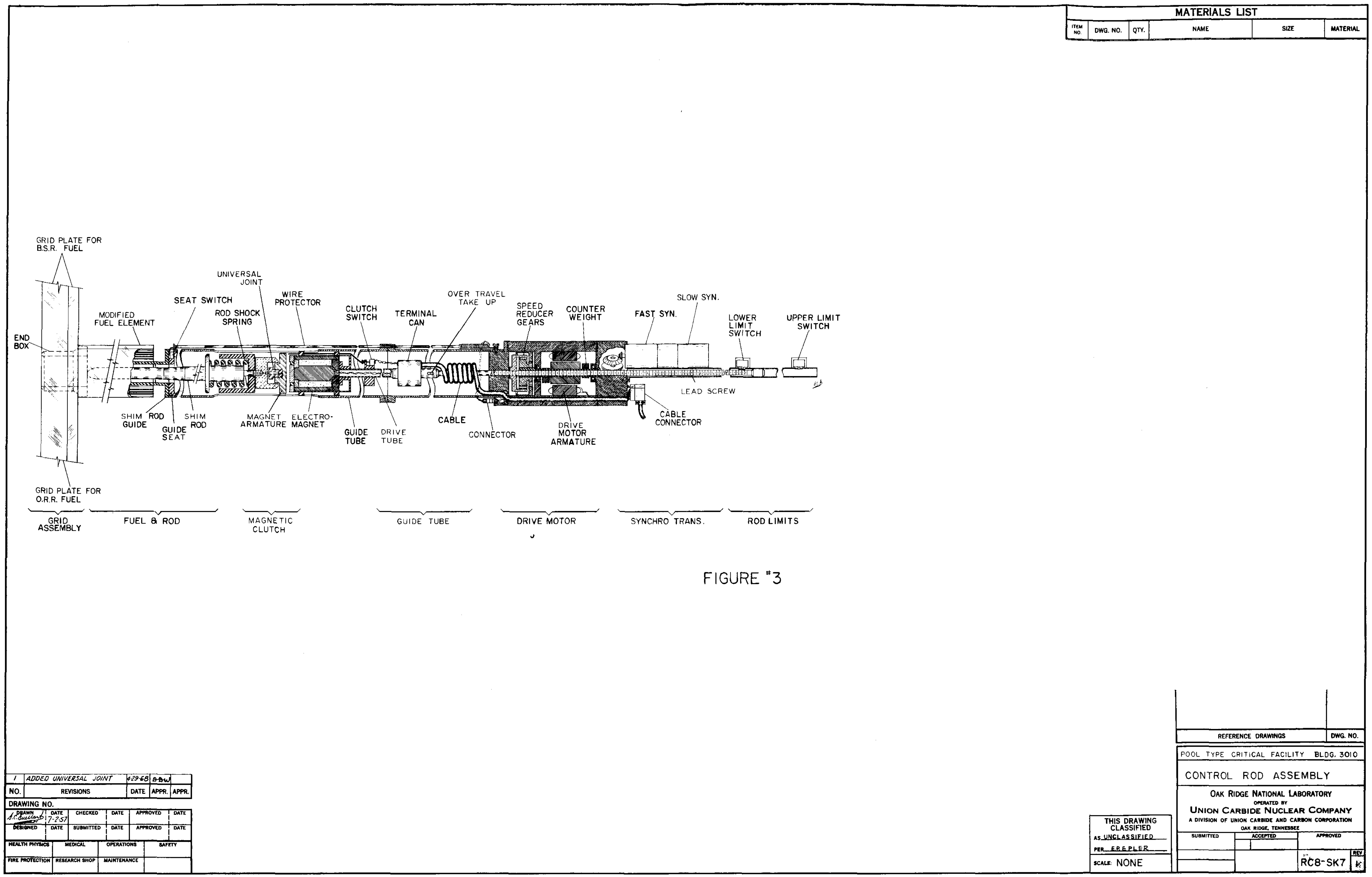


Fig. 12. BSR-PCA Rod Drive Mechanism

The motors drive lead screws which are mechanically attached to the drive tubes which move within the guide tubes. For the shim rods, the lower end of each guide tube contains a shock-absorber cup (not shown), a supporting seat for the top of the rod, and mechanically operated seat switches. The top of each shim rod is equipped with a magnet armature and a piston-and-spring arrangement to provide shock-absorber action. The lower end of each shim-rod-drive tube is equipped with an electromagnet which, for reactor operation, is mated to the magnet armature (attached to the top of the shim rod) and energized by a magnet current supply amplifier system. The regulating rod is connected mechanically to the drive tube and, therefore, does not need the shock absorber, seat and clutch switches, or magnet. The total measured reactivity worth of each individual shim rod changes from 2 to 3% $\Delta k/k$, depending upon the particular core loading and reflector. The measured total worth of the three shim rods is normally about 7% $\Delta k/k$ in a 28- or 30-element core (using 140-g elements, water reflected) in a 5 x 6 array. If the higher value of 3% $\Delta k/k$ is assigned for each rod, the linear portion of each rod is worth about 0.21% $\Delta k/k$ per in.; therefore, with a maximum shim-rod-drive speed of about 12 in./min, the maximum reactivity addition rate capability could not possibly exceed 0.042% $\Delta k/k$ per sec for individual rods and 0.126% $\Delta k/k$ per sec for the three shim rods and is actually about 0.1% $\Delta k/k$.

The regulating rod has a motor-driven withdrawal speed of about 1 in./sec (total 24-in. withdrawal in 23 sec) and its worth at most is $\sim$0.70% $\Delta k/k$. The differential worth of the linear portion of the rod (at 0.7% $\Delta k/k$ total) would be about 0.05% $\Delta k/k$ per in. Therefore, the maximum rate at which it can add reactivity to the core is about 0.05% $\Delta k/k$ per sec. The reactivity worth of the regulating rod is always limited to less than one dollar in order to preclude the possibility of a prompt critical condition resulting from failure of the servo control system. Such a condition might otherwise develop since the full value of the regulating rod is at the disposal of the servo control unit. Normally, the full value of the regulating rod is limited to about 0.5% $\Delta k/k$.

4.4. Reactivity Requirements

In general, operation of the PCA requires little excess reactivity. This is because of the low maximum power (resulting in little fuel element or water temperature rise and fission-product-poison buildup) and the nature of the experiments normally performed.

In general, therefore, reactor core loadings are designed to provide a practical minimum excess reactivity for particular operations. In all cases, however, core loadings are in accordance with the Operating Safety Limits for the Pool Critical Assembly (Appendix C).

4.5. Control and Protection Systems

A description of the control (nonsafety) system is given in Appendix A.

The type of protection system used at the PCA has been used at many ORNL reactors and elsewhere,[5,6,7] and its characteristics are well known. A description of it is given in Appendix B.

4.6. Parameters Requiring PCA Control and Safety Action

A. Control (Nonsafety) System Actions

1. Slow Scram
   a. Manual
      (1) Activation of local scram switch at control panel.
      (2) Key switch turned to "Off" position.
      (3) Raise clutch mode switch turned to "Raise".
   b. Radiation - signal from personnel protection radiation monitor.
   c. Loss of control power.
2. Fast Scram: Log-N channel - Reactor period <1 sec positive
3. Reverse
   a. Manual operation of group rod insert switch.
   b. Power level detected by log N above servo demand setting.
   c. Power level >12 kw.
   d. Log-N period <7 sec.

e. Count-rate period <7 sec and the power level is below 3 N_L ($N_L \approx 100$ watts for PCA).

f. Safety trouble monitor detection of failure to danger in two power level safety channels or one level safety and the log-N channel.

g. Power level safety readout (any of 3) above setpoint.

B. Protection System Actions

1. Slow scram: Loss of magnet amplifier power.

2. Fast Scram

a. Power level safety system.

(1) No. 1 channel - power $\gtrsim 1.5\ N_F$.

(2) No. 2 channel - power $\gtrsim 1.5\ N_F$.

(3) No. 3 channel - power $\gtrsim 1.5\ N_F$.

C. Principal Annunciators

1. Radiation

a. Radiation level too high as detected by pool area or control room onitrons.

b. Monitron out of service.

2. Reactor System

a. Log-N reverse.

b. Log-N period <7 sec.

c. Count-rate period <7 sec.

d. Any reverse.

e. Slow scram.

f. Fast scram.

g. Safety instrumentation trouble (one failure).

h. Safety instrumentation trouble (two failures).

i. Shim request.

All in-reactor experiments are subjected to thorough and detailed safety evaluations by the Operations Division technical staff. Possible reactivity effects are always considered in the reviews and are measured prior to operation of an experiment in the reactor. No experiments are approved that could credibly cause reactivity changes that cannot be safely handled by the reactor control system.

5. ORGANIZATION AND ADMINISTRATION

The Pool Critical Assembly is under the management of the Operations Division of the Oak Ridge National Laboratory, which also has the responsibility for the operation of three other reactors, the Bulk Shielding Reactor, the Oak Ridge Research Reactor, and the High Flux Isotope Reactor. An organization chart of this division is shown in Figure 13.

