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OPERATION BREN

OPERATION PLAN AND HAZARDS REPORT—OPERATION BREN

F. W. Sanders, F. F. Haywood, M. I. Lundin,
L. W. Gilley, J. S. Cheka, and D. R. Ward

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OPERATION PLAN AND HAZARDS REPORT – OPERATION BREN

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January 1962

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ABSTRACT

Operation BREN is a continuation of the studies begun in 1956 for evaluating the radiation doses received by persons exposed to nuclear weapons, especially the residents of Hiroshima and Nagasaki, Japan. The Oak Ridge National Laboratory Health Physics Research Reactor (HPRR) will be mounted in a hoist car on the side of a 1500-ft tower in Area 4 of the Nevada Test Site and will be operated at various elevations. When experiments utilizing the reactor are completed, a 1200-curie Co^{60} gamma-ray source will be substituted for the reactor. Measurements of the energy, angular, and spacial distributions of the radiations from these sources will be performed. Various shields, including facsimiles of Japanese houses typical of those in Hiroshima and Nagasaki in 1945, will be studied. These experiments are organized into Program 1. Other selected programs that can utilize the radiation fields available on a non-interference basis are included as parts of the Operation.

Descriptions of the HPRR and the Co^{60} source, their operating procedures, the manner in which they will be used, and possible hazards are included.

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CONTENTS

CONTENTS (Continued)

ILLUSTRATIONS

CHAPTER 2 OPERATION BREN ORGANIZATION AND RESPONSIBILITIES

CHAPTER 3 RADIATION SOURCES

CHAPTER 5 OPERATING PROCEDURES

TABLES

CHAPTER 3 RADIATION SOURCES

TABLES (Continued)

CHAPTER 6 SAFETY

Chapter 1

INTRODUCTION

Operation BREN (Bare Reactor Experiment Nevada) is a continuation of the studies commenced in 1956 for evaluating the radiation doses received by persons exposed to nuclear weapons, especially the residents of Hiroshima and Nagasaki, Japan. To obtain experimental data that can be used to determine the relation between the dose of radiation received and radiation-induced biological changes in man, the Atomic Bomb Casualty Commission has studied the populations of Hiroshima and Nagasaki for several years. The Health Physics Division of the Oak Ridge National Laboratory (ORNL) has undertaken the task of developing a method of evaluating the doses received by the survivors of the bombings. Data for use in this study, Ichiban, were obtained during nuclear-weapons test Operations Teapot, Plumbbob (Ref. 1), and Hardtack (Phase II) (Ref. 2). The purpose of Operation BREN is to provide additional data that are necessary to Ichiban.

During Operation BREN the radiation fields from nuclear weapons will be simulated by means of an unshielded nuclear reactor, the ORNL Health Physics Research Reactor (HPRR), suspended at several elevations on a 1500-ft tower located in Area 4 of the Nevada Test Site (NTS). Such a reactor provides a good simulation of the neutron field due to a nuclear weapon because in both devices the leakage neutrons escape from the assembled fissile material and are moderated by air. At distances from the source greater than a few hundred yards, equilibrium obtains in the neutron-energy distribution, and energy and angular distributions are not strongly influenced by the design of the neutron source. Use of a reactor operating in the steady-power-level mode permits use of sensitive "in-laboratory" type instruments that could not be used for brief intervals at high dose rates or with blast conditions present.

Several components of the gamma-ray field of a nuclear device are well simulated by means of a bare reactor. Because the neutron field so closely approximates that from a weapon, the gamma rays originating in neutron—air interactions will have the same energy and angular distributions as the corresponding weapons gamma rays. Prompt gamma rays from fission leak through the assembled fissile material as do those from fission in weapons. The gamma rays from fission products in a reactor, however, leak from the assembled fissile material, whereas fission products from a weapons detonation are distributed in an expanding and rising cloud.

To separate, at least semiquantitatively, the components of the gamma-ray field, energy and dose measurements (as functions of angle of incidence) will be made during steady-power-level operation of the HPRR. Similar measurements will be made as functions of time after the reactor is operated in the burst mode. A 1200-curie Co^{60} point gamma-ray source will be substituted for the HPRR, when the use of the reactor is completed, and energy and dose distributions (as functions of angle of incidence) will be measured.

The HPRR was designed and fabricated for use at the Dosimetry Applications Research Facility at ORNL, and its use during Operation BREN represents only a part of its operating life. For preservation of a high neutron-dose-rate to gamma-ray-dose-rate ratio, fission-product buildup in the core must be minimized by frugal expenditure of the watt-hours of operation necessary to obtain required data. Besides the experiments of Ichiban, selected experiments that can be made on a noninterference basis on the radiation fields available during Program 1 scheduling are included in Operation BREN as programs of higher number (see Fig. 2.1).

Chapter 2

OPERATION BREN ORGANIZATION AND RESPONSIBILITIES

2.1 ORGANIZATION

The organization chart for Operation BREN, showing the distribution of responsibility, is given in Fig. 2.1. Program 1 is organized as follows:

Program 1: Spectra, Distribution, and Attenuation of Mixed Radiation
Program Director, F. W. Sanders, ORNL

Project 1.1: Gamma-ray Dose Measurements in Houses
Project Officer, J. S. Cheka, ORNL

Project 1.2: Energy and Angular Measurements Made with Collimators
Project Officer, J. H. Thorngate, ORNL

Project 1.3: Neutron Dose Measurements in Houses
Project Officer, F. W. Sanders, ORNL

Project 1.4: Energy and Dose Measurements as Functions of Distance and Height
Project Officer, F. F. Haywood, ORNL

2.2 RESPONSIBILITIES

2.2.1 Technical Director

The Technical Director shall be responsible to the Director of the Civil Effects Test Operations (CETO) for the performance of all operational and technical phases of Operation BREN. He may delegate responsibility for various phases of the Operation, under his supervision. Operation of major radiation sources (the HPRR and the 1200-curie Co^{60} gamma-ray source) will be under his direct control. A complete, detailed operating log will be maintained for the Technical Director by the reactor operator which will document starting and stopping times, power level, and source height for each operating period, along with other pertinent information as directed by the Technical Director.

Use of relatively small calibration sources will not be under direct control of the Technical Director.

The Deputy Technical Director shall represent the Technical Director in operational and Technical matters as required.

The Operations Officer (C-3) shall assist the Technical Director in the preparation of operations, maintenance, and construction schedules. He will also maintain contact with the Office of Field Operations, NTS (OFO-NTS) through CETO and will adjust BREN operating schedules to avoid conflict with other activities at NTS. He will inform OFO-NTS of sched-

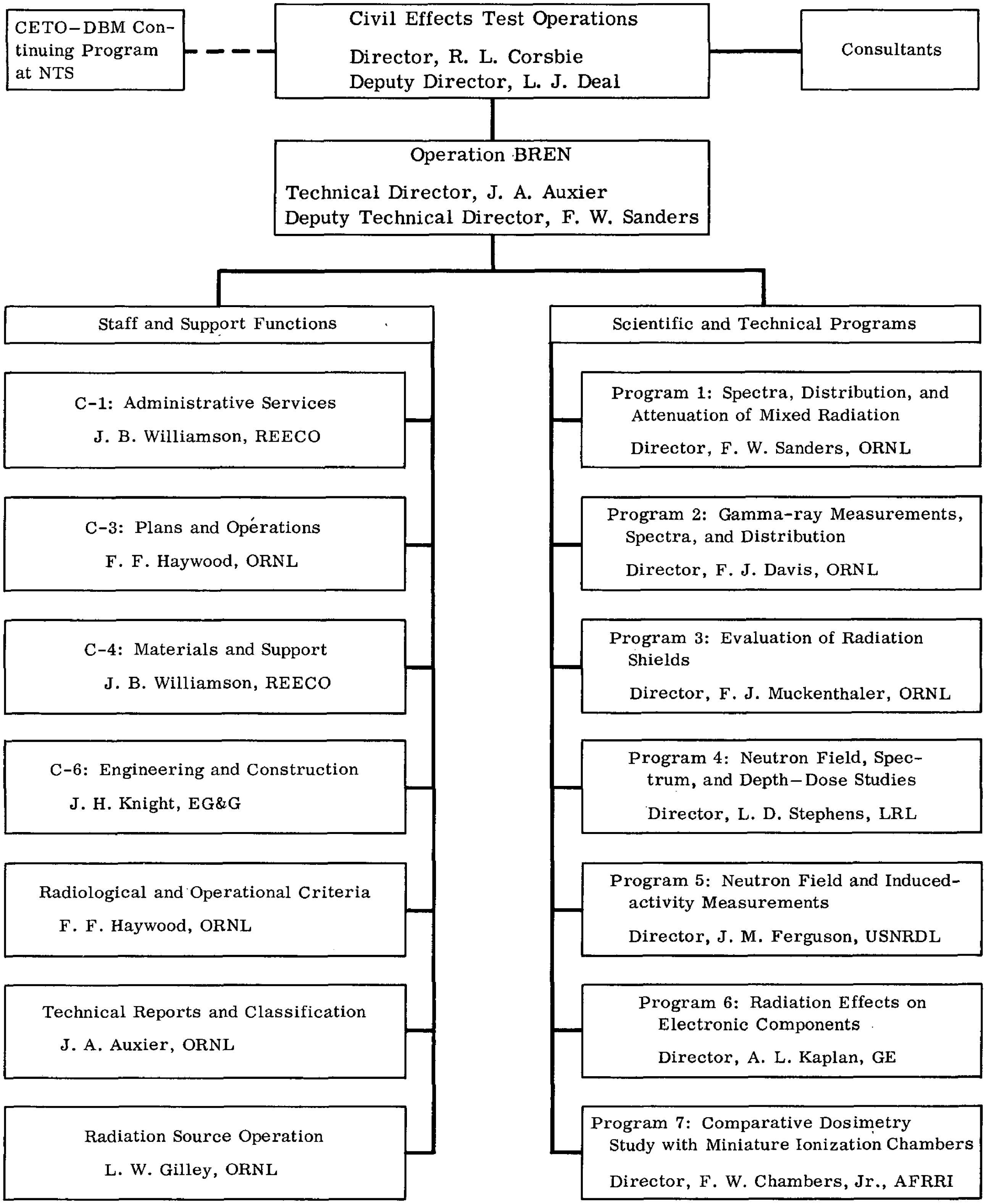

Fig. 2.1—Operation BREN organization chart.

uled reactor or Co^{60}-source operation at least 24 hr in advance and will report start-up and shutdown times for each operating day. He will keep OFO-NTS informed of operational developments that may affect other activities at NTS.

The Radiological and Operational Criteria Officer, referred to in this report as the Rad-Safe Officer, shall assist the Technical Director in formulating and carrying out the Rad-Safe (radiation-safety) plan to prevent radiation injury to participants in Operation BREN or to others at NTS and off-site areas. He shall be responsible for area surveys before, during, and after periods of operation, as required, and shall keep the Technical Director informed of radiation hazards. He shall maintain liaison with the NTS Rad-Safe Officer and with CETO to ensure compliance with applicable NTS procedures. He shall maintain a log that will document all radiation surveys, instructions issued, and important decisions concerning radiation safety.

The C-1, C-4, and C-6 shall be responsible to the Technical Director for administrative services, material and support services, and engineering and construction services, respectively, under normal CETO arrangement.

2.2.2 Radiation Source Officer

The Radiation Source Officer shall be responsible to the Technical Director for the safe operation of the reactor and the Co^{60} source. He will ensure that at least two qualified persons are present in the control room during start-up, operation, and shutdown of the reactor.

He will also be responsible for training additional operators and assistant operators. He will impress upon all personnel who work with or around the reactor (especially the operators) the need for strict adherence to the established operating procedure and the possible consequences of careless or deliberate misuse of the reactor.

2.2.3 Program Directors

The Program Directors shall be responsible to the Technical Director for the conduct of their respective programs and of the personnel involved therein.

The Program Directors shall keep the Technical Director informed regarding the progress, needs, and safety and the preferred schedule of their respective programs.

The Program Directors must inform the Technical Director, according to an established schedule, as to whether their programs will have personnel in the Forward Area during each operational period. The Program Directors shall provide the Technical Director with a muster list of program personnel in Area 4 at a specified time before the start of any operational period.

The Director of Program 1 shall coordinate the projects in Program 1 to obtain the necessary data with a minimum of reactor operation and on a schedule that will provide adequate time for maintenance and construction.

2.2.4 Project Officers

Project Officers shall be responsible to their Program Directors for obtaining the data required by the objectives of their respective projects.

The Project Officers shall be responsible for the procurement, placement, maintenance, and operation of instrumentation required by their respective projects.

Project Officers shall keep their respective Program Directors informed of progress, safety, scheduling, special needs, disposition of personnel, and other matters pertinent to their respective projects.

2.2.5 Reports

All reports, including day-to-day liaison, to OFO-NTS from the Technical Director or subordinates shall be channeled through the Technical Director and CETO.

Written progress reports shall be submitted through the Technical Director to CETO on a mutually agreeable schedule.

All Project and Program Officers shall submit interim reports to the Technical Director before they depart from NTS at the completion of experimentation.

REFERENCES

1. G. S. Hurst and R. H. Ritchie, Project 39.5, Operation Plumbbob Report, WT-1504, Sept. 19, 1958. (Classified)
2. J. A. Auxier, J. S. Cheka, and F. W. Sanders, Projects 39.1 and 39.2, Operation Hardtack (Phase II) Report, WT-1725, March 1961. (Classified)

Chapter 3

RADIATION SOURCES

3.1 SOURCE-SUPPORT TOWER

The major radiation sources to be used in Operation BREN (the ORNL Health Physics Research Reactor and a 1200-curie Co^{60} gamma-ray source) will be supported at various heights above grade in the source hoist car of a 1500-ft tower constructed in Area 4 at NTS.

3.1.1 Basic Tower

The tower is located in Area 4, NTS, approximately 150 ft northwest of Building 4-300, which will serve as a shielded control room for operation of the reactor, the Co^{60} source, and the source hoist. The total height of the tower is 1527 ft 4 in. above grade. It is triangular in cross section, with three legs on 10-ft centers. The tower is braced by 36 guy wires that are anchored to the tower at six elevations and to the ground at two distances in three directions. Two parallel, vertical sets of six guy wires each extend east, southwest, and northwest from the respective sides of the tower. For reduction of wind loading, the members of the tower are of round stock wherever possible; the lower half of the tower will withstand 40 psf, and the upper half, 50 psf, corresponding to wind velocities of 110 and 119 mph, respectively. Lights are provided as approved by the Federal Aviation Agency and the Interstate Commerce Commission.

3.1.2 Source Hoist

The source hoist will consist of a car of approximately 18 ft in height and 4 ft 6 in. by 4 ft 6 in. in horizontal dimensions, a hoist "shaft" of rails and braces, and a double-hung single cable and hoisting engine.

The car will be divided into upper and lower compartments. The upper compartment will house the relay rack holding the preamplifiers of the HPRR and suitable shielding. The lower compartment will house the HPRR or the 1200-curie Co^{60} source. The upper plate of the HPRR will be bolted to plates provided in the ceiling of the lower compartment. For Operation BREN a square upper plate will be used which will have holes for four bolts of $^{3}/_{4}$ in. diameter. The shield containing the Co^{60} source will be supported on the floor of the lower compartment. Both compartments will be provided with gasketed doors to exclude dust and to protect the sources from unfavorable weather. For minimum perturbation of the radiation fields from the sources, the design of the lower compartment is uncluttered, and all major parts of the car are of aluminum. The car is provided with an automatic emergency braking system which, when tested, demonstrated the capability of stopping an elevator car within $^{3}/_{4}$ in. of free fall.

The two control cables of the HPRR will be supported on the tower to an elevation of 770 ft. One end of each of the two "traveling" cables will be suspended at the 770-ft level; the other ends will be attached to the hoist car. For prevention of excessive displacement of the cable loops by wind pressure, tension is maintained in the loops by means of two large weighted pulley wheels that are supported semi-independently by the cable loops and are guided by the rails that guide the hoist car. In addition, the traveling cables are automatically placed in, or removed from, guides made of flaps of heavy belting material (rubberized fabric).