The qualifications required of the operating and technical personnel of the Operations Division have been described in detail elsewhere.[8] Also described there are the support and review organizations which are available to the division.

The Pool Critical Facility is, like all other ORNL reactors, operated only by qualified personnel utilizing written procedures. However, because the PCA is normally used in a nonroutine way, its operation is generally permitted only when under the close supervision of a qualified reactor engineer.

5.1. Operating Procedures

To minimize the possibility of any type of reactor incident, operation of the PCA is governed by carefully prepared, written, standard procedures.[9] These procedures are designed to ensure that the operation of the reactor is carried on in a safe, well-regulated manner. The operating procedures describe in detail the steps required for all routine operations and for as many nonroutine operations as can be anticipated.

In addition to step-by-step detail, the procedures supply information concerning the need for the particular method of operation, special hazards which may be encountered, and references to various types of descriptive material such as blueprints or component operating manuals. The operating manual for the PCA contains material covering the following subjects:

a. Startups
b. Steady-state power operation
c. Shutdowns
d. Instrumentation and controls
e. Storage, refueling, and pool work

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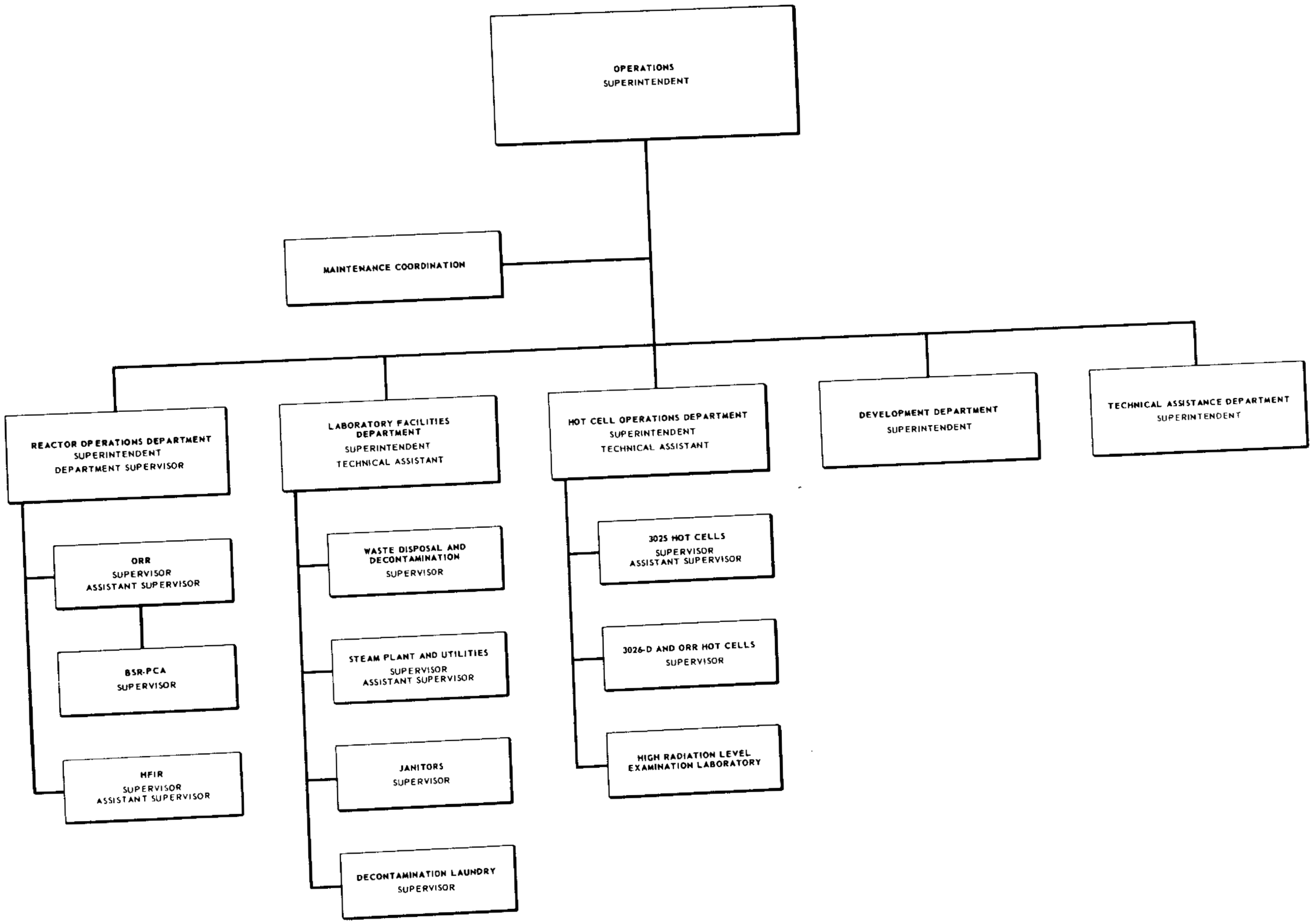

Fig. 13. Operations Division Organization Chart

f. In-core work

g. Replacement of components

h. Replacement of experiments

i. Research

j. Reactor primary coolant system

k. Reactor cleanup system

l. Maintenance

m. Emergency procedures and evacuation

n. Radioactive waste systems

o. Radiation safety and control

p. Records and data accumulation

The operating procedures are written by Operations Division personnel and are carefully reviewed and approved by senior staff members of the Division. All procedures are numbered and maintained in books or procedure manuals for ready reference by the operating personnel. As procedure revisions become desirable or as the necessity for new procedures arises, these are prepared by Operations Division personnel; and, after review and acceptance by appropriately designated specialists and by the Superintendent of the Reactor Operations Department, they are then made part of the procedure manual.

In some cases in which the operation is quite complex or in which errors cannot be tolerated, the procedure requires that a checklist be used. Most of these checklists are completed by the engineer in charge and are reviewed by the reactor supervisor. A few examples of the operations with which checklists are used are reactor startup, reactor shutdown, and major maintenance operations.

Temporary procedures are required when nonroutine operations or experiments are performed with the reactor. Such procedures are prepared in advance and approved, as in the case of new or revised procedures. During shutdowns many operations may be performed and, in such cases, a temporary or shutdown procedure is written in advance to ensure that no work is forgotten and that all standard procedures are followed.

Emergency procedures are provided for those types of malfunctions which can be anticipated. These include methods of coping with contamination or radiation incidents, fires, loss of electrical power, loss of ventilation, and instrument malfunction. Closely associated with this is the Laboratory-wide emergency plan[10] which details the action to be taken in case of a serious emergency.

Communication from shift to shift or day to day is augmented by means of the PCA log book in which the details of the work of the shift are recorded and which is, therefore, a history of the operation. In general, the information contained in the log book can be summarized under the following headings:

1. Operations
2. Shutdowns
3. Trouble
4. Maintenance
5. Service to research
6. Routine checks
7. Experiment description
8. Miscellaneous

The strip charts from the various reactor instruments serve to supplement this information.

In cases in which it is practical, procedures are written to describe the various maintenance operations. For example, there are procedures for the standard routine maintenance of the instrument and control complement. In critical cases, instrument test procedures are supplemented by the use of a checklist which may be considered a part of the operating procedure. In addition, operating parameters are confined within limits which have been judged adequate to ensure the safety of the reactor. The operating safety limits for the PCA are reproduced in Appendix C.

5.2. Safety Reviews

In addition to the intradepartmental safety reviews, independent safety surveillance is provided by a number of safety-oriented committees which report to the Laboratory Director. These committees are composed of senior members of the ORNL staff selected for their competence in the particular field but, in general, not directly associated with the projects they review.

5.2.1. Reactor Operations Review

The Reactor Operations Review Committee performs an independent annual safety review of all the Laboratory's operating reactors. In the course of this annual review, the Committee examines operating reports published by the reactor operating groups, which include such operational data as power levels, shutdown experience, and an analysis of unusual occurrences. Consideration is given by the Committee to the condition and usage of operating procedures, the facility maintenance program, operating personnel changes, operator training programs, and mechanical and electrical changes to the reactor system. Each member of the Committee is assigned a continuing responsibility for keeping up to date on the operating history, major design changes, and safety status of a particular Laboratory reactor. When an annual inspection is made, the cognizant Committee member and two other Laboratory staff members who are not associated with the RORC or the reactor operating organization constitute a subcommittee which inspects the reactor facility. This subcommittee meets with the reactor operating staff as often as its members deem desirable to familiarize themselves with the design, operation, and other details of the facility. The subcommittee observes startup, reloading, and shutdown procedures and examines facility log books, operating reports, facility drawings, etc. The subcommittee submits a written report to the RORC presenting its findings and suggesting areas for discussion at the annual review with the reactor operators. As a result of the review, specific recommendations may be made to Laboratory management by the Committee.

In addition to its recurrent facility-inspection duties, the RORC is sometimes called upon in other situations in which ORNL management desires an independent, safety-oriented opinion on Laboratory reactor policy. Such instances include reviews of ORNL reactor safety analysis documents prior to their dissemination from the Laboratory; evaluating the significance of changes in local reactor operating policies; and, in special cases, joint evaluation, together with the Reactor Experiment Review Committee, of the effect of experiment assemblies inserted into the reactor.

While most of the Committee's tasks originate from its continuing inspection function or from management requests, it may also initiate inquiries as a result of questions put before it by reactor operating groups, requests from Safety and Radiation Control, or its own concerns on any aspect of local reactor operations.

5.2.2. Reactor Experiment Review

The Reactor Experiments Review Committee (RERC) reviews any new or unusual experiments proposed for insertion in the Laboratory's reactors. The experiments reviewed are generally of the in-pile type where credible failure or malfunction of the experiment cannot create a positive change in reactivity greater than the reactor protective system was designed to accommodate. If a credible experiment failure could create a serious reactor transient, it will also be reviewed prior to approval by the Reactor Operations Review Committee (RORC). Experiments are reviewed by the RERC from the standpoint of personnel and equipment safety and continuity of reactor operations. Appropriate limits are placed upon any materials, systems, components, effluents, or operations that may present a hazard to personnel or to the reactor. Experiments proposed for reactor operation are first carefully examined for safety by the reactor operating group. The experiment may be operated without further review if it is similar to experiments previously reviewed and approved by either RERC or RORC. If the experiment is of a new or different type or unusual potential hazards exist, it is submitted, with the recommendations of the reactor operating group, to the RERC. When the Committee concurs with the operators that the experiment may be safely operated, the experiment may be inserted in the reactor. The Committee may make recommendations or establish conditions on the design, construction, and operation of the experiment.

In addition to examining new experiments, the Committee periodically reviews all the experiments in the reactor to ensure that they are being operated safely. The Committee also has the prerogative of requiring additional review of any experiment if it deems this necessary.

5.2.3. Criticality Review

The Criticality Committee has review and approval jurisdiction over operations which involve the handling, storage, transportation, and disposal of significant quantities of fissile material. The fissile materials include the isotopes ^{235}U, ^{233}U, Pu, and the combined elements of americium and curium. Approval for operations with significant quantities of the above materials must be obtained in advance on a Nuclear Safety Review form submitted to the Committee. This form is initiated by the requester of the operation, approved by his divisional Radiation Control Officer, and finally approved for a limited period of time by the Criticality Committee.

Disposal of fissile material must be in accordance with the procedures in the Health Physics Manual with the approval of the Committee.

Reactor fuel within a reactor core is the responsibility of the Reactor Operations Review Committee; however, procedures for storage and handling of fuel before insertion and after removal must be approved by the Criticality Committee. The Committee acts in many respects as a consulting group and gives assistance in problems involving criticality. It also conducts an annual review of each facility or balance area possessing significant amounts of fissile material to ensure that approved procedures are being followed.

5.2.4. Waste Disposal

The Radioactive Operations Committee reviews Laboratory facilities handling or processing significant quantities of radioactive materials and the practices used in disposal of radioactive solid, liquid, and gaseous waste. The Committee is particularly concerned with ensuring containment, the completeness and accuracy of the operators' safety analysis, detailed operating procedures, and the possibilities of interactions, either chemical, mechanical, or procedural, which might lead to unplanned exposure or contamination.

All new radiochemical facilities or processes are reviewed prior to operation; existing facilities are reviewed whenever changes in purpose or scope are proposed. The more important facilities are reviewed by the full Committee at intervals of one to three years even though no changes of purpose or scope have been made. Frequency is dependent on the magnitude of the operation and the hazard involved.

The membership of the Radioactive Operations Committee is chosen from senior Laboratory personnel experienced in various of the following disciplines: health physics, engineering (mechanical, chemical, or electrical), chemistry, and instrumentation.

5.2.5. Safety and Radiation Control

The Laboratory has established, as a staff function of the Director's office, a Safety and Radiation Control Department. The responsibility of this organization is to establish, on behalf of Laboratory management, policies with respect to radiation protection and to ascertain that this policy is met at all times. It promulgates criteria, for example, for facility containment and provides a liaison function between the various Laboratory divisions. Staff members of the Safety and Radiation Control Department are assigned responsibilities for following closely the activities of those Laboratory divisions which handle significant quantities of radioactive materials. Specialists in key elements of the radiation safety program, such as containment, waste disposal, criticality, reactor safety, etc., are available to the staff of the Director of Safety and Radiation Control.

6. REACTOR CONTAINMENT SYSTEM

The Bulk Shielding Facility is equipped with a containment system similar in principle to that for the ORR[11] (see Figure 14). Basically, this consists of: (1) the building, which is reasonably leak tight; (2) a building exhaust system (including particle filters and iodine absorbers); (3) an air-conditioning system; and (4) adequate instruments and control devices to effect "containment conditions" automatically upon the detection of predetermined radiation levels either in the building or in the building exhaust duct. The control features include provisions for manual override of the automatic devices so that containment conditions can be obtained even if the radiation monitors do not detect high levels of radioactivity. The building ventilation is adjusted so that under

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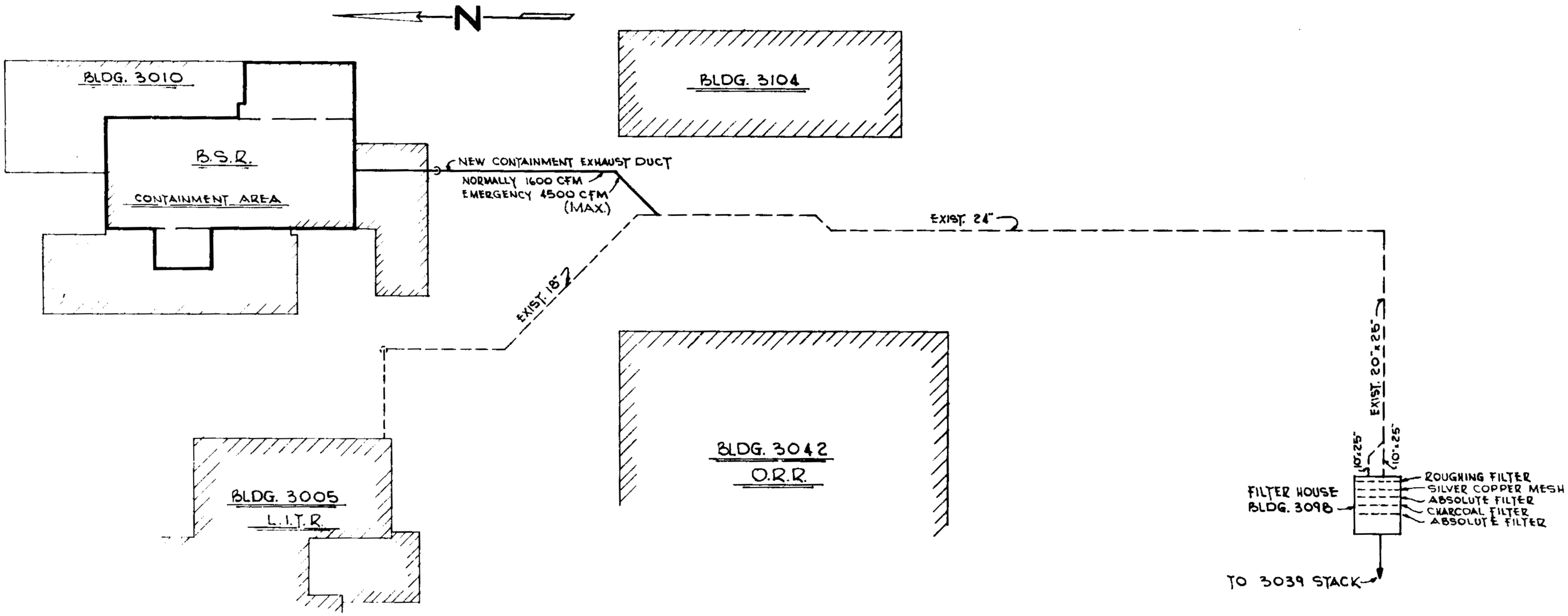

Fig. 14. Sketch of the Containment System of the Bulk Shielding Facility

containment conditions the building air leakage is inward. The building exhaust air is adequately filtered prior to its release to the atmosphere through a 250-ft-high stack. Although not required for the PCA, the containment system must be functioning properly when the BSR is operated at 2 Mw.

7. NORMAL OPERATIONAL HAZARDS AND SAFEGUARDS

7.1. Reactivity Considerations

There are several potential ways in which undesirable changes in reactivity could occur. The following is a discussion of the most important of these along with certain inherent and engineered safeguards.