The vertical speed of the hoist car will be approximately 60 ft/min. The maximum elevation will be approximately 1500 ft; the minimum elevation will be approximately 25 ft. An access platform will be constructed at a height of 31 ft.

The source hoist shaft will be composed of round $2\frac{3}{4}$-in.-diameter rails supported at 10-ft intervals by angle-iron braces that will surround the shaft. At three elevations (500, 1125, and 1500 ft), the braces will be modified to reduce perturbation of the radiation field. The shaft will be located on the east face of the tower.

A steel cable ($\frac{3}{4}$ in. in diameter) will be attached to the top of the tower and run through a sheave on the top of the hoist car, over a sheave at the top of the tower, down the inside of the tower, under a sheave at ground level, and to a drum and electric hoisting engine located southwest of the tower. Operation of the hoist will require the presence of a key and will be possible either from the control bunker (Building 4-300) or from a point at the foot of the tower. The key will be firmly attached to one of two interlock keys, both of which must be in position on the reactor control console before the main key switch can energize the reactor controls. Dual tandem limit switches will be provided at the top and bottom limits of travel of the car.

Some of the railings and other structures required by elevator codes are not used in the source hoist because they would perturb the radiation field and complicate source-hoist control. Therefore the source hoist will be considered to be a radiation-source positioner, *not* an elevator, and will *not* be used as an elevator by personnel.

3.1.3 Personnel Elevator

A standard Dresser-Ideco Company personnel elevator will be provided within the tower. It will operate independently of the source hoist. This elevator will be provided with standard dual tandem limit-switch controls at the top and bottom limits of its travel and with automatic emergency brakes like those of the source-hoist car, but with an additional manual control. The elevator will be controlled from the base of the tower or from the elevator car. Operation of the elevator will require the presence of a key that will be firmly attached to the second interlock key previously mentioned.

The elevator will be used for servicing lights or instruments mounted on the tower and will be the only elevator used by personnel.

3.2 ORNL HEALTH PHYSICS RESEARCH REACTOR

3.2.1 Description

(a) *Mechanical.* The ORNL Health Physics Research Reactor is shown in Figs. 3.1 to 3.5. (Details of operating procedures and interlock-system requirements are given in Sec. 3.2.4.)

(i) Core. The basic configuration and components of the HPRR are shown in Fig. 3.3. The core consists, essentially, of an annulus of U–10 wt.% Mo enriched to 93.14% in U^{235}; it has an 8-in. outside diameter, a 2-in. inside diameter, and a 9-in. height, and it surrounds a 2-in.-diameter stainless-steel central core. Attached to the steel core at the lower end is an 11.19-kg annulus of U–10 wt.% Mo which has an outside diameter of 3.375 in. and a height of 6.50 in.; this is the safety block. The basic uranium–molybdenum annulus consists of a stack of annular disks ranging in nominal thickness from $1\frac{5}{16}$ to $\frac{3}{16}$ in. [Four additional disks have been shipped to NTS (see Sec. 3.2.2).] Nominal thicknesses, weights of U^{235}, and order of appearance in the core are given in Table 3.1. The disks are held together with nine $\frac{3}{4}$-in.-diameter U–10 wt.% Mo bolts that thread into the bottom disk. Each bolt has a $\frac{3}{8}$-in.-diameter hole extending for $7\frac{9}{16}$ in. along its length, from bolt head downward, and seven of these holes are normally filled with 0.28-in.-diameter uranium–molybdenum plugs. The remaining two holes will contain

(Text continues on page 26.)

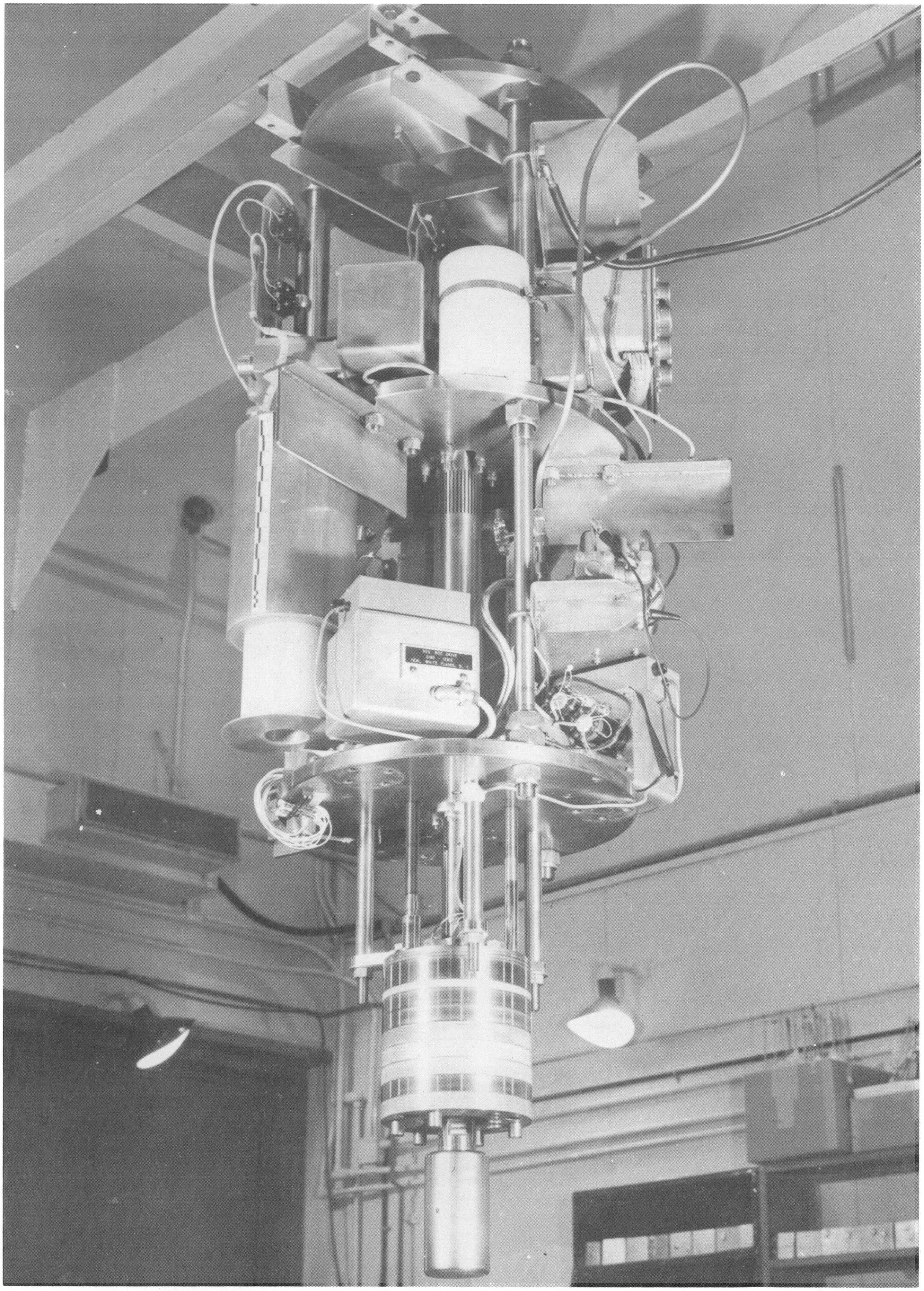

Fig. 3.1—ORNL Health Physics Research Reactor.

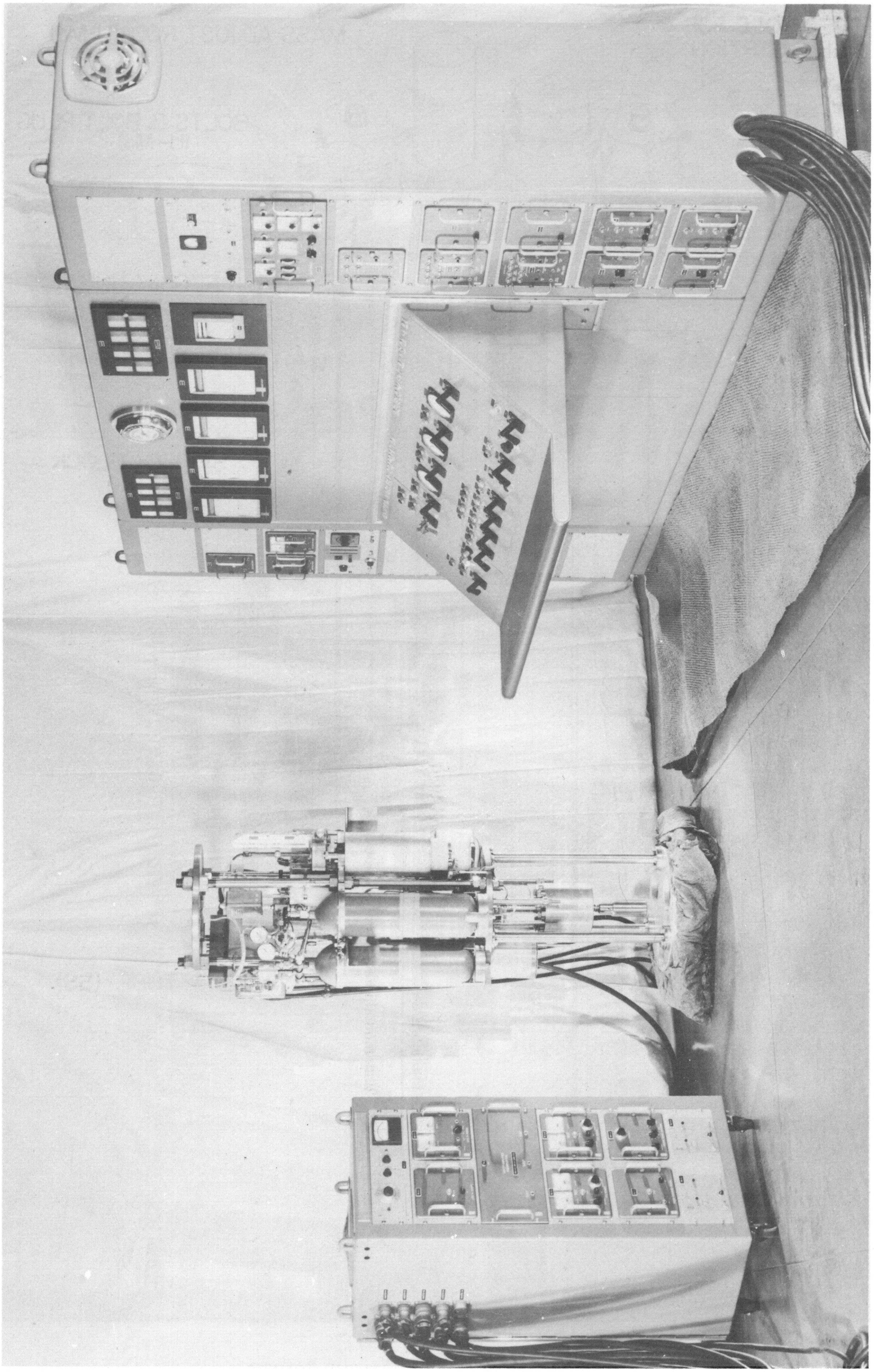

Fig. 3.2—HPRR with preamplifier cabinet and control console.

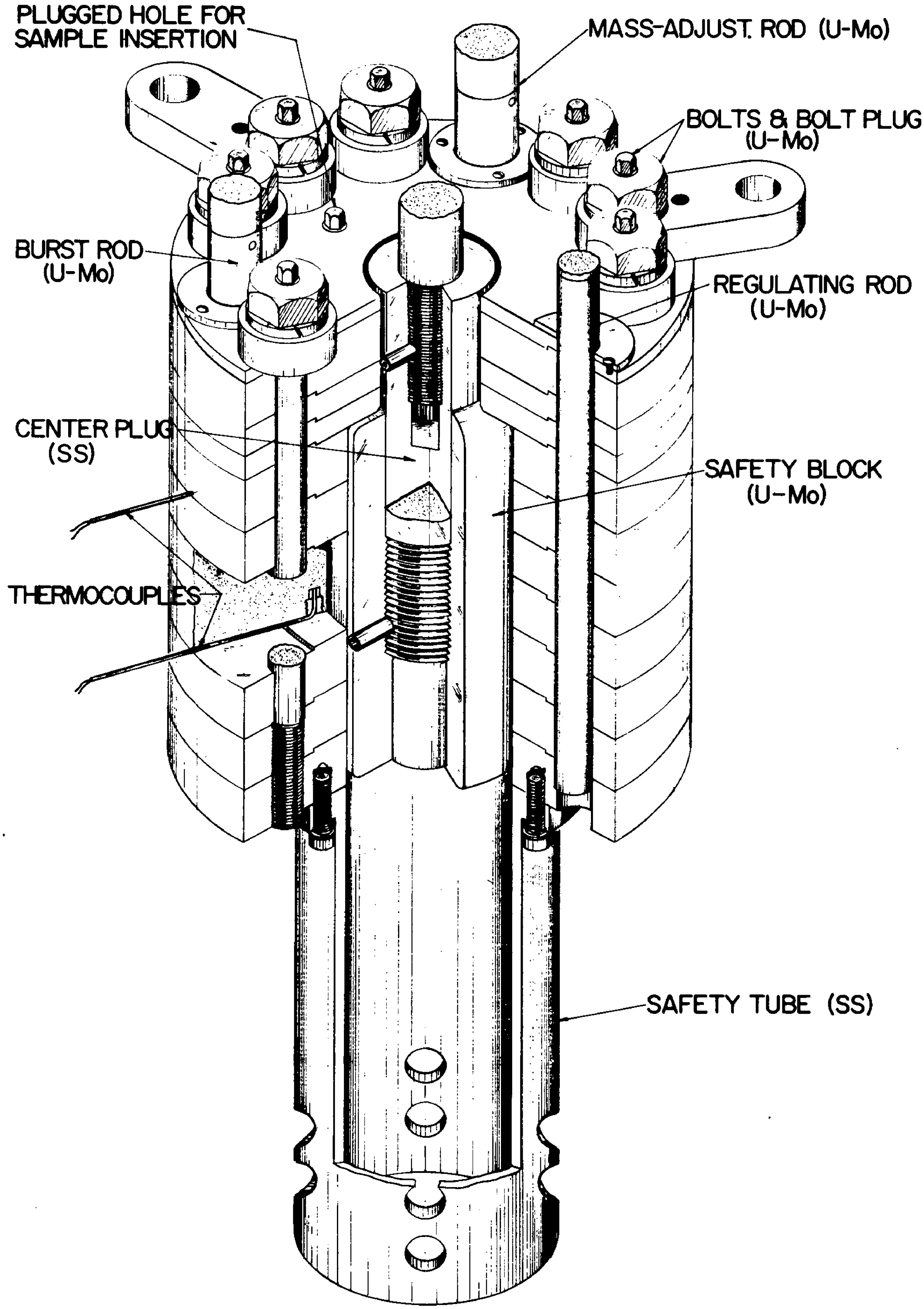

Fig. 3.3—HPRR core.

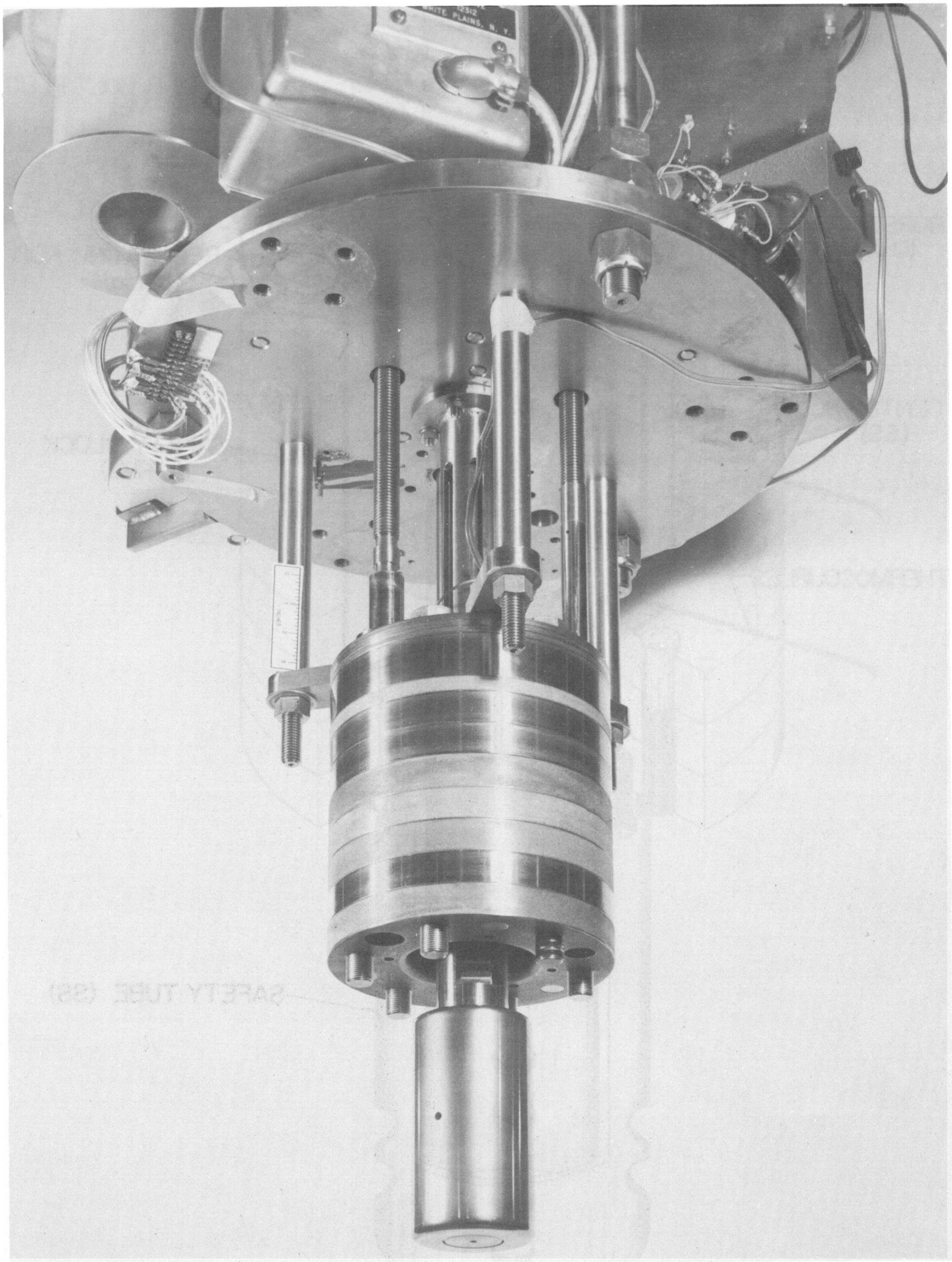

Fig. 3.4—HPRR core without safety tube.

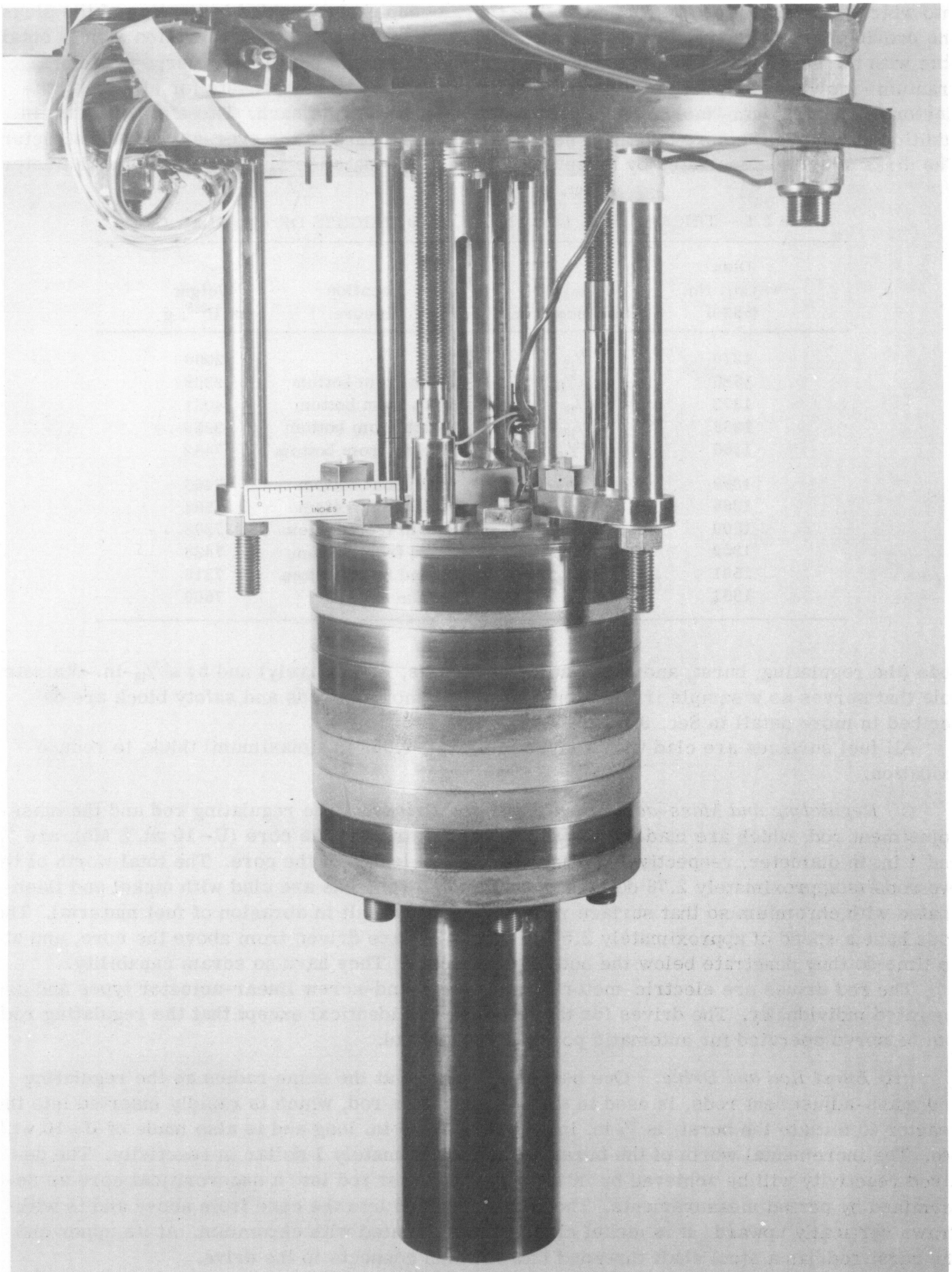

Fig. 3.5—HPRR core with safety tube.

stainless-steel plugs, extending to the center of the core, with holes drilled along their axes into which thermocouples are inserted. The thermocouples are welded to the tips of the plugs; one provides greater low-temperature monitoring sensitivity for burst operation than is obtainable with the scram-circuit thermocouples, and the other is a spare. The purpose of the uranium–molybdenum plugs is to provide shim adjustment to compensate for effects of reflectors. The uranium–molybdenum plugs are worth 15.6 cents each. Three of the bolts, in addition to holding the core together, serve to attach the core to the over-all reactor structure. The disks are also penetrated by three holes to accommodate the $^5/_8$-, $^3/_4$-, and 1-in.-diameter

Table 3.1 — THICKNESSES, LOCATIONS, AND WEIGHTS OF ANNULAR DISKS

Disk waybill No. (883-)	Nominal thickness, in.	Location in core	Weight of U^{235}, g
1370	$^3/_{16}$	Top	2000
1586	$1^5/_{16}$	Tenth from bottom	2308
1373	$^7/_{16}$	Ninth from bottom	4031
1368	$^7/_{16}$	Eighth from bottom	3956
1360	$^{15}/_{16}$	Seventh from bottom	7482
1399	$^{15}/_{16}$	Sixth from bottom	7465
1365	1	Fifth from bottom	7388
1500	$^{15}/_{16}$	Fourth from bottom	7308
1359	$^{15}/_{16}$	Third from bottom	7428
1501	$^{15}/_{16}$	Second from bottom	7318
1361	1	Bottom	7600

rods (the regulating, burst, and mass-adjustment rods, respectively) and by a $^5/_{16}$-in.-diameter hole that serves as a sample irradiation space. The movable rods and safety block are described in more detail in Sec. 3.2.1(d).

All fuel surfaces are clad with a layer of nickel, 0.005 in. (maximum) thick, to reduce oxidation.

(ii) Regulating and Mass-adjustment Rods and Drives. The regulating rod and the mass-adjustment rod, which are made of the same fuel material as the core (U–10 wt.% Mo), are $^5/_8$ and 1 in. in diameter, respectively. They run the full length of the core. The total worth of the two rods is approximately 2.78 dollars in reactivity. The rods are clad with nickel and flash-coated with chromium so that surface rubbing will not result in abrasion of fuel material. The rods have a speed of approximately 2.5 in./min. They are driven from above the core, and at no time do they penetrate below the bottom of the core. They have no scram capability.

The rod drives are electric-motor-driven worm-and-screw linear-actuator types and are operated individually. The drives for the two rods are identical except that the regulating rod can be servo operated for automatic power-level control.

(iii) Burst Rod and Drive. One burst rod, located at the same radius as the regulating and mass-adjustment rods, is used in the reactor. This rod, which is rapidly inserted into the reactor to initiate the burst, is $^3/_4$ in. in diameter and 9 in. long and is also made of U–10 wt.% Mo. The incremental worth of the burst rod is approximately 1 dollar in reactivity. The desired reactivity will be achieved by introducing the burst rod into a supercritical core as determined by period measurements. The rod is inserted into the core from above and is withdrawn vertically upward. It is nickel clad and flash-coated with chromium. At its upper end the burst rod has a steel shaft threaded into it which connects to the drive.

The drive for the burst rod is a 2-in.-bore pneumatic cylinder. The cylinder is mounted on the drive mounting plate above the core and inserts the rod into the reactor in approximately 40 msec. After the burst the rod is withdrawn in the upward direction by reversing the air pressure on the piston. Normally the burst rod is either at its IN or its OUT limit position. In the event of loss of air pressure or electric power, an automatic scram is initiated. For this emergency a spring is provided around the upper extension shaft. This spring is compressed during the insertion stroke and is capable of expelling the burst rod from the core.

(iv) Safety Block and Drive. The safety block, which is located in the center of the core, is $3\frac{3}{8}$ in. in diameter and $6\frac{1}{2}$ in. long. It is made of U−10 wt.% Mo and has a central stainless-steel plug of approximately 2 in. in diameter. For support the safety block is threaded and pinned onto the steel plug, which in turn is connected to the drive shaft by a threaded and pinned connection. The safety block is attached to the steel plug at its center so that it is free to expand in all directions without putting any appreciable net force on the drive shaft. It is supported from above and inserted into the core from below. It is withdrawn vertically downward. A clearance space of about $\frac{1}{16}$ in. is provided between the safety block and the core to minimize the possibility of jamming and to provide a gap for cooling-air flow. The safety block is normally either fully in or fully out of the reactor and is used as the primary mechanical shutdown device.

The safety block has an electromechanical drive similar to the regulating-rod and mass-adjustment-rod drives. The safety block is scrammed in the downward direction by de-energizing the holding magnet at the end of the lead screw. Also, a spring is provided around the drive shaft to decrease the scram time. The spring is capable of expelling the safety block about halfway out of the core if the reactor structure should accidentally become inverted. A shock absorber mounted on the drive shaft prevents damage to the safety block or structure during a scram.

For insertion the safety block is pulled upward into the core at a speed of approximately 1.9 in./min. Normally only two positions are permitted for the safety block: fully in or fully out. To assure that the safety block is always inserted to the same position, a positive stop in the form of a steel cylinder is provided. Electrical contacts on the steel cylinder provide a signal (used as an interlock) when the safety block is fully inserted. An overtravel spring assures positive seating since the force required to compress the spring will not result in magnet disengagement.

(v) Neutron Source and Drive. It is expected that a polonium−beryllium source with an activity of about 10^5 neutrons/sec will be used with the reactor. It must yet be demonstrated that this source is large enough for start-up and small enough for burst operation. The source is actuated by a $1\frac{1}{2}$-in.-bore pneumatic cylinder mounted above the reactor. During start-up operations the source is positioned about 2 in. away from the reactor surface. The source can be withdrawn into a borated paraffin−borated polyethylene shield when desired.

(vi) Structural Arrangement. The structural arrangement of the reactor is shown in Fig. 3.1. The outside dimensions of the unit are about 6 ft 3 in. long by 2 ft 8 in. in diameter. All structural material is stainless steel. The core is suspended from the structure by three mounting lugs. The drive mounting plate, approximately 8 in. from the top of the core, supports the drives for the regulating rod, mass-adjustment rod, burst rod, and source. This steel plate is attached to the top plate by means of three steel rods. The intermediate plate serves to make the entire structure more rigid and supports the drives for the safety block and the fission chamber. The top plate* is designed to permit secure mounting of the reactor onto the reactor hoist car. The structure around the sides and bottom of the core has been minimized so that space will be available for experiments. A steel safety tube is attached by screws to the bottom of the core to prevent a serious accident in the event that the reactor is dropped onto a reflecting surface. This tube encloses the volume occupied by the safety block when in its out position. With the tube in place, dropping the reactor onto a hard surface could not force the safety block into the core (Figs. 3.3 and 3.5). The tube is sufficiently strong that, if great force were applied, the core would disassemble rather than collapse the tube.

The gas supply for the burst-rod and neutron-source drives is located in two tanks, one on the reactor structure and the other at some convenient place on the hoist. One is a high-pressure (about 2000 psi) reservoir that feeds into the second tank, which is maintained at operating pressure (70 psi).

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*A replacement for the top plate is shown in Fig. 3.1.

(b) *Instrumentation.** The instrument diagrams are shown in Figs. 3.6 and 3.7. Five main channels of instrumentation provide information to the control system. These channels are the count rate, Log N period, level safety, level control, and temperature safety.

(i) Count-rate Channels. There are two count-rate channels, one fed by a fission chamber and the other by a BF_3 chamber. Since count-rate instruments are limited to 5 decades, provision is made to withdraw the fission chamber to a position of lower sensitivity as the neutron-flux level increases, giving it a range of approximately 6×10^6. Pulses from the fission chamber are fed through linear preamplifiers and amplifiers to a logarithmic count-rate meter, recorder, and scaler. The BF_3 channel also feeds a recorder and a scaler. The count-rate-meter recorder is interlocked with the safety-block, mass-adjustment-rod, regulating-rod, and burst-rod control circuits to prevent any addition of reactivity during start-up until a minimum reading of approximately 2 counts/sec is obtained.

(ii) Log N–Period Channel. The useful range of the Log N and period instruments extends to approximately 6 decades below full power. The level signal is fed from a noncompensated ionization chamber through a Log N–period amplifier and preamplifier to Log N and period recorders. The amplifier initiates a trip signal to the fast scram bus in the event of an excessively short period (less than 1 sec). Recorders are interlocked with the safety-block, mass-adjustment-rod, regulating-rod, and burst-rod control circuits to prevent any addition of reactivity when the period is less than 5 sec.

(iii) Level-safety Channels. Two safety channels driven by ion chambers are provided which activate a scram signal upon detection of excessive neutron levels. It is the function of these channels to scram the reactor when the operating power level reaches either 1.5 or 15 kw in conjunction with low- or high-power operation, respectively. Receipt of a 50-μa signal from the ion chamber by a resistor in the level-safety-channel preamplifier will initiate a scram. Initially the ion-chamber location, shielding, and resistor size in the safety amplifier will be such that a 50-μa signal corresponds to a 1.5-kw power level. For subsequent higher power operation, the resistor size will be changed so that a 50-μa signal corresponds to a 15-kw power level. Scram results in rapid de-energizing of the safety-block-magnet and burst-rod-solenoid valves when the Σ-bus voltage is raised by a high-level or period trip signal.

(iv) Level-control Channel. An accurate linear indication of neutron-flux level extending from full power down approximately 6 decades is accomplished by the use of a multirange micromicroammeter to measure the current from a noncompensated ionization chamber. In addition, this channel furnishes the signal necessary for automatic level control.