7.1.1. Failure of Control (Nonsafety) and Protection (Safety) Systems

The reactor system is thoroughly checked in accordance with PCA instrument check-out procedures.[9] In addition, operational checks, in accordance with reactor startup procedures, are made prior to any reactor operation. However, failure of the control system is credible; and, under certain conditions of failure, withdrawal of the shim rods at their maximum speeds could conceivably occur either manually or automatically. It is emphasized, however, that no single failure could result in these conditions; thus, the probability of such an event is quite low.

The PCA protection and control systems are designed in accordance with a principle originally established by Newson[12] for the MTR and which has been applied to all ORNL reactors. This principle assumes a startup accident, originating at source level, brought about by the catastrophic failure of the startup instrumentation, the control system including interlocks, and manual operation. This failure is postulated to leave the reactor supercritical with the shim-safety rods withdrawing simultaneously at their maximum rate. Except for intrinsic shutdown mechanisms, the reactor protection system remains the only means of stopping the reactivity addition and of turning the excursion by releasing the shim-safety rods. The minimum required performance of the protection system is, therefore, the protection of the reactor core from catastrophic failure of the startup instrumentation and control (nonsafety) system.

This protection system[5,6,7] has been used in many ORNL reactors and elsewhere, and its characteristics are well known. Usually, as in the PCA, three completely independent neutron level channels are so arranged that action of any one of three can scram the shim rods if the monitored variable exceeds predetermined values. The time response from first exceeding the preset limits to first motion of the shim-safety rods (release time) is maintained at approximately 10 msec. This includes magnet flux decay time starting with a magnet current sufficient to support twice the rod weight. Extensive operating experience has been obtained with the PCA system, as well as with identical systems in other reactors.

The BSR II (Ref. 13) system, with electronic components identical to those in the PCA, has been tested in the SPERT facility.[7] As far as can be determined, this is the only protection system which has been required to intervene in a deliberately initiated series of excursions wherein a failure to respond would have resulted in unacceptable damage to the core.

The system under test consisted of two power-level safety channels, two period safety channels, and four shim-safety rods. The BSR II rods were given an initial acceleration of 6.5 g thus providing somewhat better performance than that obtainable in the PCA system. Reactivity was added at source level at the average rate of $20/sec with the following results:

Test	Shortest Period	Result
Self shutdown only	14 msec	Peak at 226 Mw; slight damage to the core
Power-level safety trip at 100 kw	4.6 msec	Peak at 92 Mw; no damage
Period trip set at 1 sec; sourceless start	3.0 msec	Peak at 79 Mw; no damage

The existence of a protection system of proven high performance reduces to a minimum the consequence of failure of the control system.

The startup instrument and reactivity control system is independent of the power-level safety system and, with certain reservations regarding the possibility of a sourceless start, has no safety function. The control system does, however, include features intended to reduce the number of scrams that would otherwise result from control system failure. As an example, the log N and power-level servo are given identical instructions. Should the power-level become appreciably greater than the demand level, the log N will call for rod insertion and thereby avoid an excursion requiring safety action. In a similar manner a signal from each power-level safety channel can call for rod insertion should the power-level safety level trip point drift or otherwise become too close to the operating point.

During the 13 years of operating experience, the PCA control system has not failed so as to require intervention of the protection system. Also during 13 years of PCA operation, the protection system has not once been incapable of protecting the reactor should the control system have allowed an excursion to develop. It is, therefore, established that the probability of failure of each of the three safety channels is also small. The probability of simultaneous failure of the independent protection and control systems becomes the product of four small probabilities such that the probability of an uncontrolled excursion by this mechanism becomes vanishingly small.

7.1.2. Maximum Rate of Reactivity Addition by Shim Rods

The maximum rate at which reactivity can be added to the reactor by use of the shim rods is limited to approximately the same value as that for the ORR and the present BSR rod system; i.e., $\sim$0.1% $\Delta k/k$ per sec. SPERT data[14,15] for a ramp insertion of reactivity at 0.1% k/sec indicate that the following maximum conditions could be expected in a PCA startup runaway; i.e., the shim rods are withdrawn at full speed until they are scrammed by the safety system.

Rate of reactivity addition	= 0.1% k/sec
Maximum power level obtained	$\cong$ 70 Mw
Minimum period	$\cong$ 48 msec
Maximum energy released to peak of excursion	$\cong$ 4.3 Mw sec
Maximum fuel-surface temperature attained	$\cong$ 160°C

These data were obtained from SPERT tests using a core that was similar to the 28-element PCA core discussed in this report, the major difference being in the water head (2 ft for SPERT vs 17 ft for the PCA). Tests with SPERT IV (Ref. 16) indicate that the reactor response was not significantly affected by increasing the water head to 18 ft. The fuel surface temperatures, however, were increased somewhat by the greater head.

The peak power listed above was self-limiting in the SPERT tests (temperature increases, void formation, etc.) and was not a result of scramming the control rods. The rods were scrammed several seconds after this peak was attained and the power level had declined to a much lower value (only a few megawatts). Therefore, it appears that the conditions stated are upper limits for the PCA if one assumes proper functioning of the safety system. Moreover, since the SPERT test using ramp rates as above resulted in no fuel element melting or mechanical damage to the reactor, it is concluded that the startup accident at the PCA would not cause appreciable damage to the reactor system or injury to personnel.

7.2. Radiation Levels

Figure 15 shows measured values of equilibrium gamma radiation levels at various points in the reactor area during operation of the BSR at a power level of 1 Mw with natural convection cooling and at 2 Mw with forced-convection cooling. Contributions due to PCA operation at full power (10 kw) are negligible.

The pool cooling system (BSR system) includes a decay tank which provides adequate delay time to result in almost complete elimination of very short half-life nuclides such as ^{16}N and a considerable reduction in the activity of those nuclides (generated by operation of the BSR) having longer half-lives. An off-gas connection to the tank removes a substantial portion of the gases carried by the water. The filter and demineralizer systems continuously remove a fraction of any radioactive particles and dissolved materials from the water. The cooling water is returned to the pool (during BSR operation at steady-state condition) at a temperature slightly lower than the pool-water temperature thus reducing any tendency for the return water to rise to the pool surface. Therefore, the quantity

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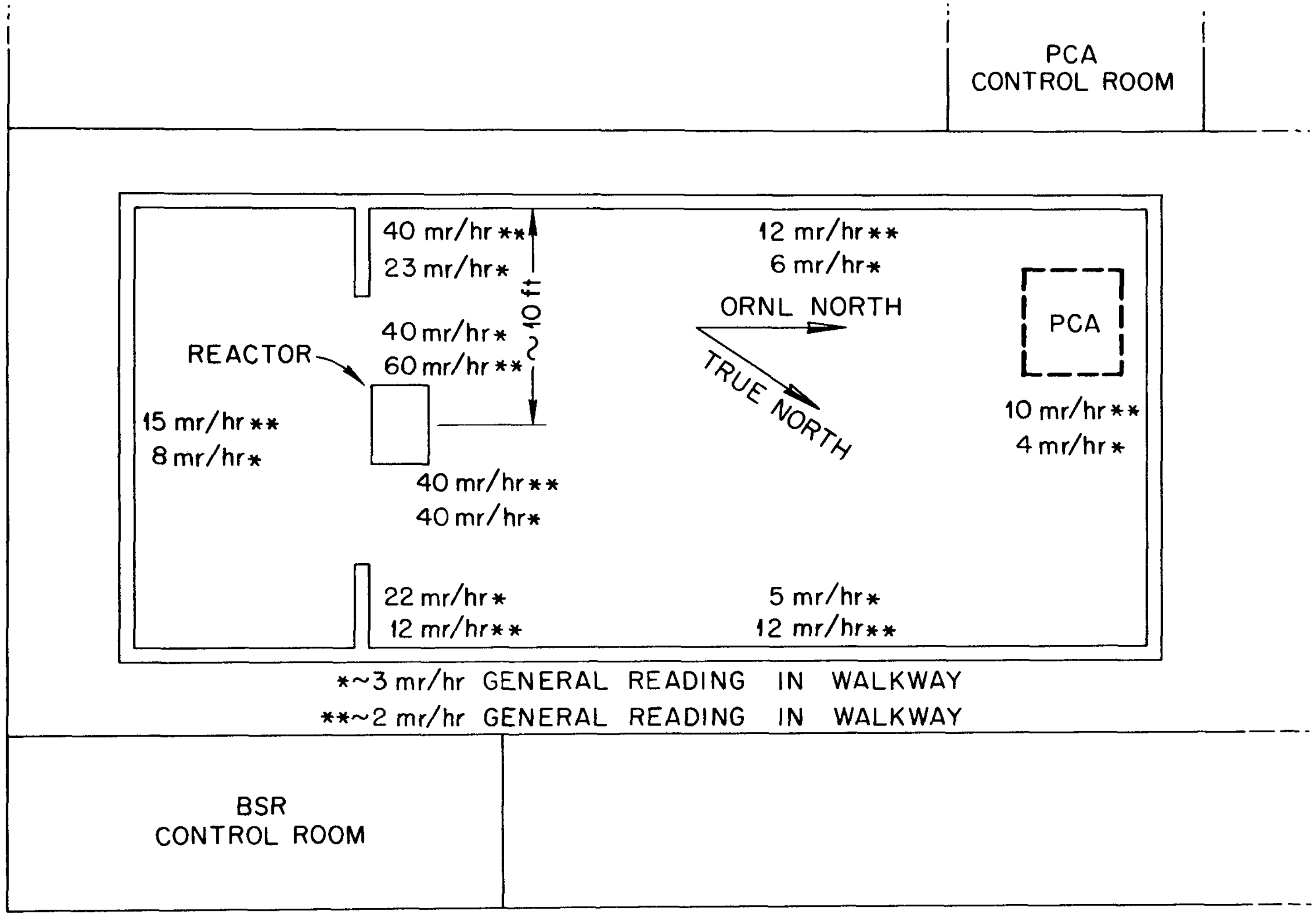

Fig. 15. Equilibrium Gamma Radiation Levels Due to BSR Operation: (1) **at 1 Mw with natural convection cooling; (2) *at 2 Mw with forced convection cooling.