(v) Temperature-safety Channel. Reactor-core temperatures are measured at two locations by thermocouples with connections to a multipoint recorder. Excessive temperature sensed at one or both thermocouples initiates a slow scram. A third thermocouple, which is connected to a second, more sensitive recorder, supplies temperature-variation information for burst operation but is not in the safety system.

(vi) Scrams. A slow scram is actuated by a relay type circuit that de-energizes the safety-block-magnet and burst-rod-solenoid control valves. A slow scram is initiated by low gas-supply pressure and manual push buttons, as well as by high reactor temperature. A fast scram, on the other hand, is initiated by the level and period signals previously described and is actuated by an electronic rather than a relay type circuit. The following conditions will result in a scram.

1. Neutron level that is too high.
2. Period that is too short.
3. Low air pressure for drives.
4. Loss of electric power.

*Some safety interlock circuits designed for use at ORNL are not applicable to NTS and will not be used.

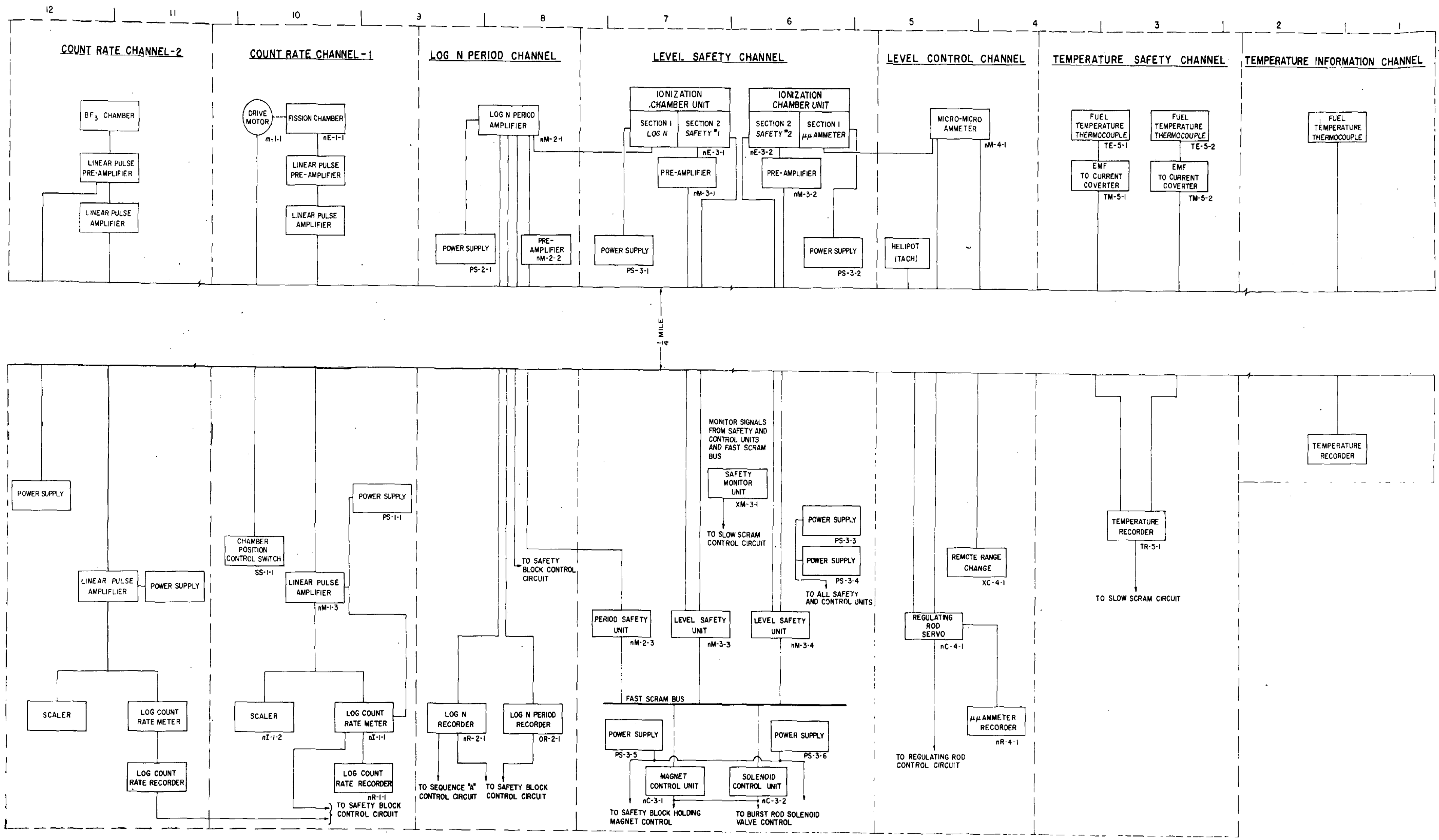


Fig. 3.6—HPRR instrument block diagram.

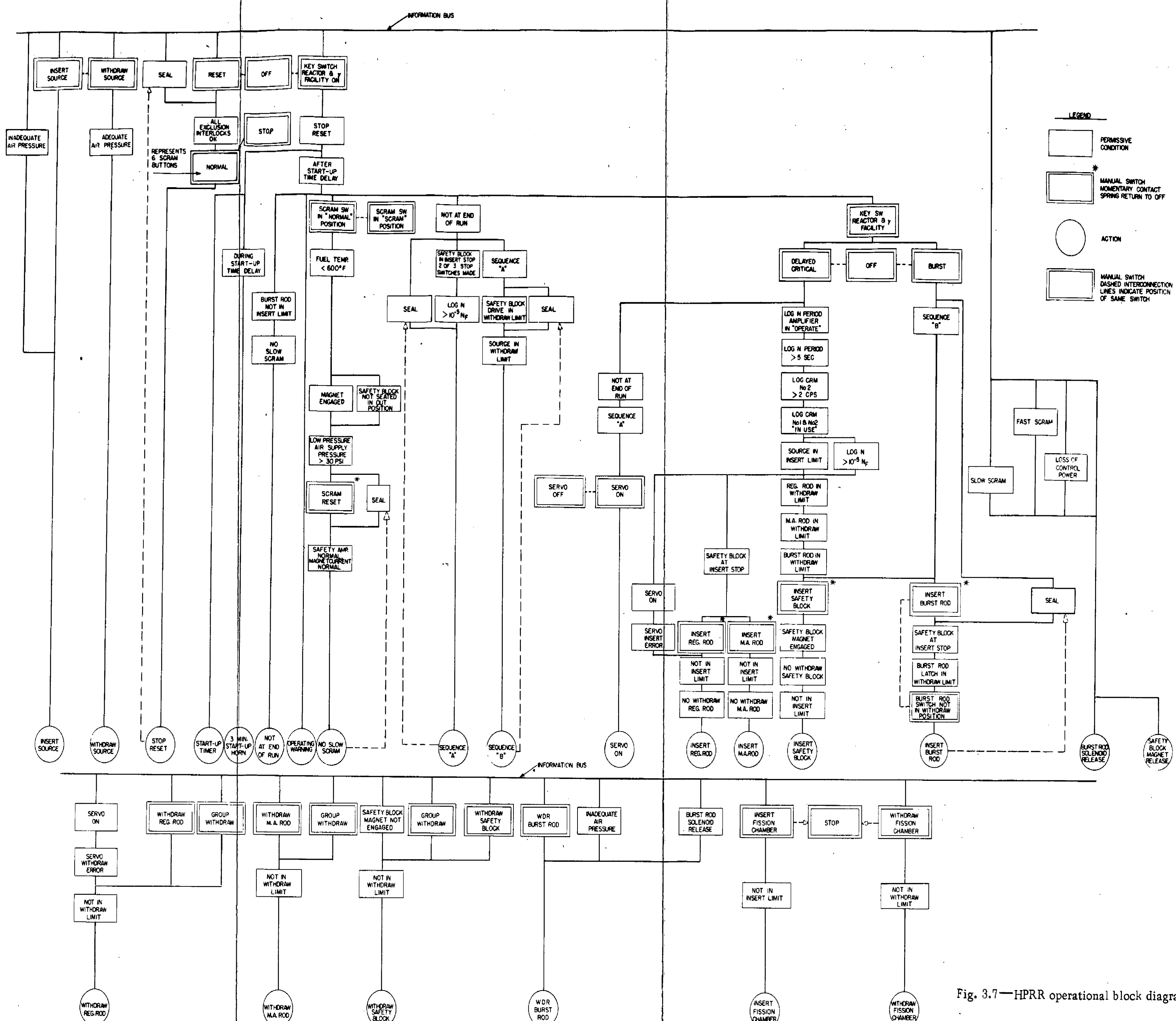


Fig. 3.7—HPRR operational block diagram.

5. Excessive core temperature.
6. Manual scram.
7. Scram buttons that have become depressed.
8. Excessively high magnet current.

(vii) Annunciator System. The following conditions will cause the annunciator system to become actuated.

1. Safety-monitor trouble.
2. Stop reset that is open.
3. Log N period of less than 5 sec.
4. Fast scram.
5. Fuel temperature that is too high.
6. Air pressure that is too low in low-pressure cylinder.
7. Air pressure that is too low in high-pressure cylinder.
8. Slow scram.
9. Facility stop button that is open.

The operating sequence for the annunciator system is given in Table 3.2.

Table 3.2—ANNUNCIATOR-SYSTEM OPERATING SEQUENCE

Step	Condition	Red light	White light	Audible alarm
1	Normal condition	Dim	Dim	Off
2	Abnormal condition, before acknowledgment	Bright	Bright	On
3	Abnormal condition, after acknowledgment	Bright	Off	Off
4	Normal condition returns, before reset	Off	Bright	Off
5	Normal condition, after reset	Dim	Dim	Off

Note 1:
If, after step 2, normal condition returns before the ACKNOWLEDGE button is operated, the annunciator remains in condition 2 until the ACKNOWLEDGE button is operated, after which the annunciator goes to step 4.

Note 2:
If, after step 4, abnormal condition returns before the reset button is operated, the annunciator returns to step 2.

(viii) Interlock System. The interlock system used to control the assembly sequence has been designed so that, with a single exception, only one component can be inserted at any given time and the other components must be at the proper limit position before this movement can be made. The exception is that the mass-adjustment rod and the regulating rod can be inserted simultaneously. This is prohibited by the operating procedure and is controlled administratively.

(ix) Automatic Power-level Control Servo. The HPRR was designed for use at ORNL, and Operation BREN will use only a minor fraction of its operating life. Some of the objects irradiated at ORNL, e.g., burros, add enough reactivity to necessitate reactor operation with the mass-adjustment rod partially withdrawn and the regulating rod partially inserted and controlled by the automatic power-level control servo. Frequent changes in the type of objects being irradiated make it desirable to have enough range of reactivity in the mass-adjustment rod to minimize shim changes of reactivity.

(c) *Performance Characteristics.* The HPRR has been designed to achieve bursts yielding 1×10^{17} fissions. Preliminary analysis of the experimental data, subject to revisions of a few per cent, yields the following associated characteristics.

Fission yield from burst	1.1×10^{17}
Initial reactivity insertion	1.11 dollars
Initial reactor period	18 μsec
Burst width at half maximum	50 μsec
Leakage neutrons from burst	1.4×10^{17}
Total equilibrium temperature coefficient	0.30 cent/°C 0.17 cent/°F
Maximum temperature rise	572°F
Average temperature rise	302°F

The reactor can also be operated in the steady-power-level mode. Continuous operation is possible at power levels of 1 kw or less. At higher power levels, up to 10 kw, operating time is governed by the temperature of the core. A limit of 600°F has been selected as the maximum allowable core temperature. Steady-power-level operation at 1 kw-hr entails approximately 1×10^{17} fissions. The number of neutrons leaking from the core per kilowatt hour is therefore approximately 1.3×10^{17}.

(d) *Design Characteristics.* The following tabulation presents the design characteristics of the HPRR.

Reactor diameter	8 in.
Reactor height	9.063 in. (23.0 cm)
Fuel material	U–10 wt.% Mo
Enrichment	93.14%
Critical mass with burst rod out	114.8 kg of U–Mo 96.3 kg of U

Regulating Rod

Travel	Full core length
Weight	0.726 kg of U–Mo 0.658 kg of U
Diameter	0.625 in.
Length	9 in.
Reactivity worth	83 cents
Type of drive	Electric motor
Speed	2.5 in./min

Burst Rod

Weight	1.065 kg of U–Mo 0.965 kg of U
Diameter	0.750 in.
Length	9 in.
Stroke	7.476 in.
Reactor worth in stroke	1 dollar
Acceleration	~30 g
Type of drive	Pneumatic cylinder
Insertion time	~0.04 sec

Insert Plugs

Bolt-insert-plug weight	120 g of U
Bolt-insert-plug worth	15.6 cents
Glory-hole-plug weight	113 g of U
Glory-hole-plug worth	19.9 cents

Neutron Source

Material	Po–Be
Strength	10^5 neutrons/sec
Type of drive	Pneumatic cylinder

Mass-adjustment Rod

Travel	Full core length
Weight	1.928 kg of U–Mo 1.746 kg of U
Diameter	1 in.
Length	9 in.
Reactivity worth	1.95 dollars
Type of drive	Electric motor
Speed	2.5 in./min

Safety Block

Travel	8.5 in.
Weight	11.19 kg of U–Mo 10.109 kg of U
Diameter	3.375 in.
Length	6.500 in.
Reactivity worth	20 dollars
Reactivity worth per centimeter for first 2.5 cm	80 cents
Initial acceleration	2.28 g
Type of drive	Electric motor
Speed	1.9 in./min
Release time of electromagnet	0.004 sec
Drop-out time for removing 1 dollar of reactivity with reactor-safety system	0.038 sec

(e) *Critical Experiments.* A complete set of critical experiments of the types indicated in this section has been performed. These tests were preceded by a series of mechanical and electrical check-outs and tests of the equipment. The reactor control panel and interlock system were tested, and all instrumentation was calibrated. Tests measuring assembly speed and positioning accuracy of the mass-adjustment rod, regulating rod, and safety block were performed.

(i) Preliminary Critical Experiment. Prior to the final machining of the pieces, a series of subcritical assemblies and critical experiments were conducted at the ORNL Critical Experiments Facility.[1] These assemblies simulated the final parts and were useful in setting the anticipated critical mass of the reactor core. The tests consisted in the determination of the critical heights of the following configurations.

1. U – 10 wt.% Mo cylinder having an outside diameter of 8 in.
2. U – 10 wt.% Mo cylinder having an outside diameter of 8 in. with an annulus having an inside diameter of 2 in.
3. Annulus, having an outside diameter of 8 in. and an inside diameter of 2 in., filled with a 2-in.-diameter stainless-steel plug.

These assemblies also provided preliminary information on void coefficient and fission-rate distributions as well as on the effects of Plexiglas reflectors of various thicknesses which can be extrapolated to determine the effects of personnel and experimental samples in the vicinity of the core during both experiments and maintenance operations.

(ii) Final Critical Experiment. After criticality was obtained with the final machined assembly, the reactivity worths of the mass-adjustment rod, regulating rod, burst rod, and safety block were calibrated over their full lengths of travel, and the resulting calibration curves vs. position for the rods are shown in Fig. 3.8.

After calibration of the movable fuel pieces, other characteristics of the reactor were determined experimentally, e.g., temperature coefficient, power distribution, neutron leakage, and reactivity effects of various materials.

3.2.2 Shipping Methods

The reactor has been transported from ORNL to NTS and will be returned to ORNL by trailer truck. The reactor structure, fuel, neutron source, preamplifier rack, and control console, with other ORNL equipment, comprised a full trailer load, which was loaded and unloaded within the regulated areas of ORNL and NTS.