Activity Levels Due to Full-Power (10 kw) Operation of the PCA Are Negligible Compared to the Values Shown.

of radioactive materials in the cooling water is usually of only slight significance. Because of the low power level and low accumulated fission product inventory in the PCA fuel elements, PCA operation contributes negligibly to the pool water activity. In general, except relatively soon after operation of the PCA, the gamma-ray intensity from PCA fuel elements is only a few mr/hr at contact.

7.3. Fuel-Element Rupture

It is extremely unlikely that a fuel-element rupture will occur in the PCA. Moreover, if such an event did occur it could result in little or no release of fission products because of the low fission-product inventory in the fuel elements normally used in the reactor. However, even if this were not the case, radiation monitoring and warnings to personnel would certainly be adequate to permit building evacuation without any severe exposure of personnel.

7.4. Loss of Coolant

The maximum power level of the PCA is administratively limited to 10 kw. At this power level the maximum heat flux is only $\sim$150 Btu/hr-ft . Further, since there is little buildup of fission-product inventory in the PCA fuel, there is little heat production once the reactor is shut down. Upon loss of pool water the PCA, if previously operated, would be shut down due to the loss of moderators and reflector. It has been demonstrated[17] that no damage to fuel elements occurs in such a reactor under these conditions even if it had previously been operated at a power level of 1 Mw or greater for several hours prior to coolant loss. Therefore, coolant loss would cause little problem for the PCA; a more serious problem would be radiation from the exposed BSR core.

7.5. Work On Or Near the Reactor

Any alteration of the reactor is performed in accordance with the general operating procedures for the PCA[9] (Appendices C and D). These require that, except for routine operations (covered by standing detailed procedures), special procedures be prepared and approved by the reactor supervisor and the Operations Division technical staff prior to the performance of any work that may affect the reactor.

In addition to these procedural precautions, other safeguards include the following:

1. Mechanical stops (welded to the rails upon which the BSR bridge travels) ensure that the BSR will not be located close enough to the Pool Critical Assembly (PCA) to create problems due to nuclear coupling of the two reactors.
2. All fuel is stored in racks that have been shown to be critically safe when filled with 240-g fuel elements and arranged in any credible manner. These are sufficiently subcritical when filled with fuel elements as to present no problem even if placed directly against the reactor. Their use has been reviewed and approved by the ORNL Criticality Committee.
3. Any work (related to the PCA) performed in or over the pool requires prior approval by the reactor supervisor and direct supervision by an authorized member (technically trained) of the Operations Division.

With these and other associated safeguards, the inadvertent insertion of any large amount of reactivity is considered incredible. A "large amount" is defined to be an amount in excess of that which can be safely compensated for by the control system or which could cause serious personnel injury or equipment damage. Since the shutdown margin is required to be at least equal to the available excess reactivity,* the shutdown reactor could safely tolerate reactivity additions up to approximately this same amount; i.e., the total rod worth must be at least twice the available excess k. Moreover, procedures[9] require that the reactor be shut down prior to, and throughout, the performance of operations which may add any significant amounts of reactivity.

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*For the purposes of the report, "available excess reactivity" is defined as "that excess reactivity which can be added to the core by manipulation of the control rods or by credible or inadvertent movement, change, or failure of experiments or other materials, systems, or components associated with the reactor".

8. HAZARDS RESULTING FROM ACTS OF GOD AND SABOTAGE

8.1. Fires, Floods, and Windstorms

The first floor of Building 3010, which houses the PCA and BSR, is located at elevation 828 ft, which is 40 ft above the level of the highest known flood; hence, there is little, if any, possibility of flooding. Even should this occur, it would not cause a reactivity problem to the reactor which is normally under water.

Because of the type of construction, there is little danger of fire. A fire involving the reactor control and protection system could conceivably cause a malfunction of the control systems; however, because of the design of the safety system, this would not inhibit its ability to shut down the reactor.

A severe windstorm could damage the building and perhaps reduce or destroy the containment potential. As in the case of fire, however, this would not inhibit the scram function of the protection system.

8.2. Earthquakes

The reactor, if initially operating, would be shut down in the event of a severe earthquake either by the rods falling off the magnets or because of the fact that the reactor is undermoderated and thus stable under crushing deformations. Even assuming an immediate loss of water from the pool and some crushing of the reactor fuel, the fission product inventory in the PCA fuel is so low that no radiation problem would result. The most probable consequence of moderately severe earthquake damage leads to the loss-of-pool-water problem previously discussed. The Uniform Building Code classifies this area as Zone 1 (minor damage) for seismic probability.

8.3. Sabotage

Sabotage is, of course, possible and could probably be most readily effected by the use of some timer-operated explosive placed near the reactor core. However, this reactor is located in the controlled area of the Oak Ridge National Laboratory and the reactor building is either locked or under the constant surveillance of operating personnel. When the building is unoccupied by operating personnel, it is inspected on a regular schedule.

9. MAXIMUM HYPOTHETICAL ACCIDENT

As has been pointed out above, both the site and the nuclear characteristics of the PCA are virtually identical to those of the BSR with the exception of the fact that operation of the PCA is limited to a maximum power level of 0.005 that of the BSR. Actually the PCA is operated intermittently, usually only for short periods of time and at power levels well below the authorized power level of 10 kw. It is estimated that during the period since its installation in 1958 through calendar year 1969, the total energy output of the PCA has been less than 500 kw hours (0.02 Mw days).

It is difficult to credit the possibility of an accident which would cause any significant damage to the PCA fuel; however, if one considers a maximum hypothetical accident of the same magnitude as that postulated for the BSR, namely the melting of 50% of the fuel followed by the release of 50% of the halogens, 100% of the noble gases, and 2% of the solids from the effected region of the core, the consequences would be far less than that calculated for the BSR. This can easily be seen by a perusal of Table 1 which gives a comparison of the more important fission product inventories present in the two cases.

TABLE 1

Fission Product Inventories Associated with the Maximum Credible Accident

	Inventory (Curies)	
Nuclide	BSR[a]	PCA[b]
^{131}I	4.34×10^4	1.86×10^1
^{133}I	1.10×10^5	3.16×10^2
^{133}Xe	1.16×10^5	7.44×10^1
^{135}Xe	9.39×10^4	4.81×10^2
^{87}Kr	4.20×10^4	2.93×10^2
^{88}Kr	6.02×10^4	3.69×10^2
^{89}Kr	7.74×10^4	2.64×10^4
^{90}Sr	1.84×10^3	6.97×10^{-2}
^{137}Cs	2.96×10^3	1.10×10^{-1}

[a]Based on 14 months operation at 2 Mw.

[b]Based on 500 kw hrs plus 24 hrs recent operation at 10 kw and augmented by a 10^{18} fission burst.

The accident postulated above has been analyzed in detail for the case of the BSR[1] and it was found that the consequences both on and off site were acceptable. Since the only difference between the postulated BSR and PCA accidents is the lower fission-product inventory in the latter, it is concluded that the PCA Maximum Hypothetical Accident is also acceptable. It should perhaps be pointed out in this connection that should it be desirable to operate the PCA with fuel containing a larger inventory of fission products than that contained in the BSR this would require an internal safety review.

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APPENDIX A

A GENERAL DESCRIPTION OF THE PCA INSTRUMENTATION AND CONTROL SYSTEM

A. E. G. Bates

The purpose here is to describe the special features of the PCA instrumentation and control (nonsafety) system (see Figure A.1). Reactivity control is by means of three shim-safety rods which are coupled to their drives by direct lift electromagnets (clutches) and one regulating rod which is directly coupled to its drive. All shim-safety rods are withdrawn simultaneously in startup and are held in close alignment during operation to place them in their most effective position for taking corrective action if needed.

Meanings of Permissive and Selector Designations

The information presented in Figure A.1 occasionally is relatively cryptic. The description of features in later paragraphs will clarify most of the designations but not all. Others are

1. Local Scram - Manual scram-initiating switch or pushbutton on the reactor operating panel, cubicle, or console.
2. Monitron Level Too High - This means that the radiation detected by one or more of several suitably located area monitors is in excess by an established maximum. The assumed source of radiation in this case is the PCA or BSR core.
3. Servo Level Too High (or Too Low) - Information to the servo system that the reactor power is above (or below) the demand set point.
4. Log-N Level Greater Than Set Point - Reactor power as determined by the log-N channel is compared to that established by the demand set point of the servo system. If the level is appreciably higher than the set point, the servo is suspect, corrective action (reverse with alarm) ensues, and the servo is automatically turned off.

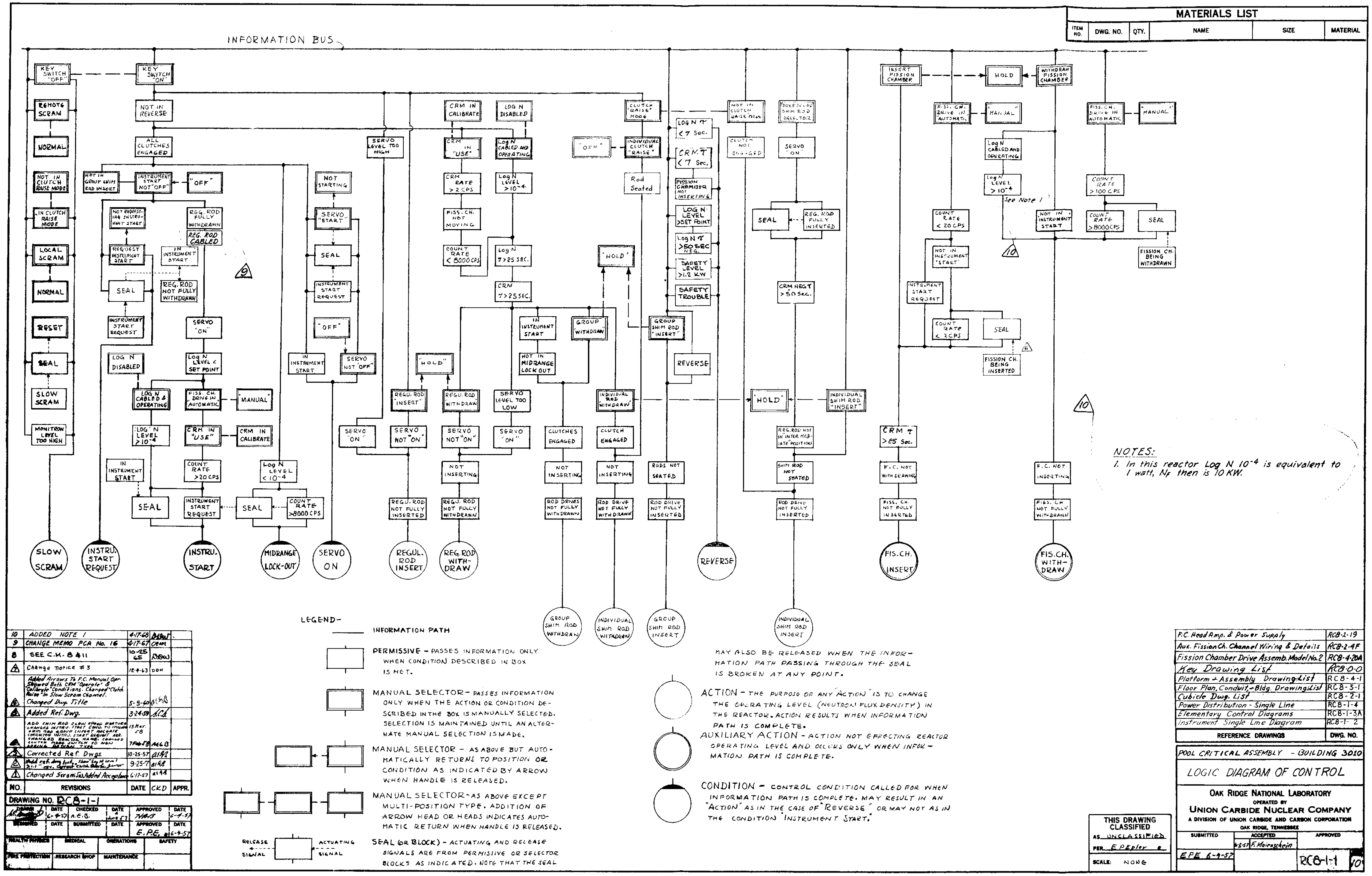

Fig. A-1. PCA Logic Diagram of Control

5. Preferred Shim Rod Selector - Means for selecting the shim-safety rod to be under servo control for insertion only.

Features

The features found in ORNL control systems for reactors of the PCA type include the following:

1. Instrument start
2. Automatic fission-chamber positioning
3. Raise-clutch mode
4. Automatic rundown of rod drives
5. Servo system
6. Key switch

While two modes of starting and operating the reactor are provided in the control system (manual and instrument start), only the latter will be described since the manual mode is completely straightforward. It is assumed that all instrument channels have been checked out and that all parts of the safety and nonsafety systems are functioning properly.

The picoammeter and servo demand are set at the desired power level and then the "instrument start request" pushbutton is depressed. Provided that the count rate exceeds 20 cps, the servo system immediately starts withdrawing the regulating rod since the reactor power is below the demand set point. When the upper limit of the servo rod is reached, the upper-limit switch establishes the "instrument-start" condition. All shim-safety rods will begin withdrawing at full speed until a transient 25-sec positive period is detected by the counting channel at which point withdrawal is stopped. The selection of 25 sec for the minimum operating period is arbitrary. Shorter periods are practical for automatic start but may be troublesome for the operator if manual start is being used. The startup time saved by using short periods is too small to be of any interest. Returning to the startup, the transient short period will grow longer than 25 sec and rod withdrawal will resume. The process is repeated but with shorter and shorter run and longer and longer stop intervals until a steady 25-sec positive period is established. The power rise continues and, on passing a level corresponding to 8000 counts/sec on the CRM, two changes occur. First, the control is placed in "midrange lockout" which prevents further automatic rod withdrawal. Second, the fission chamber withdraws until it reaches a flux corresponding to something less than 100 counts/sec (this puts the counting channel back near the lower end of its operating range). In addition, rod withdrawal is blocked as soon as a value of 8000 counts/sec is reached and the blockage continues until the chamber stops moving; this action is meaningless except during manual starts. None of these changes affect rod positions and so the reactor power continues to rise on a 25-sec period until at log-N $>10^{-4} N_F$ the "midrange lockout" is cleared. Except for period permissives, instrument control shifts from the counting to the log-N channel. If, after midrange lockout is cleared, the period grows longer than 25 sec, all rods will withdraw again as needed to shorten the period. Instrument start is terminated when the regulating rod leaves its fully withdrawn position. Instrument start is also terminated instantly in the event of a rod drop or a reverse. Both actions are called to the operator's attention by the annunciator system.

No automatic shim rod withdrawal is permitted once instrument start is terminated. During a long run, if the regulating rod does become fully withdrawn, an alarm is sounded and the operator will have to correct the situation by withdrawing one or more shim-safety rods. The servo system, however, is permitted to insert a single shim-safety rod if full regulating rod insertion is insufficient to limit reactor power. Servo failure can cause the reactor power to float along or go down or up. Defense-in-depth gives ample protection against power rises and the other two failures have only nuisance value.

Automatic fission chamber positioning is, of course, a necessary part of the instrument-start system. It is a convenience, however, when starting the reactor manually and is used routinely. In its automatic mode the system repositions the fission chamber as needed to keep the counting channel within its operating range. Following a reactor shutdown, the chamber is inserted until a counting rate of at least 2 counts/sec is detected. If an instrument start is initiated, the chamber will be inserted automatically until a counting rate at least 20 counts/sec is detected before rod withdrawal starts. When the counting rate reaches

8000 counts/sec (upper end of range), the chamber withdraws until the low end of the range is again reached. While manual insertion of the chamber is permitted at any time, manual withdrawl is blocked until the log-N channel assumes control. The most undesirable result of inadvertent insertion of the chamber would be the generation of an apparent short period which, if short enough, could block rod withdrawal or initiate a reverse. Inadvertent chamber withdrawal would interfere with an instrument start while the startup is under the control of the counting channel and is therefore prohibited. The chamber-drive control may be placed in a manual mode, but doing so blocks the instrument start.

The "raise-clutch" mode permits the shim-safety rod drives to be withdrawn for maintenance purposes without withdrawing the control rods. When the manual selector is turned to establish this mode, a "slow" scram is effected. This declutches the rods and should permit the drives to withdraw without moving the rods. There is an instrumented check on this which requires that, if the rods do not remain seated, the rod-drive withdrawal is immediately and automatically stopped.