The core of the reactor was disassembled at ORNL, and the parts were then shipped to NTS in fissionable-material containers of the type used by the Union Carbide Nuclear Company Y-12 Plant. These so-called "bird cages" are cylindrical containers with bolt-on tops. The cylinders were fabricated from $\frac{1}{4}$-in.-thick steel. Five of them have 10-in. diameters and $5\frac{1}{4}$-in. heights; three have $5\frac{1}{2}$-in. diameters and 12-in. heights. Each cylinder is supported in the center of a 20- by 20- by 20-in. steel-angle framework, which in turn has $\frac{1}{4}$-in.-thick sheets of plywood attached to each face. Each container is lined with $\frac{1}{2}$ in. of lead. The purpose of the lead shielding and plywood covering is to make the containers safe from the standpoint of gamma radiation.

As stated in 3.2.1(a), four additional disks were shipped to NTS along with those comprising the basic core. These disks have thicknesses of $\frac{7}{16}$, $\frac{3}{16}$, $\frac{1}{8}$, and $\frac{1}{16}$ in. If this stack of uranium – molybdenum plates, having a total thickness of $\frac{13}{16}$ in., is added to the regular reactor core, their worth is conservatively estimated to be less than 6 dollars, compared to the approximately 20-dollar worth of the safety block. This estimate of worth is based on previous critical experiments with the $8\frac{3}{4}$-in.-high core.

Table 3.3 lists all uranium – molybdenum pieces shipped to NTS, giving the cage number in which each piece was shipped, a description of each piece, the weight of uranium in each piece, and the total weight of uranium in each cage.

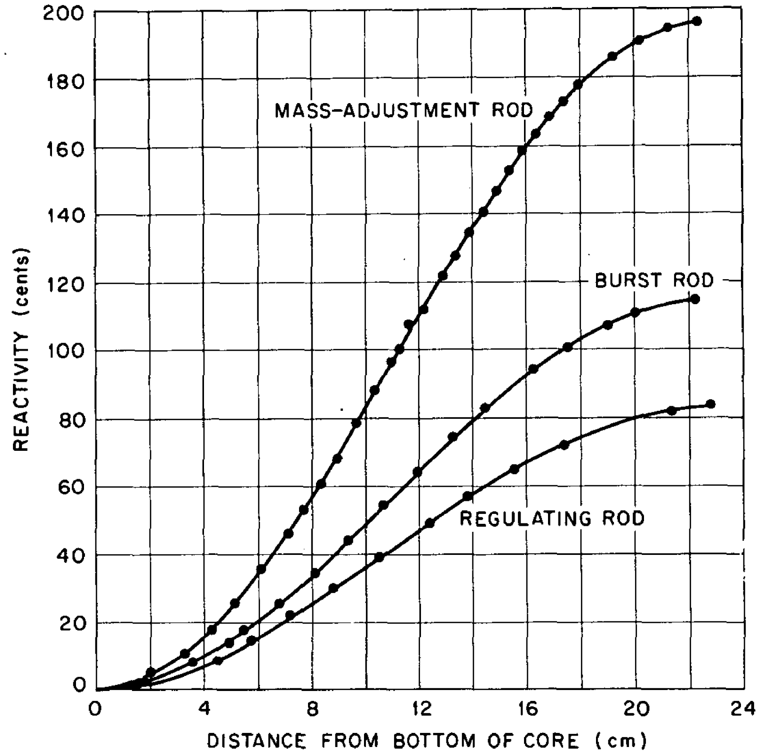


Fig. 3.8 — Reactivity values of HPRR control rods. Values based on the linear portion of the curves are as follows:

Rod	Diameter, cm	Worth, cents/cm
Mass-adjustment	2.54	13.3
Burst	1.91	7.7
Regulating	1.59	4.1

3.2.3 Initial Core Assembly

Because the critical conditions for the uranium–molybdenum pieces used in critical experiments at the ORNL Critical Experiments Facility are known, the checkoff procedure given in Fig. 3.9 will be used to reassemble the same pieces at NTS. The assembly procedure is graphically illustrated in Fig. 3.10.

The reactor superstructure will be secured to the hoist before assembly. The basic core will be assembled on a supporting device capable of raising the disks to a position in which they can be attached to the reactor superstructure. The supporting device consists of a hydraulic jack mounted on a steel base about 24 in. square, which in turn rests on four roller wheels for mobility. An aluminum "cup" having an inside diameter of about $8\frac{1}{8}$ in., an over-all height of $\frac{3}{8}$ in., and a base thickness of $\frac{1}{8}$ in., is rigidly attached to the top of the 1.5-in.-diameter lifting cylinder. The bottom disk will sit securely in the cup, with the remaining disks stacked over it. The disks are not flat but are shaped so that the core is divided into an upper section and a lower section, and disks from one section will not mate with disks from the other section. The possibilities for accidental rearrangement of the disks, compared to the critical experiments, are (1) interchange of two lower $\frac{15}{16}$-in.-thick pieces, (2) interchange of two upper $\frac{15}{16}$-in.-thick pieces, and (3) interchange of two upper $\frac{7}{16}$-in.-thick pieces. The interchange of any two disks as described would not alter the reactivity of the core more than a few cents since the difference in uranium content in any two interchangeable disks is not more than 150 g of uranium. It is desirable, however, to position thermocouples at their former locations and to locate the $\frac{7}{16}$-in.-thick disk with rounded edges below the other one so as not to restrict air flow through the center of the core.

Table 3.3 — URANIUM-MOLYBDENUM PIECES SHIPPED TO NTS

Cage No.	Piece	Thickness, in.	Waybill No. (883-)	Weight of uranium, g	Total weight of uranium per cage, g
49-1173	Disk	$1\frac{5}{16}$	1586	13,212	19,684
	Disk	$\frac{7}{16}$	1373	4,326	
	Disk	$\frac{3}{16}$	1370	2,146	
49-1170	Disk	$\frac{15}{16}$	1360	8,030	18,545
	Disk	$\frac{7}{16}$	1368	4,247	
	Disk	$\frac{7}{16}$	1400	4,395	
	Disk	$\frac{1}{8}$	1371	1,247	
	Disk	$\frac{1}{16}$	1376	626	
49-1172	Disk	1	1365	7,932	15,941
	Disk	$\frac{15}{16}$	1399	8,009	
49-1176	Disk	$\frac{15}{16}$	1500	7,846	15,817
	Disk	$\frac{15}{16}$	1359	7,971	
49-1187	Disk	$\frac{15}{16}$	1501	7,856	18,004
	Disk	1	1361	8,162	
	Disk	$\frac{3}{16}$	1362	1,986	
47-26	Bolts (nine)		1503 to 1508 1524 to 1526	9,223	10,302
	Inserts (nine)		1533 1512 to 1519	1,079	
47-42	Mass-adjustment rod		1521	1,746	3,680
	Burst rod		1523	965	
	Regulating rod		1522	658	
	Glory-hole plug		1578	113	
	Glory-hole plug		1594	130	
	Plug ($\frac{1}{2}$ length)		1598	68	
47-25	Safety block		1531	10,109	10,109
47-57	Glory-hole plug		1786	131	4,088
	Glory-hole plug		1811	128	
	Glory-hole plug		1812	128	
	Glory-hole plug		1813	129	
	Glory-hole plug		1814	129	
	Glory-hole plug		1815	129	
	Plug ($\frac{1}{2}$ length)		1752	56	
	Plug ($\frac{1}{2}$ length)		1816	56	
	Plug ($\frac{1}{2}$ length)		1817	57	
	Plug ($\frac{1}{2}$ length)		1818	56	
	Bolt		1819	1,030	
	Bolt		1821	1,029	
	Bolt		1832	1,030	
			Total weight of uranium shipped	116,170 g	

A set of special tools has been designed and built which will allow complete assembly or disassembly of the core while the persons doing the work remain approximately 4 ft from the core. These tools, along with the check list in Fig. 3.9 will be used to ensure the proper assembly of the core with a minimum of exposure to personnel.

HPRR CORE-ASSEMBLY CHECK LIST

____ 1. Install source.
____ 2. Screw burst rod and attached adapter onto burst-rod shaft.
____ 3. Pin adapter to burst-rod shaft.
____ 4. Screw mass-adjustment rod and attached adapter to mass-adjustment-rod drive shaft.
____ 5. Pin mass-adjustment-rod adapter to shaft.
____ 6. Screw regulating rod and attached adapter to regulating-rod drive shaft.
____ 7. Pin regulating-rod adapter to shaft.
____ 8. Place bottom core disk (1361) in support cup.
____ 9. Place lower $^{15}/_{16}$-in.-thick disk (1501) on bottom piece.
____ 10. Place second lower $^{15}/_{16}$-in.-thick disk (1359) on top of first.
____ 11. Place $^{15}/_{16}$-in.-thick disk (1500) containing lower thermocouple on assembly.
____ 12. Place center disk (1365) on assembly.
____ 13. Place $^{15}/_{16}$-in.-thick disk (1399) containing upper thermocouple on assembly.
____ 14. Place upper $^{15}/_{16}$-in.-thick disk (1360) on assembly.
____ 15. Place $^{7}/_{16}$-in.-thick disk (1368) having rounded inner edges on assembly.
____ 16. Place $^{7}/_{16}$-in.-thick disk (1373) without rounded edges on assembly.
____ 17. Place $1^{5}/_{16}$-in.-thick disk (1586) on assembly.
____ 18. Place top $^{3}/_{16}$-in.-thick disk (1370) on assembly.
____ 19. Place stainless-steel guide tube in burst-rod hole, and bolt on guide-tube holding ring.
____ 20. Place stainless-steel guide tube in regulating-rod hole, and bolt on guide-tube holding ring.
____ 21. Place stainless-steel guide tube in mass-adjustment-rod hole, and bolt on guide-tube holding ring.
____ 22. Place supporting lugs on appropriate holes in top disk.
____ 23. Insert the nine bolts in bolt holes.
____ 24. Place thermocouple in remaining bolt-insert holes.
____ 25. Place seven bolt inserts in bolt-insert holes.
____ 26. Screw glory-hole plug in top of glory hole.
____ 27. Withdraw burst rod, mass-adjustment rod, and regulating rod to their withdrawn limits.
____ 28. Raise safety-block magnet to insert position.
____ 29. Jack core into position and fasten with three nuts (to be done with the regulating rod, mass-adjustment rod, and burst rod in the withdrawn positions).
____ 30. Place safety-block spring on safety-block supporting shaft with safety-block drive in the fully withdrawn position.
____ 31. Screw safety block, which is already pinned to 2-in.-diameter stainless-steel cylinder, onto safety-block shaft.
____ 32. Pin safety-block holder to safety-block shaft.
____ 33. Bolt safety-block guard to bottom disk of core.
____ 34. Lock burst rod "in."

Fig. 3.9—HPRR core-assembly check list.

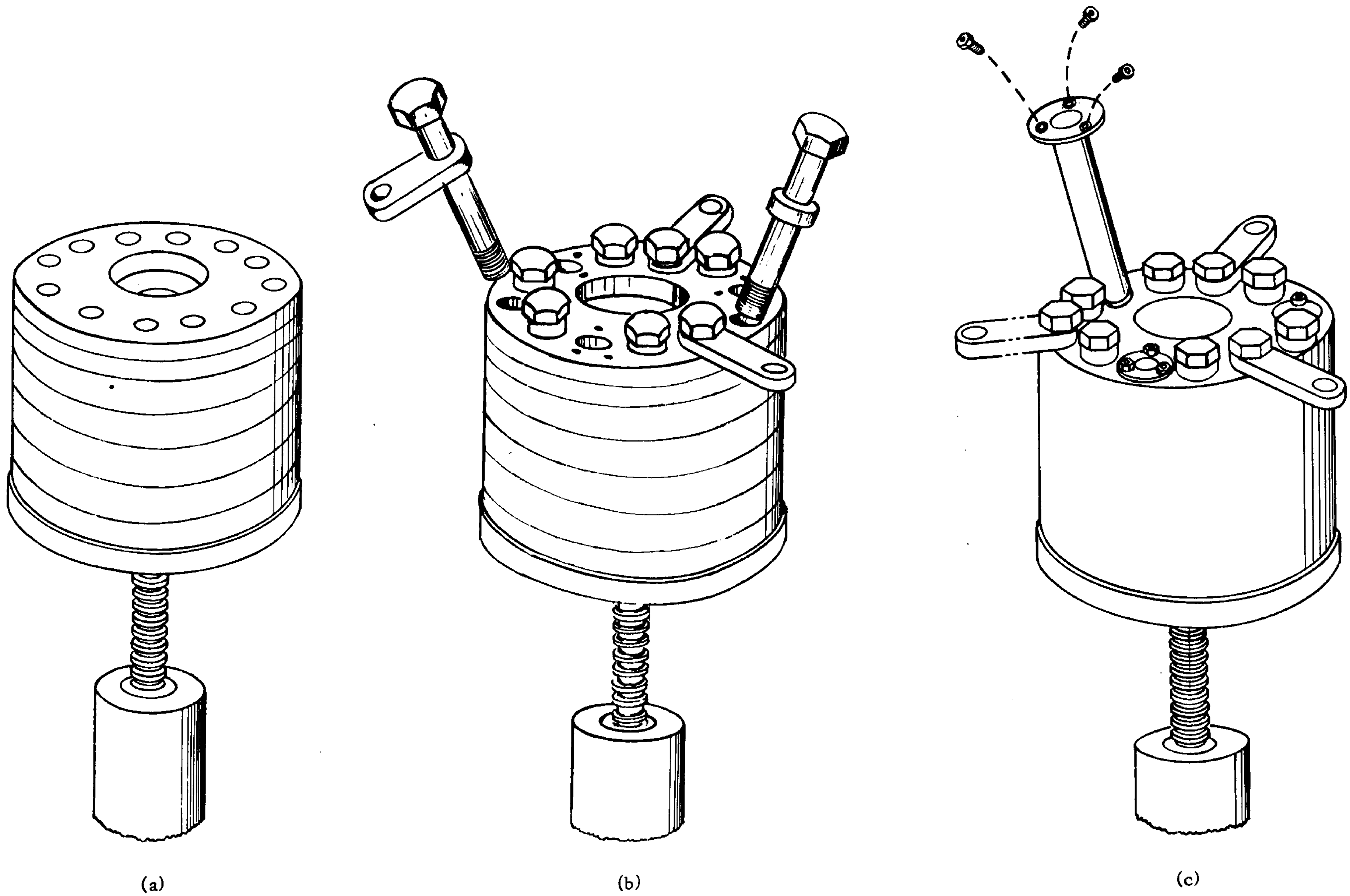
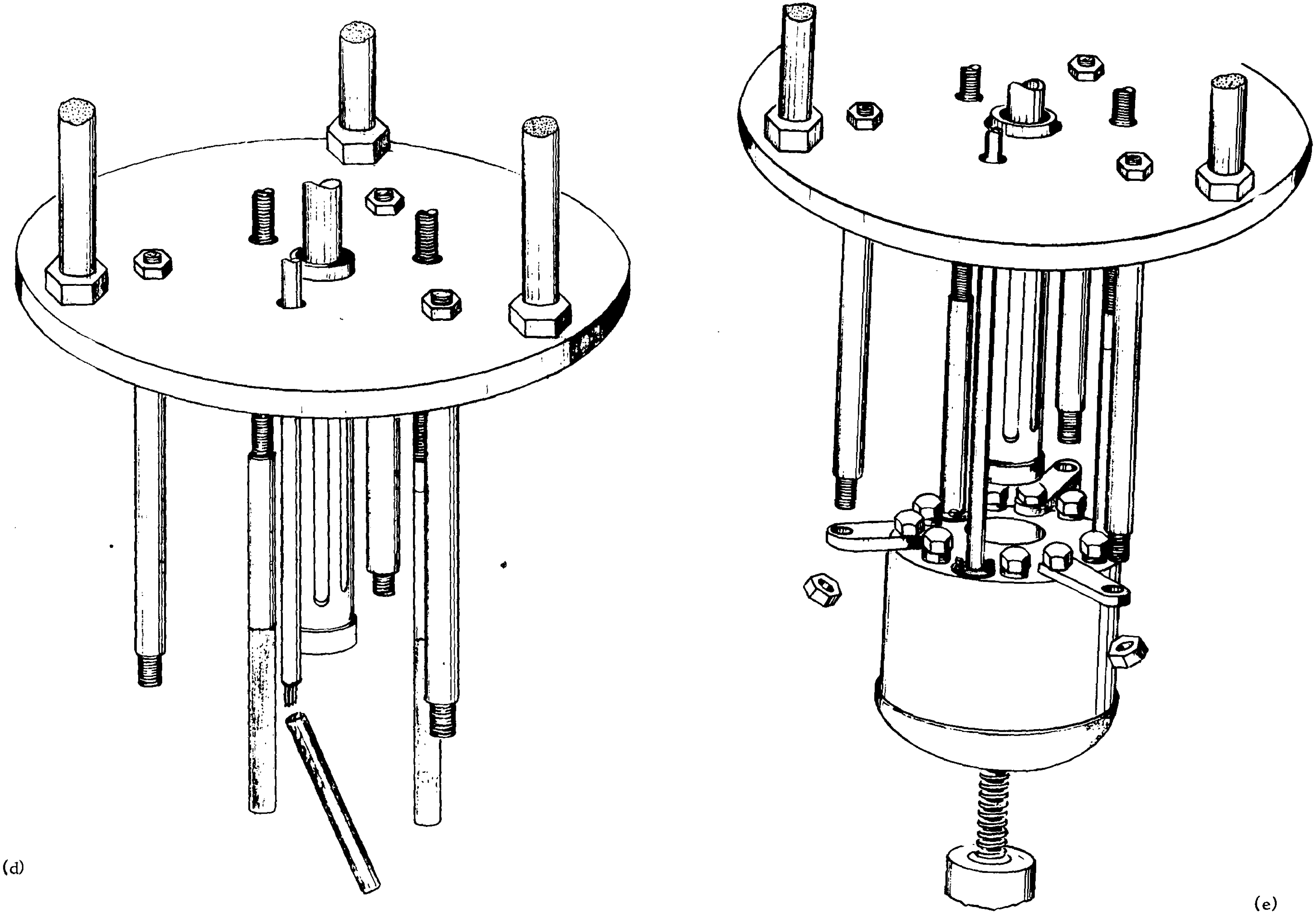


Fig. 3.10—HPRR assembly procedure (continues on page 38).