The "automatic rundown" of the rod drives is a convenience for the operator. Whenever a rod and its drive become separated (scram or if a rod should fall), the rod-drive automatically inserts until the magnet contacts the rod. This feature is blocked when the control is in the "raise clutch" mode to permit drive withdrawal.

The servo system relieves the operator of the tedium of holding the reactor power at the desired level and, in fact, holds reactor power closer to the control point than the operator can. It must be on and operating before the instrument start mode can be established. It cannot thereafter be turned off without first taking the reactor controls out of the "instrument start" mode. When starting the reactor manually, the operator turns the servo on and checks to see that it withdraws the regulating rod fully. Then, and only then, does he withdraw the shim-safety rods. The intent is to place the regulating rod in the best possible position to stop the reactor power rise at the set point. The servo may fail in such a way that it withdraws the regulating rod fully placing the reactor on a positive period. If the operator does not discover this and take corrective action, it will be done for him by the log-N and power-level channels through the medium of a reverse. The servo is turned off by the reverse and can be turned back on only by the operator.

The "key switch" is an aid to administration and is used to prevent unauthorized manipulation of rod drives. When the switch is off, the reactor is scrammed and the withdraw circuit control power is disconnected. The switch can be turned on only when the proper key is in its lock and, further, the key cannot be removed until the switch is turned off. Rod insertion circuits are not turned off and, in fact, are arranged such that rod insertion requests always take precedence over those for rod withdrawal.

APPENDIX B

THE PROTECTION SYSTEM

A. E. G. Bates

The protection (safety) system, as the name implies, is intended as a protective means for the reactor rather than as instrumentation primarily for operating. In fact, it is the "last ditch" protector and, as such, is designed for maximum reliability. Power supplies, signal circuits, and connecting cables are monitored to detect failures which might originate in the equipment itself. An audible alarm calls the operator's attention to the fact that trouble has developed.

Although the protection afforded by the system is actually against high-temperature operation of the fuel (and, in some cases, possibly the moderator) rather than against radiation damage to the reactor assembly, flux measurements under some conditions may be substituted for temperature measurements in a reactor since the rate of heat generation in the core is proportional to the fission rate. A major advantage of this method is that it has little inertia, responding almost instantly to changes in the flux at the chamber. Some care must be exercised in selecting the chamber position, however. Placing it outside a thick graphite reflector, for example, will result in undesirable delays in the detection of flux changes in the core because of the finite velocity of neutron propagation through the reflector material.

Figure B.1 illustrates the basic arrangement of a typical protection system. It makes use of three independent safety channels each consisting of an ion chamber, a safety preamplifier and a sigma amplifier, all feeding into a common sigma bus. Attached to this bus are the several magnet amplifiers which control the current to the shim safety rod magnets. During normal steady-state operation of the associated reactor, the sigma amplifiers maintain the sigma bus at a constant electric potential with respect to ground. As long as this potential is applied to the input of the magnet amplifiers, the current through the rod magnets will be maintained at a value somewhat greater than is required to support the shim safety rods. Dropping the reactor power at a uniform rate from N_F will

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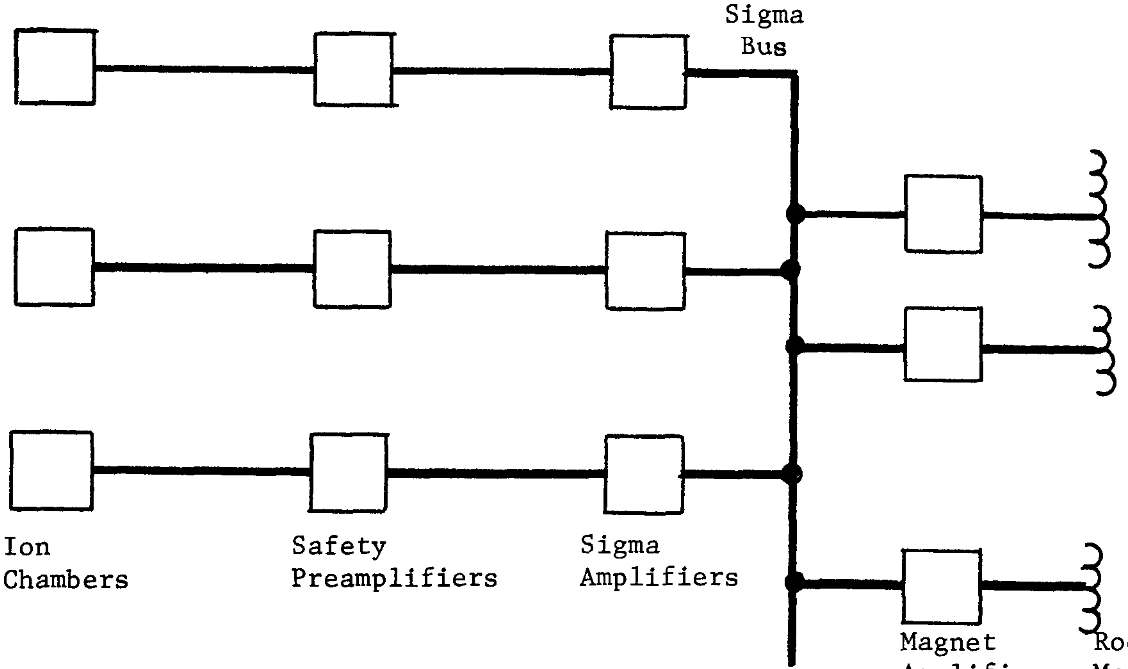

Fig. B-1. Basic Arrangement of Typical Safety System

result in the sigma bus voltage falling slowly until about 10^{-2} N_F is reached, but below this no further change occurs. The rod magnet currents, therefore, rise slowly and then level off at a constant value which is maintained at all reactor powers below 10^{-2} N_F to provide maximum holding force during startup rod withdrawal.

Raising the power level above N_F increases the sigma bus potential slowly at first but more rapidly thereafter. At a power level slightly below 1.5 N_F the rod magnet currents have fallen to such low values that the magnets can just support the shim safety rods. Any slight further rise in the reactor power will result in a sufficient reduction in currents through all the magnets that the rods will be released to scram the reactor.

It will be noted that the log-N period channel also ties to the sigma bus. The log-N period scram is intended to limit the power excursion of a reactor during a startup accident. The power-level safeties also provide protection, but they do not act until the power has reached 1.5 N_F or so, while the log-N channel will normally scram the reactor at much lower levels thus reducing the power excursion proportionally. The period amplifier of the log-N channel raises the sigma bus voltage an amount which is inversely proportional to the length of the period detected and is adjusted to scram when this is one second or shorter.

The one-second period value is a compromise. In the power range the servo may induce periods of ten seconds or shorter during its normal control operation. Obviously, the period channel should be set to scram only at periods shorter than these to prevent its interfering with servo operation. On the other hand, waiting until the period is quite short before taking corrective action is undesirable. There is some delay (10 to 30 milliseconds) between the detection of an undesired period and the time when corrective action actually begins to take effect. This means that the shorter the period selected for scramming, the greater the power overshoot. Any scram setting longer than about 100 milliseconds is adequate, and one second provides considerable margin beyond this.

The sigma bus was so named because it is a common or summing point for the outputs of the safety and log-N period channels. A more descriptive name might be "auction bus" since the voltage on the bus is very nearly that of the channel having the highest output. Therefore, any channel can independently raise the bus potential high enough to produce a reactor scram. The sigma bus is a critical point in the system and is carefully insulated electrically and protected mechanically in every installation. Insulation failures resulting in partial or complete grounds on the bus might, in a system such as has been described, result in reduced sensitivity or total loss of protection. To guard against such occurrences the magnet amplifiers are designed to be sensitive to both increases and decreases in bus potential. To permit operation of the reactor, then, it is necessary to maintain the sigma bus voltage within a narrow margin above or below a value fixed by the design of the magnet amplifiers.

In determining the number of safety channels it is apparent that the use of only one would not provide sufficient reliability and would not be conducive to continuity of operation. Although two channels would probably provide as high a degree of reliability and safety as can be justified, this would not provide for continuity of operation in the event of instrument trouble. With two, the probability of having instrument trouble is much higher than with one so the number of shutdowns due to instrument trouble will be greater because it is not permissible to operate the reactor when it is protected only by a single channel. Three, then, seems to be a minimum to ensure that two are always available in the event that one is out of service for repair. The protection reliability is higher with three but so is the probability of a channel failure. The point is, though, that the probability of two successive failures is very low and therefore the number of shutdowns from such causes is reduced to an acceptably low value.