(d)

(e)

Fig. 3.10—HPRR assembly procedure (continued).

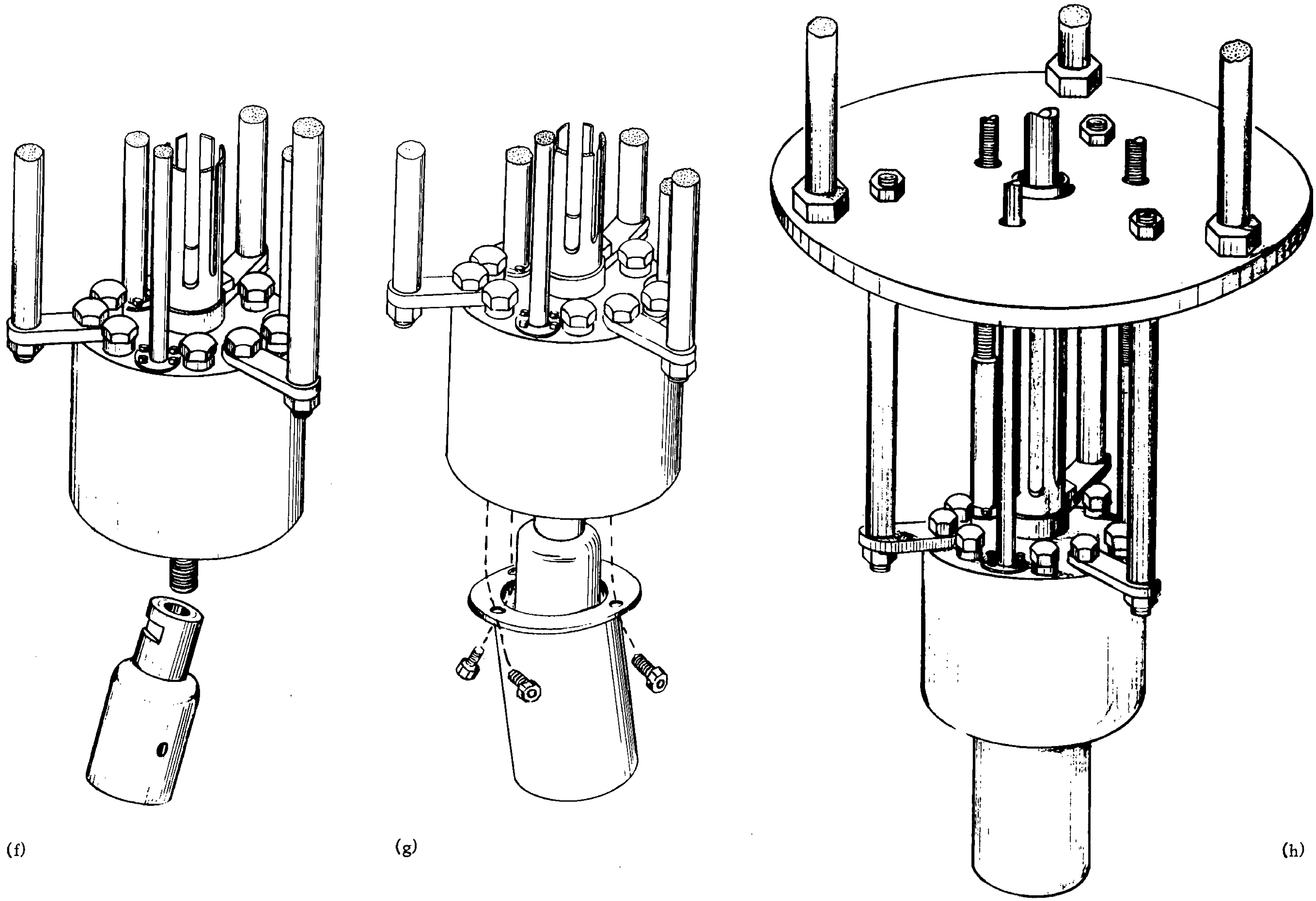


Fig. 3.10—HPRR assembly procedure (continued).

3.2.4 Operating Procedures

(a) *Introduction.* The reactor supervisor is responsible for the operation, maintenance, and safety of the reactor. In the event of an emergency connected with the reactor, the reactor supervisor has the responsibility and authority to take the proper remedial action. Any reactor abnormality that has not been and cannot be corrected by the controls on the reactor console is defined as an emergency. In an emergency the reactor operator is empowered to act as the reactor supervisor if the reactor supervisor is not readily available.

The reactor must be recognized as a potentially dangerous device, and either the Technical Director, who has the responsibility for personnel safety, or the reactor supervisor, who is primarily responsible for safe operation of the reactor, can suspend operation of the reactor if either believes that it is not safe from a nuclear standpoint. Unsafe conditions may arise due to unfavorable weather, inoperative reactor equipment, changing reactor-core temperature, or other conditions that can be recognized only at the test site. No reactor assembling, check-out, operation, or maintenance shall be carried out under the duress of a time limit.

The reactor supervisor will, if both he and the Technical Director agree that safe operating conditions prevail, operate the reactor at the direction of the Technical Director according to his previously distributed written plan. The reactor supervisor will not operate the reactor unless directed to do so by the Technical Director.

The reactor supervisor will be in charge of all maintenance on the reactor. He will work together with the Rad-Safe Officer whenever it is necessary to remove or replace the core or to do maintenance work when the core is in place to ensure that neither he nor others who assist him receive radiation exposures in excess of permissible levels.

The reactor supervisor will also call upon the Rad-Safe Officer as needed for determining radiation levels during routine reactor check-out, hoisting, and operation (see Fig. 3.11).

The reactor operator will report to the reactor supervisor. Both will be at the console at the time of reactor start-up, and at least one of them will be at the console at all times while the reactor is in operation.

The instrument engineer will report to the reactor supervisor and will be responsible for maintaining the reactor instrumentation and electrical system in proper working order. He will be responsible for maintaining a supply of spare parts, test equipment, tools, and other items needed in performing his duties.

(b) *Steady-state Operation.* The reactor operator* will proceed according to the steady-state-operation check list given in this section; he will annotate each step, where applicable, in the spaces provided on the BREN HPRR Reactor Check-out Sheet (Fig. 3.12) and the BREN HPRR Reactor Steady-power Log Sheet (Fig. 3.13). The reactor supervisor, with the concurrence of the reactor operator and the instrument engineer, may make minor amendments to this procedure by describing the changes in the HPRR Log Book (Fig. 3.14). Significant deviations, however, will require prior approval of the appropriate ORNL committee.

The items listed in Groups A, B, C, D, and G will normally be performed only once per shift, and items in E and F will be performed as needed.

A single run number will normally be used per shift, with appended letters used to define the various increments of reactor operation during the shift, e.g., 26A, 26B, and 26C (see Fig. 3.13).

The operator will carefully describe in the HPRR Log Book his observations during reactor check-out and operation. He will also keep careful notes in this log book regarding repairs, maintenance, or alterations made to the reactor.

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*The term "reactor operator" refers to the person performing the work. He may have the title "reactor supervisor" or "reactor operator."

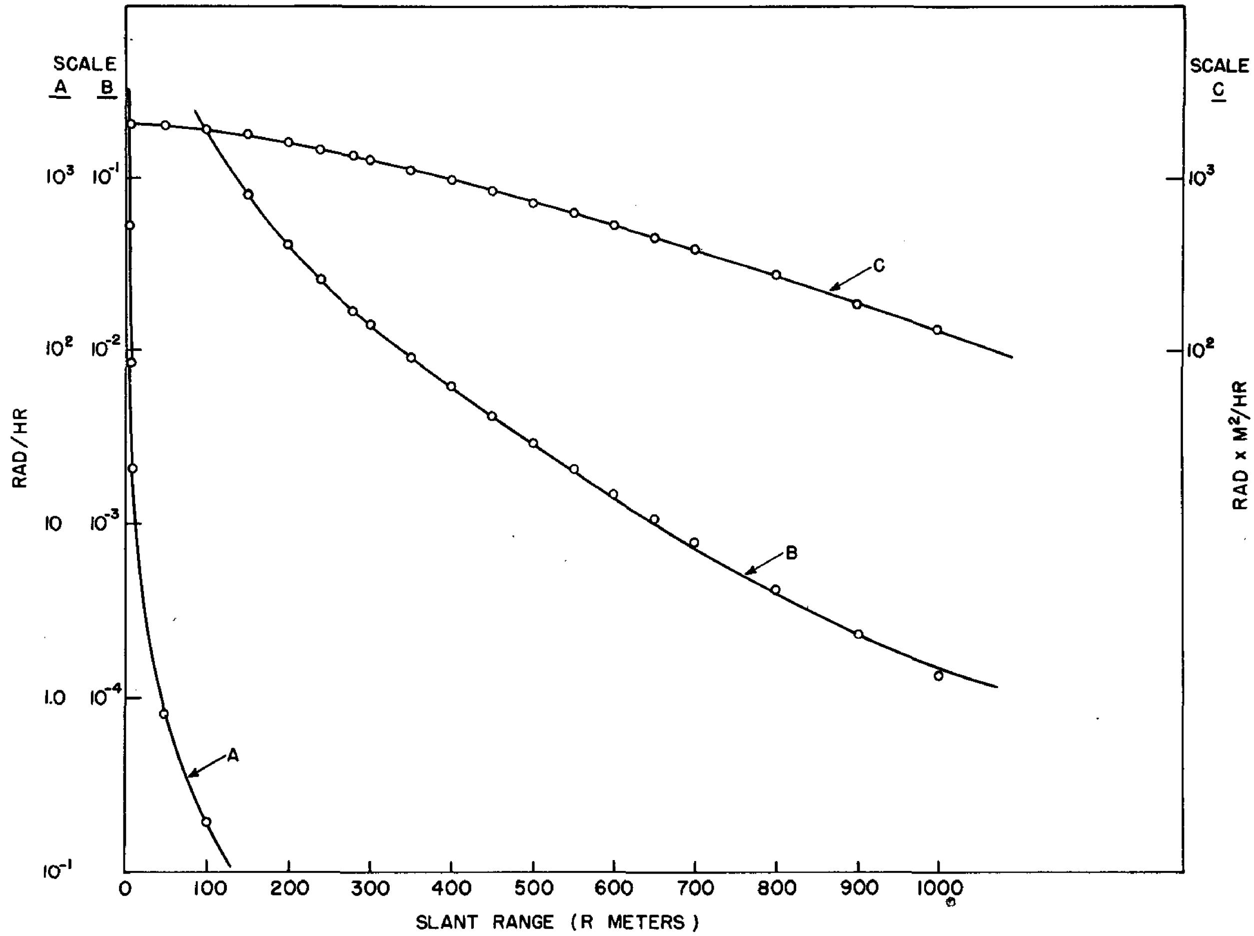

Fig. 3.11—Neutron dose rates vs. distance for HPRR operation at 1 kw.

STEADY-STATE-OPERATION PROCEDURE

Group A: Preliminary

Items in Group A are performed at the console with the key switch turned off.

1. Get the keys for the reactor, the elevator, and the hoist.
2. Review the previous HPRR Log Book entries for information regarding reactor abnormalities or special precautions.
3. Post on the reactor console a fresh reminder (if needed) of any such special conditions.
4. Inspect the annunciator panel to see if any unexpected annunciations have occurred since the previous shutdown (indicated by white annunciator lights).
5. Inspect the console for evidence of burned-out bulbs or other abnormal conditions.
6. Confirm that SAFETY BLOCK WITHDRAW LIMIT is indicated and that the reactor SCRAM light is showing.
7. On the Check-out Sheet record the reactor run number.
8. Record the date.
9. Record the type of run (if a special description is available).
10. Record the weather information.
11. Record the names of the reactor supervisor and the reactor operator.
12. Transcribe the accumulated total running time (hr) and integrated power (kw-hr) from the previous Check-out Sheet.
13. Lower the reactor from its overnight storage elevation to the 31-ft-high platform while an observer with field glasses watches from the doorway of the control bunker. The observer must have direct communication with the man at the hoist switch.

Reactor Run No.________________
Date________________
Time________________
Type of Run________________

BREN
HPRR REACTOR
CHECK-OUT SHEET

Operator________________
Assistant Operator________________

	Running Time, hr	Integrated Power, kw-hr
To date		
This run		
TOTAL		

Weather________________

Wind________mph coming from________________
Air temp. at tower________; ground________________

AT REACTOR

Burst Rod: inserted________ withdrawn________
Air Supply: Off________ On________
Reactor Visually Inspected: Yes________ No________
Loose Objects?________________

Log N calibrated________________
$\mu\mu$ ammeter zeroed________________
hoist________________
elevator________________

AT CONSOLE

Preoperation Check-out

Previous Log Book entries were read________________

Console lights look normal________________

Recorders: standardized________________
paper supply________; ink supply________________
paper rolls straight________; pens________________

Safety Channels: power on________; reset________

Count-rate Meter:
calibrated________; pulse height________; gain________

Scaler Counts:
source out________ source in________________

Safety Block Magnet Current: normal________________
after scram________; upper set-point________________

Rod Drives:	Safety B.	Reg. Rod	M.A. Rod
withdraw readings			
jog test			
full-travel test			

Start-up Permission

Start-up requested by________________
Request received (time)________ by________
Time-out started________________
Siren________________ red light________________

After Time-out

All annunciators cleared except:________________

Scram checks:
manual________; level No. 1________; level No. 2________

Source in________; fission change in________________
$\mu\mu$ ammeter range________; demand________________

Shutdown Procedures

Withdraw rod drives	Safety B.	Reg. Rod	M.A. Rod
Selsyn readings			
Seat lights			

Source inserted________________
Control room________; hoist________; elevator________
________________ has keys.

REMARKS:

TX-3195 (12-61)

Fig. 3.12—HPRR Check-out Sheet.