The fundamental problem in the control of reactors built with large excess reactivity has been discussed by Newson.* He shows that for rates of change of k consistent with reasonable startup times, prompt critical may be reached before accurate instruments are in range to indicate the flux. The short periods on which the reactor then rises preclude, or make unlikely, corrective action by the reactor operator in time to avert a disaster. A fast safety device which acts when the reactor reaches a predetermined value above the normal operating level may be made fast enough to shut down the reactor before damage results. Such a device will shut down the reactor in a time determined by the period of the reactor, by the delay in the safety system, and by the rate at which the safety system can decrease k.

In the PCA the primary safety device is a system of shim-safety rods. These are suspended from electromagnets and are driven up and down by electric motors. When it becomes necessary to decrease k quickly, the current in the electromagnets is turned off and the rods fall due to gravity to the fully seated position in the core. (Normally the total flight time of the PCA rods is about 500 msec.) This letting-go of one or more rods is known as a scram.

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*H. W. Newson, The Control Problem in Piles Capable of Very Short Periods, MonP-271 (April 21, 1947).

A number of electrical devices will perform the duties of some of the components of the protection system. In particular, the ordinary sensitive relay has many attractive features. It is a device which has been manufactured successfully for many years. Its reliability when used conservatively is very great. With little additional trouble it may, by use of multiple contacts, be made to serve several functions. Since in this application the primary signals are generated in ionization chambers, relays of adequate ruggedness cannot be operated directly with the small currents available. An amplifier of some sort is therefore required. This is a drawback, but not a serious one. A much more serious shortcoming of the relay is its unpredictability; that is, the impossibility of applying any test which will predict its response (or lack of response) to the next signal requiring operation. Any trial operation by way of such a test is, by its nature, a "test to destruction" proving only that the relay did work, not that it will. What is required, then, is a device that acts like a relay in its operation but provides some means of testing or monitoring its condition. The electronic systems described briefly above approach this mode of operation while retaining the feature that, with care, possible malfunction may be detected before a sufficiently dangerous condition develops that proper functioning of the ailing circuit becomes imperative.

APPENDIX C

OPERATING SAFETY LIMITS FOR THE POOL CRITICAL ASSEMBLY (PCA)

R. A. Costner, Jr.

Introduction

The Pool Critical Assembly (PCA) operating safety limits designate limits within which the reactor can be operated safely. The limits specified were determined by design or experience and are listed according to the functional components. Included are limitations placed on experiments and those placed on administration of the operation. AEC approval is required for changes to the operating safety limits.

The purpose of this document is to establish a limit for each operating variable which has direct reactor-safety significance. Each limit designates a realistic boundary to the operating range of the variable; therefore, each limit can be approached with confidence that the safety of the reactor will not be compromised. The operating procedures shall be prepared so as to provide reasonable assurance that the reactor will be operated within the stated operating safety limits.

The PCA does not operate continuously but on an "as-needed" basis for research and training. The power level is limited to 10 kw (administratively), but most of the operation is at very low power levels.

Operating Limits

A. Reactor Core

1. Maximum fuel loading - The maximum total mass of fuel in the core shall be adjusted so that the ganged shim-safety rods will have to be withdrawn at least 50% of their worth before criticality is achieved. At no time shall the reactor be allowed to continue critical operation with the ganged rods inserted more than 50% of their worth.
2. Maximum steady-state power level - 10 kw nominal (administrative limit).
3. Safety power-level scram limit - Maximum setting (nominal) shall be 150% of maximum steady-state power level (administrative limit).

4. Maximum heat flux in the core for natural-convection flow and 120°F inlet water temperature - 0.96×10^5 Btu hr^{-1} ft^{-2}.*
5. The separation distance between the BSR core and the PCA core shall be maintained at more than 10 in. by means of permanently fixed mechanical stops on the rails carrying the BSR core.

B. Primary Cooling System
1. Maximum pool-water bulk temperature - 120°F.
2. Minimum flow rate - natural convection.
3. The reactor cooling water is monitored periodically and necessary adjustments are made to maintain a pH of between 5.5 and 6.5.**
4. Maximum radioactivity of coolant water - The radioactivity of the water shall be maintained at a level such that no excessive exposure to personnel will occur, as specified in AEC Manual Chapter 0524.
5. The water level in the reactor pool normally shall be >12 ft above the reactor core.

C. Emergency Cooling Facility

It has been determined experimentally that the heat-generation rate following shutdown is sufficiently low that no special provisions for afterheat removal are required.

D. Control and Safety Systems
1. Mechanical control system
    a. Minimum number of control elements - three.
    b. Minimum number of control rods withdrawn to attain criticality - The reactor shall be so arranged that criticality cannot be achieved by complete withdrawal of any one control rod while the others are completely inserted.

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*This value is 15% below the burnout heat flux interpolated from the experimental data of W. R. Gambill and R. D. Bundy described in ORNL-3026, Burnout Heat Fluxes for Low-Pressured Water in Natural Circulation, and is approximately a factor of 6 greater than the highest heat flux to be expected during operation.

**These are nominal limits which may be exceeded for short periods of time without requiring a reactor shutdown.

c. Maximum release time of control elements (magnet and latch combined) - 30 msec.

d. Maximum scram time (maximum time-of-flight from fully withdrawn to bottom seat) - 800 msec (includes release time).

e. Maximum rate of increase of reactivity (using control rods) - 0.15% $\Delta k/k$ per sec.

f. Servo control - The amount of positive reactivity controlled by the servo system shall be limited to 0.5% $\Delta k/k$.

g. Operational checks - Maximum time between successive checks of the release time and the scram time shall be 3 months or before reactor operation is required (whichever is longest).

2. Control and safety instrumentation

   a. Minimum reactor control and safety instrumentation required for startup.*

      (1) Two power-level safety channels.

      (2) One log-N period channel (to initiate a reactor scram at periods less than 1 sec).

      (3) One neutron-level detection channel (log-N or fission chamber) that is reliably detecting the neutron level in the reactor.

      (4) One radiation monitor located in the reactor room.

      (5) One continuous air monitor located in the reactor room.

   b. Minimum safety and control instrumentation required during steady-state operation.

      (1) Two power-level safety channels.

      (2) One radiation monitor located in the reactor room.

      (3) One continuous air monitor located in the reactor room.

   c. Control and safety instrument checks

      (1) Functional checks of instruments - Prior to each reactor startup which occurs more than eight hours following a shutdown.

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*When more than one instrument of the same kind is specified, it is to be understood that either instrument will provide the required safety action.

(2) Maximum period between comprehensive instrument calibration and checkout tests - Three months or before reactor operation is required (whichever is longest).

E. Radiation Monitoring Systems

Radiation level monitors - A minimum of two operable radiation monitors which provide audible alarms shall be located at appropriate points within the reactor building.

F. Experiments

Each in-reactor experiment is subjected to comprehensive reviews and hazards evaluations by the Laboratory's Reactor Experiment Review Committee and/or the Operations Division. In this way, an experiment is approved for operation within safety limits applicable only to that specific experiment. Appropriate limits are placed upon any materials, systems, or components that may (for any credible reason) affect the reactor reactivity in such a manner or to such a degree that unsafe conditons could result.

1. With respect to reactivity effects, experiments* are considered and approved as follows:
   a. An experiment is approved routinely if the maximum change in reactivity that can be caused by the experiment is conservatively less than the total amount of reactivity controlled by the servo system (less than 0.5% $\Delta k/k$).
   b. Experiments having reactivity worths greater than that in item F.1.a are considered in more detail--particularly if failure or malfunction of the experiments may cause changes in these worths. Consideration is given to the total worth, rates of change of reactivity, and to particular situations that may be associated with these changes. Experiments shall be approved only if it is found incredible that the experiment could cause unacceptable hazards. Review by the Laboratory's Reactor Experiment Review Committee is required.

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*These limits on reactivity effects of experiments apply to those experiments for which the reactor is used only as a source of radiation. Those experiments in which the reactor itself is part of the experiment (e.g., approach-to-critical-loading experiments) are limited by item A.1.

2. With respect to energy release, experiments are considered and approved as follows:

   a. The amount of any explosive or mixture of materials (such as hydrogen and oxygen) to be placed in or near the reactor shall be limited to the equivalent of 1 g of TNT (1.1 kcal).

   b. The energy release which might result from the reaction of any of the so-called reactive materials, such as Na, Li, or K, with the reactor coolant or with the experiment coolant by any credible mechanism shall be limited to 100 kcal unless a monitored double barrier exists between the material and the coolant. Experiments of this type in which the potential energy release is greater than 500 kcal shall not be installed without prior approval by the AEC.

G. Administrative and Procedural Safeguards

1. Personnel qualifications - The reactor shall be operated only by qualified personnel approved by the Operations Division Superintendent.

2. Minimum staff requirements for the PCA operation during any shift - One supervisor qualified to operate the PCA shall be in the operating area whenever the reactor is operating.

3. Procedures - The reactor is operated in conformance with documented operating procedures. In no instance shall the operating procedures authorize operation of the reactor in excess of any operating safety limit listed above.

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