Reactor Run No.________________ Date________________ Type of Run________________

BREN
HPRR REACTOR
STEADY-POWER
LOG SHEET

Sheet_______of_______ Operator________________ Assistant Operator________________

		Prev.	A	B	C	D	E	F	G	H
Physical Conditions										
Reactor height										
Weather (wind velocity or general comment)										
Temperatures	Air									
	TC No. 1									
	TC No. 2									
	TC No. 3									
Operation requested (time)										
Power level (requested by:__________)										
Begin test (time)										
End test (time)										
Length of test										
Rod Readings										
Safety block										
Regulating rod										
Mass-adjustment rod										
Burst rod	in									
	out									
Source	in									
	out									
Instrument Readings										
Log N										
$\mu\mu$ ammeter	demand									
	range									
Count rate meter	cps									
	off scale									
Fission-chamber position										
Safety B. magnet current										
All annunciators cleared except:										

REMARKS:

Fig. 3.13—HPRR Steady-power Log Sheet.

Reactor Run No.__________ Operator__________

Date__________ Assist. Op.__________

Type of Run__________

____

	A	B	C					
Reactor Height								
Weather								
Air Temperature								
Core Temperature (TC No.)								
Test Began (time)								
Test Ended (time)								
Length of Test								
Power Level								

At the conclusion of this run:

Total reactor power__________ kw-hr Total reactor running time__________

Fig. 3.14—Rubber-stamp form to be used in permanent HPRR Log Book.

STEADY-STATE-OPERATION PROCEDURE
Group B: At the Reactor (25-ft Elevation)

Duties in Group B are performed at the reactor with the console key switch off.

1. Make sure the reactor, elevator, and hoist are incapable of being operated while the reactor is being inspected. As assurance, the inspector should carry these keys with him during the inspection trip.
2. Measure the radiation level at a predetermined location near the reactor (e.g., at the bottom of the reactor preamplifier cabinet) to determine how much time can safely be spent near the reactor during this inspection trip.
3. Calibrate the Log N, making sure that the selector switch is left in the OPERATE position.
4. Zero the micromicroammeter.
5. Check the response of the fission chamber and the start-up BF_3 chamber.
6. Confirm that the burst rod is latched in the inserted position.
7. Note the readings on the high-pressure gauge and the low-pressure gauge at the air-supply-tank pressure regulator. The low-pressure gauge should read zero.
8. Visually inspect the reactor core and superstructure.
9. Inspect for loose objects above or near the reactor.
10. Observe the condition and position of the reactor hoist cables and instrument cables to help guard against trouble when the reactor is hoisted to its required elevation later on.
11. When leaving the tower, see that access to the elevator and hoist and their outdoor controls are properly secured.
12. Record the names of the inspector and his assistant.

STEADY-STATE-OPERATION PROCEDURE
Group C: Preoperation Console Check-out

1. At the request of the Technical Director, turn the reactor key switch to RESET and then to REACTOR AND γ-FACILITY ON; record the time at which the time-out is started.

2. Confirm that the siren and the REACTOR ON warning light are operating properly.
3. Standardize the recorders and inspect each one for paper supply, ink supply, proper inking, proper roll take-up, etc., and put these in good working condition.
4. See that the safety channels are turned on, properly reset, and ready for operation.
5. Turn on the fission-chamber scaler, the BF_3 scaler, and the BF_3 high-voltage supply.
6. Calibrate the fission-chamber count-rate meter (C.R.M.) and see that the pulse height and gain are set at their posted values.
7. Do likewise with the BF_3 C.R.M.
8. Confirm that the fission chamber is in its fully inserted position.
9. On the Check-out Sheet record the withdrawn dial readings for the safety block, the regulating rod, and the mass-adjustment rod.
10. If the 3-min time-out has been completed, reset the scram.
11. Reset the annunciators. Record those which do not clear, determine why they do not clear, and take appropriate action.
12. Record the safety-block-magnet current; set the high-current drop-out pointer at the posted value and record it.
13. Raise the safety block until its seat light (SAFETY BLOCK WITHDRAW LIMIT) goes out, but not more than 1 in. Operate the manual scram. Repeat this operation (item 13) using level scram No. 1 or 2 (Jordan button).
14. Record the safety-block-magnet current after the scram.
15. Place the micromicroammeter selector switch in its most sensitive range of 10^{-9}.
16. Confirm that the micromicroammeter demand setting is at its posted value.
17. Confirm that the reactor-core thermocouples* read close to ambient-air temperature.
18. Run the safety block to its withdrawn limit.
19. Turn off the key switch.
20. Fill in the basic information requested at the top of the Steady-power Log Sheet.

STEADY-STATE-OPERATION PROCEDURE

Group D: Raising the Reactor

The 750-yd radius around the tower will remain cleared while the reactor is being hoisted. The REACTOR ON warning light will not be operating at this time because the hoisting is done with the reactor key switch turned off.

1. Confirm that the key switch is off, that the reactor is scrammed, and that the safety block is in its withdrawn-limit position.
2. Announce that the reactor is to be hoisted and record the reactor height requested by the Technical Director.
3. With the aid of an observer, hoist the reactor to the required elevation.
4. When the hoisting is completed, announce that the reactor is at the required elevation.

STEADY-STATE-OPERATION PROCEDURE

Group E: Going to Power

While taking the reactor to power, the operator must be especially watchful for signs of unusual reactor behavior. For example, the reactor must not go critical upon insertion of the safety block alone, nor must the reactor fail to achieve criticality when all control rods are moved to their proper places. If these or other deviations from normal behavior are observed, the operator will return the reactor to its shutdown condition, and he will proceed no further until the causes for the irregular behavior are understood and appropriate corrective measures have been taken.

1. Transcribe a set of readings from the previous Steady-power Log Sheet onto the present Steady-power Log Sheet, using the column provided for this purpose. During the present run compare the readings with the corresponding previous readings, taking special note of any sizable variations.
2. Turn the reactor key switch to RESET and then to REACTOR AND γ-FACILITY ON.
3. Confirm that the selector switch is in the DELAYED CRITICAL mode rather than in the BURST mode.
4. Reset the scram button. This turns off the scram-button light. The following lights on the control panel should now be on:
    a. REACTOR ON.
    b. OPERATION PERMIT.

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*Two active thermocouples tied to the two-pen recorder are required for delayed-critical reactor operation above 1 kw. One active thermocouple tied to one of the pens in the two-pen recorder is required for delayed-critical reactor operation at lower powers.

c. SOURCE INSERT LIMIT.
d. FISSION CHAMBER INSERT LIMIT.
e. MAGNET ENGAGED.
f. SAFETY BLOCK OUT.
g. SAFETY BLOCK WITHDRAW LIMIT.
h. SERVO OFF.
i. BURST ROD INSERT LIMIT.
j. REGULATING ROD WITHDRAW LIMIT.
k. MASS ADJUSTMENT ROD WITHDRAW LIMIT.

Interlocks prevent further action unless the following conditions exist.

a. Calibration switch (at reactor) for Log N–period amplifier is on OPERATE.
b. Period is greater than 5 sec.
c. Log count-rate-meter calibration switch is on USE.
d. Either the fission-chamber C.R.M. recorder or the start-up BF_3 C.R.M. recorder shows at least 2 counts/sec.
e. Source is in inserted limit or Log N recorder reads more than 0.0004 (10^{-5} of full power).
f. Regulating rod is in withdrawn limit.
g. Mass-adjustment rod is in withdrawn limit.
h. The 3-min time-out has been completed.

5. Reset the annunciators. Record any that do not clear, determine why they do not clear, and take appropriate action.
6. Verify what reactor power is desired,* and on the Steady-power Log Sheet record the power requested, the time, and the name of the Technical Director (or his representative) specifying the power.
7. Reset the micromicroammeter demand (if necessary) to match the power desired, according to the posted table of values.
8. Turn the safety-block position switch to INSERT. This "dead-man" switch must be held on INSERT continuously during insertion. Note that the safety-block position indicator moves down scale, the MAGNET ENGAGED light stays on, the SAFETY BLOCK OUT light goes out, and the SAFETY BLOCK WITHDRAW LIMIT light goes out. As the safety block moves in, observe the increasing counting rates exhibited by the fission-chamber channel and by the BF_3 start-up channel. Pause occasionally during the insertion process and note that the count rate levels off each time.
9. Continue to insert the safety block until the SAFETY BLOCK INSERT LIMIT light comes on. Continue inserting a small fixed distance beyond this reading† to ensure the proper seating of the safety block against its stop. Interlocks prevent future steps involving the regulating rod and the mass-adjustment rod unless the safety block remains firmly in the inserted position.
10. Turn the regulating-rod position switch to INSERT. This is the same type of dead-man switch as that used for the safety block. Observe that the REGULATING ROD WITHDRAW LIMIT light goes out and that the indicator hand moves down scale. Continue to insert while carefully observing the increase in counts on the count-rate-meter channels; stop the insertion when a dial reading of 4 in. is reached. This places the regulating rod near the mid-point of its travel, from which adjustments later can be made conveniently in either direction. (The mass-adjustment rod will not be inserted at the same time. This rod is not interlocked but is administratively controlled.)
11. Before starting to insert the mass-adjustment rod, review the Steady-power Log Sheet and note the position of the rods during the previous level-power operation. The operator must not take the reactor to critical by relying upon dial readings. He should be alert, however, to note any inconsistencies in the dial readings. If such should be observed, he will return the reactor to a safely subcritical condition until such deviations are thoroughly understood.
12. Keeping these thoughts in mind, turn the mass-adjustment-rod position switch to INSERT. This is the same type of switch as that used for moving the regulating rod and the safety block. Observe that the MASS ADJUSTMENT ROD WITHDRAW LIMIT light goes out and the indicator hand moves down scale. (The regulating rod will not be inserted at the same time. This rod is not interlocked but is administratively controlled.) Pause occasionally during the rod-insertion process and note

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*The reactor power is limited to 1 kw for ordinary operation. Steady-state powers from 1 to 10 kw may be achieved only after the resistor size in the safety amplifier has been converted so that the 50-μa output signal corresponds to a 15-kw power level. The details of this conversion are described in Sec. 3.2.4(d).

†This is in the order of 0.1 in. and will be set in the field. It represents overtravel of the drive mechanism only and not of the safety block itself.

that the count rate levels off each time the rod motion is stopped. When the point is reached at which the count rate continues to rise steadily while the rod motion is stopped, criticality has been achieved.

13. Switch off the high-voltage supply to the BF_3 channel when its recorder goes off scale.
14. Withdraw the fission chamber to keep it on scale as the power rises.
15. Advance the micromicroammeter selector switch to higher ranges as the power rises so that the micromicroammeter recorder will be kept on scale.
16. Level off at the desired power by matching the micromicroammeter-recorder pointer with that on the DEMAND meter.
17. The reactor may be put on servo (automatic operation) if Sequence "A" has been satisfied, as indicated when the SEQUENCE "A" light shows on the control panel. Sequence "A" requires that the safety block be in its inserted limit, that the Log N recorder reads at least 0.0004, and that the END OF RUN light is off. The reactor is placed in servo operation by turning the servo switch to the ON position. This turns off the SERVO OFF light and turns on the SERVO ON light. Switching into servo operation can be achieved with a minimum of fluctuations in power and regulating-rod position if it is done at a time when the micromicroammeter recorder agrees closely with the DEMAND meter and when the reactor period is reasonably long (greater than ±30 sec).
18. When the reactor begins operating at the desired power, announce that the reactor is at power and record the "Begin test" time on the Steady-power Log Sheet.
19. Mark the recorder charts as follows:
    a. On all charts write the date, the run number, the time the reactor arrived at the desired power, and the power level.
    b. On the Log N—recorder and the micromicroammeter charts, add the micromicroammeter selector-switch setting and the demand setting.
20. On the Steady-power Log Sheet fill in the following information while comparing it with the similar entries from the previous run.
    a. Safety-block dial reading.
    b. Regulating-rod dial reading.
    c. Mass-adjustment-rod dial reading.
    d. Burst-rod position (in or out).
    e. Source position (in or out).
    f. Log N.
    g. Micromicroammeter demand.
    h. Micromicroammeter range.
    i. Fission-chamber count-rate-meter readings.
    j. Fission-chamber position.
    k. Safety-block-magnet current.
    l. Uncleared annunciators (if any).
21. Once the reactor is operating smoothly, the major task of the person operating the reactor is to watch for changes and to take the proper corrective action. These changes may consist in variations in the weather, wind, air temperature, core temperature, rod positions, or other indicators at the console. Gradual drift of certain readings with time is to be expected. These readings should be noted and their significance understood.
22. The lettered columns on the Steady-power Log Sheet (Fig. 3.13) are to be filled in during long runs as a record of such changing conditions (item 21). The frequency of recording will vary, 1 hr being defined as the maximum interval between the recording of such entries. Much shorter intervals should be used when appreciable shift is observed. A long continuous run bears the same letter designation for the duration of the run, and the letters at the top of the columns on the log sheet should therefore be altered accordingly. A new letter suffix is introduced whenever the reactor power level is changed, but trial powers achieved while seeking a desirable power level will not be recorded and will not be assigned separate letter suffixes.

STEADY-STATE-OPERATION PROCEDURE

Group F: Shutting Down

I. Emergency Manual Scram. To achieve emergency shutdown, perform the following operations in quick order.

1. Turn red scram handle either direction.
2. Turn servo handle to SERVO OFF position.
3. Turn the GROUP WITHDRAW handle. This simultaneously withdraws the regulating rod and the mass-adjustment rod. If the handle is left in this position, the withdrawing motion will continue after the handle is released.

4. Announce that the reactor has been scrammed.
5. Record the time.
6. Record the reason the scram was required.

II. Unexpected Automatic Scram. If an unexpected scram occurs, perform the following operations in order.

1. Turn servo handle to SERVO OFF position.
2. Record the annunciators that came on at the time of the scram.
3. If the reason for the scram is not immediately apparent, use the Steady-power Log Sheet as a check list, take new instrument and rod-position readings, and compare them with the previous listed readings. Group withdrawal may be performed after the regulating-rod and mass-adjustment-rod positions have been recorded by turning the GROUP WITHDRAW handle.
4. Do not attempt to go back to power until an acceptable reason for the scram can be postulated and the proper corrective measures have been taken.

III. Normal Shutdown*

1. Announce that the reactor is to be shut down.
2. Turn the servo handle to SERVO OFF position.
3. Turn the GROUP WITHDRAW handle. This withdraws the safety block, regulating rod, and mass-adjustment rod simultaneously.
4. Record the time of shutdown on the Steady-power Log Sheet and on all recorder charts.
5. Observe the power-level drop on the Log N recorder.
6. When rod travel is completed, confirm that the safety block, regulating rod, and mass-adjustment rod are in their fully withdrawn positions as indicated by their limit lights.
7. Turn the scram handle to SCRAM and leave it there.
8. Turn the fission-chamber position handle to INSERT.
9. Turn the reactor key switch to the REACTOR OFF position.
10. Announce that the reactor is shut down.

STEADY-STATE-OPERATION PROCEDURE
Group G: Postoperation Duties

The following duties are normally performed at the end of each shift.

1. Remove the reactor key from the key switch.
2. Record the three dial readings when the safety block and the rods are in their withdrawn limits.
3. Compute the total running time (hr) and the total integrated power (kw-hr) for the complete run, enter the numbers on the Check-out Sheet, and add them to the previously accumulated totals.
4. Transcribe into the permanent, bound HPRR Log Book the basic information concerning the run as suggested by the rubber-stamp blanks (Fig. 3.14) to be filled out. Add other data as requested by the Technical Director.
5. Write in the HPRR Log Book a careful description of the run and record any remarks which may be helpful during future runs or which properly belong in this permanent record book.
6. Install the Steady-power Log Sheet in its loose-leaf binder and put the binder and the permanent HPRR Log Book in their proper places.
7. Make a Rad-Safe survey measurement at the bunker door before venturing outside.
8. Lower the reactor to its overnight storage position with the aid of an observer.
9. Check the control-room lights, heaters, air conditioning, etc., to make sure they are in proper stand-by condition.
10. Take care of the reactor, hoist, and elevator keys in the approved manner.

(c) *Burst Operation*

BURST-OPERATION PROCEDURE
Group A: Review of Burst Theory

Basically, firing a burst consists in introducing excess reactivity of the order of 1 dollar or slightly greater into an otherwise delayed-critical reactor, thereby making the system

*The reactor should be shut down by scramming once each operating day to demonstrate that reliable scrams are routinely achieved under actual operating conditions.

critical on prompt neutrons alone. The power level rises rapidly until the negative temperature coefficient of reactivity makes the system subprompt critical, at which time the power level decreases because the delayed neutrons do not contribute appreciably to the reactivity during the time interval of the burst. The delayed neutrons do contribute to reactivity immediately following the burst peak, and, if there were no scram devices, the system would rise in temperature until the negative temperature coefficient reduced the reactivity to delayed critical. The reactor fast-scram system is designed to prohibit any appreciable contribution to reactivity of the delayed-neutron groups for bursts having yields of the order of 1×10^{17} fissions.

The stroke of the burst rod is fixed at 7.476 in., which has been shown to be worth approximately 1 dollar* in reactivity. It should be emphasized that the value of the burst rod may vary as much as a few cents with environment. The procedure for obtaining a burst greater than that obtainable with the burst rod alone must therefore consist in adding to the core the amount of excess reactivity which, when added to the reactivity of the burst rod, will give the desired yield. Figure 3.15 shows the variation of temperature change, and therefore yield, with reactivity added to that obtained with the burst rod; a conservative interpretation of the data indicates that a change ranging from 11.0 to 11.5 cents increases the yield by about 50%. It is therefore clear that, if an upper limit of 1.5×10^{17} fissions is to be maintained, not only will precise additions of reactivity be necessary but also a new curve such as that in Fig. 3.15 must be generated with each change of environment. The first two points should be obtained by subtracting reactivity from the core (e.g., 20 cents and 10 cents), and the third point should be obtained by adding the reactivity of the burst rod alone to that of the delayed-critical core. The data should then be compared to those in Fig. 3.15 by drawing a parallel curve. This new curve should be used to select the next increment of reactivity to be added, and such a procedure should be continued until a yield of 1×10^{17} fissions is reached.

The measurement of negative reactivities that are 10 cents or greater can be made by using the reactivity-vs.-position curve of the mass-adjustment rod shown in Fig. 3.16. Positive reactivity, ρ, will be determined by measuring stable positive period, T, by using two BF_3 chambers and correlating T with ρ with the well-known inhour relation[2]

$$\rho \text{ (cents)} = \frac{1}{0.64}\left[\frac{168.84}{T + 80.4} + \frac{459.2}{T + 32.8} + \frac{112.25}{T + 8.98} + \frac{84}{T + 3.52} + \frac{6.51}{T + 0.88} + \frac{0.89}{T + 0.33}\right]$$

The measured reactivities must agree within 0.3 cent for a burst of 1×10^{17} fissions to ensure that too much reactivity is not introduced.

BURST-OPERATION PROCEDURE

Group B: General Procedures

The Operation BREN schedule calls for all tests requiring steady-state reactor operation to be completed before the burst tests are started.

The reactor is prepared for bursting by unlatching the burst rod, turning on the reactor air supply, and demonstrating that the burst rod will perform as required. Step by step instructions for this work are given in Sec. 3.2.4(d).

Once the reactor has been prepared for routine burst operation, the duties listed in Groups C, D, E, F, and I will normally be performed once per shift, and those in Groups G and H will normally be required once per burst attempt.†

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*These and other experimental data pertaining to the HPRR referred to in this report were obtained by J. T. Mihalczo at the ORNL Critical Experiments Facility. The data are preliminary and are subject to small revisions.

†Early experimental bursts will intentionally have such low yields that they can hardly be described as bursts. Yet these and the occasional later bursts which turn out to be "fizzles" will each be assigned a separate burst number for the records.

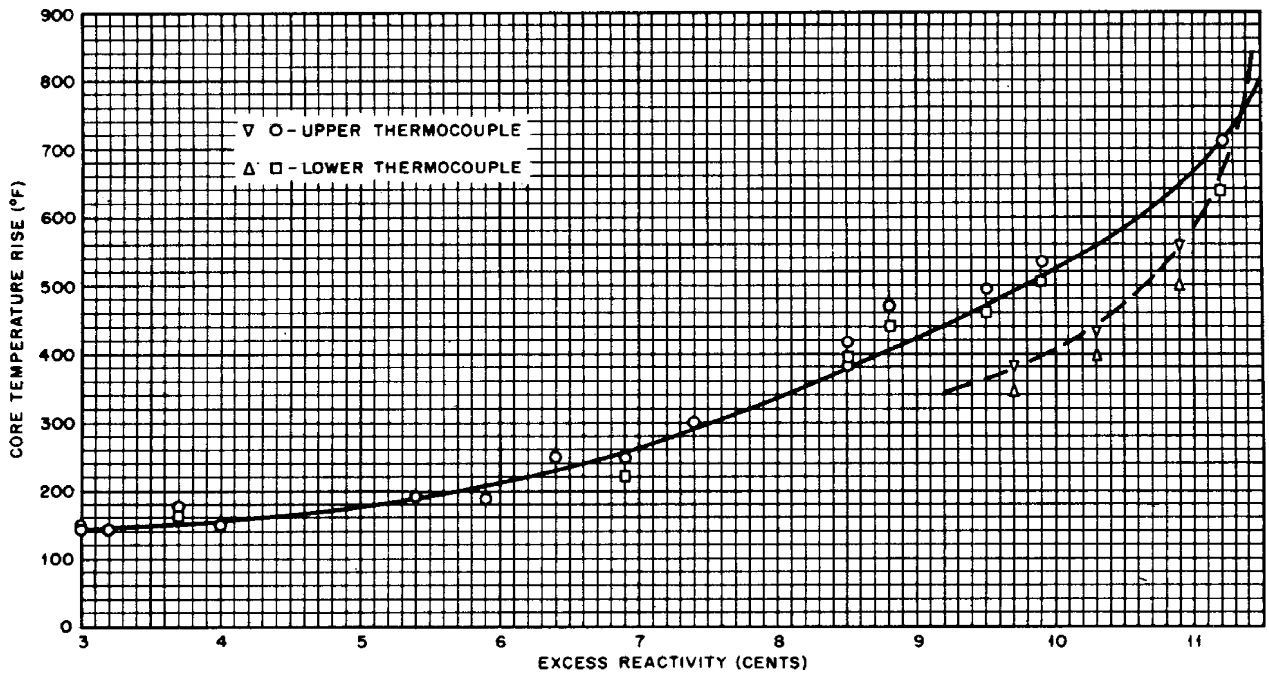

Fig. 3.15—Core-temperature rise vs. excess reactivity introduced by the mass-adjustment rod before the burst rod is fired.

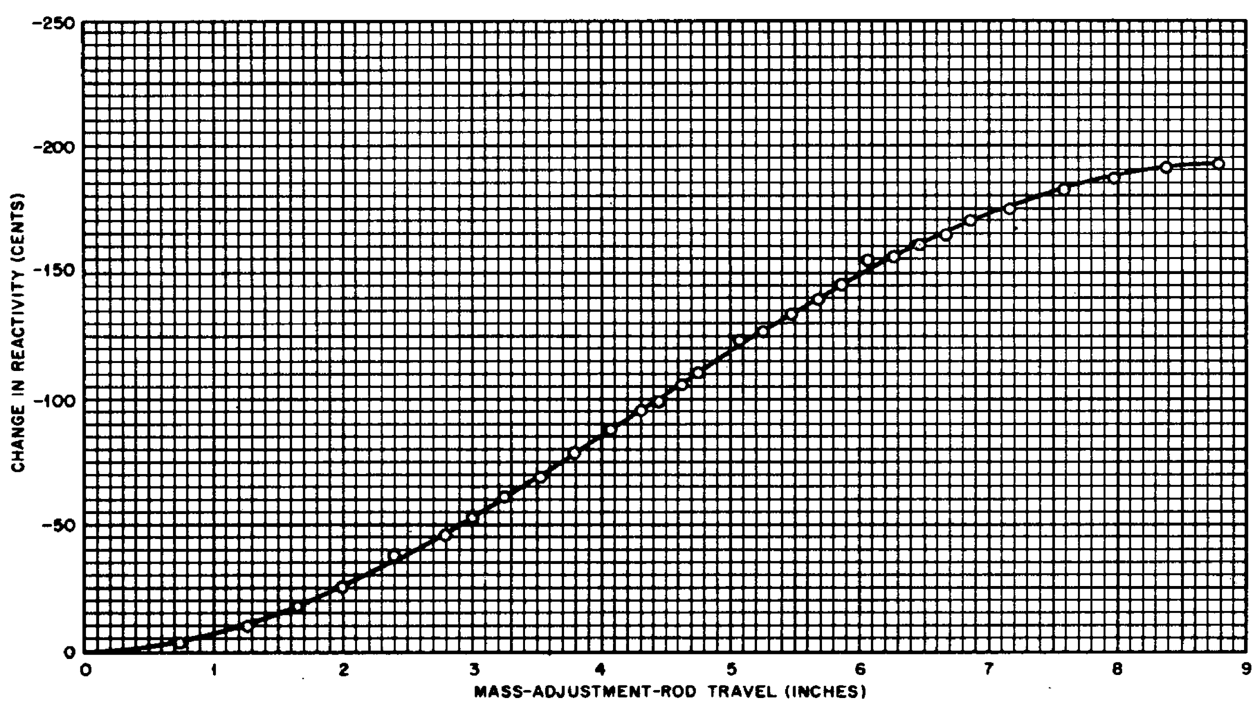

Fig. 3.16—Reactivity change vs. mass-adjustment-rod travel (measured as the distance of the bottom of the mass-adjustment rod from the bottom of the core).

The reactor operator* will proceed according to the check lists that follow; he will annotate each step, where applicable, in the spaces provided on the BREN HPRR Reactor Check-out Sheet (Fig. 3.12), Steady-power Log Sheet (Fig. 3.14), and Burst Log Sheet (Fig. 3.17). The reactor supervisor, with the concurrence of the reactor operator and the instrument engineer, may make minor amendments to this procedure by describing the changes in the HPRR Log Book. Significant deviations, however, will require prior approval of the appropriate ORNL committee.

A single run number will normally be used per shift, and a new burst number will be used for each burst attempt.

The operator will carefully describe in the HPRR Log Book his observations during reactor check-out and operation. He will also keep careful notes in the HPRR Log Book regarding repairs, maintenance, or alterations made to the reactor.

BURST-OPERATION PROCEDURE
Group C: Preliminary

Duties in Group C are performed at the console with the key switch turned off.

1. Get the keys for the reactor, the elevator, and the hoist.
2. Review the previous HPRR Log Book entries for information regarding reactor abnormalities or special precautions.
3. Post on the reactor console a fresh reminder (if needed) of any such special conditions.
4. Inspect the annunciator panel to see if any unexpected annunciations have occurred since the previous shutdown (indicated by white annunciator lights).
5. Inspect the console for evidence of burned-out bulbs or other abnormal conditions.
6. Confirm that SAFETY BLOCK WITHDRAW LIMIT is indicated and that the reactor SCRAM light is showing.
7. Open the solenoid-operated air-block valve.
8. Record the reactor run number.
9. Record the date.
10. Record the type of run (if a special description is available).
11. Record the weather information.
12. Record the names of the reactor supervisor and the reactor operator.
13. Transcribe the accumulated total running time (hr) and integrated power (kw-hr) from the previous Steady-power Log Sheet.
14. Lower the reactor from its overnight storage elevation to the 31-ft-high platform with the aid of an observer.

BURST-OPERATION PROCEDURE
Group D: At the Reactor (25-ft Elevation)

Duties in Group D are performed at the reactor with the console key switch off.

1. Make sure the reactor, elevator, and hoist are incapable of being operated while the reactor is being inspected. As assurance, the inspector should carry these keys with him during the inspection trip.
2. Measure the radiation level at a predetermined location near the reactor to determine how much time can safely be spent near the reactor during this inspection trip.
3. Calibrate the Log N, making sure that the selector switch is left in the OPERATE position.
4. Zero the micromicroammeter.
5. Check the response of the fission chamber and the BF_3 chambers.
6. Note the readings on the high-pressure gauge and the low-pressure gauge at the air-supply-tank pressure regulator. If the high-pressure gauge reads less than 500 psi, replace the tank with a full one. Adjust the pressure regulator (if necessary) to set the low-pressure gauge at 70 psi.
7. Confirm that the burst rod is in its withdrawn position.
8. Visually inspect the reactor core and superstructure.
9. Inspect for loose objects above or near the reactor.
10. Observe the condition and position of the reactor hoist cables and instrument cables to help guard against trouble when the reactor is hoisted to its required elevation later on.

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*The term "reactor operator" refers to the person performing the work. He may have the title "reactor supervisor" or "reactor operator."

BREN HPRR REACTOR
BURST LOG SHEET

Run No.________

Burst No.________

Date________

Time________

React. Sup.________

React. Op.________

Tech. Dir.________

React. Ht.________

Air Temp.________

Wind________

Weather:

Comments:

	Trial Periods 1	2	3	4	Target Burst	Related Burst
PREPARATION FOR BURST						
Source Out (confirm)						
Safety Block (dial)						
Regulating Rod						
Mass Adjust. Rod						
Core Temp. TC 1						
TC 2						
TC 3						
No. 1 BF_3, Start						
Final cps						
No. 2 BF_3, Start						
Final cps						
Fiss. ch., Start						
Final cps						
Period, sec						
Corres. ρ, cents						
NEUTRON DECAY						
Safety B. out, time						
Core Temp. TC 3						
time						
TC 3						
time						
TC 3						
time						
BURST						
Time of burst						
Core temperatures:						
TC 1 (max.)						
before burst						
TC 1 rise						
TC 2 (max.)						
before burst						
TC 2 rise						
Approx. fiss. yield						

Fig. 3.17 — HPRR Burst Log Sheet.

11. When leaving the tower, see that access to the elevator and hoist and their outdoor controls are properly secured.
12. Record the names of the inspector and his assistant.

BURST-OPERATION PROCEDURE

Group E: Preoperation Console Check-out

1. At the request of the Technical Director, turn the reactor key switch to RESET and then to REACTOR AND γ-FACILITY ON; record the time at which the time-out is started.
2. Confirm that the siren and the REACTOR ON warning light are operating properly.
3. Standardize the recorders and inspect each one for paper supply, ink supply, proper inking, proper roll take-up, etc., and put these in good working condition.
4. See that the safety channels are turned on, properly reset, and ready for operation.
5. Turn on the fission-chamber scaler, the BF_3 scalers, and the BF_3 high-voltage supplies.
6. Calibrate the fission-chamber count-rate meter (C.R.M.) and see that the pulse height and gain are set at their posted values.
7. Do likewise with the BF_3 C.R.M.'s.
8. Confirm that the fission chamber is in its fully inserted position.
9. On the Check-out Sheet record the withdrawn dial readings for the safety block, the regulating rod, and the mass-adjustment rod.
10. If the 3-min time-out has been completed, reset the scram.
11. Reset the annunciators. Record those which do not clear, determine why they do not clear, and take appropriate action.
12. Record the safety-block-magnet current; set the high-current drop-out pointer at the posted value and record it.
13. Raise the safety block until its seat light (SAFETY BLOCK WITHDRAW LIMIT) goes out, but not more than 1 in. Operate the manual scram. Repeat the operation (item 13) using level scram No. 1 or 2 (Jordan button).
14. Record the safety-block-magnet current after the scram.
15. Run the safety block to its withdrawn limit.
16. Confirm that the reactor-core thermocouples read close to ambient-air temperature. Two active thermocouples tied to the two-pen (0 to 1000°F) recorder and one active thermocouple tied to the 0 to 100°C recorder are required for burst operation.
17. Turn off the key switch.
18. Fill in the basic information requested at the top of the Steady-power Log Sheet.

BURST-OPERATION PROCEDURE

Group F: Raising the Reactor

The 750-yd radius around the tower will remain cleared while the reactor is being hoisted. The REACTOR ON warning light will not be operating at this time because the hoisting is done with the reactor key switch turned off.

1. Confirm that the key switch is off, that the reactor is scrammed, and that the safety block is in its withdrawn-limit position.
2. Announce that the reactor is to be hoisted and record the reactor height requested by the Technical Director.
3. With the aid of an observer, hoist the reactor to the required height.
4. When the hoisting is completed, announce that the reactor is at the required elevation.

BURST-OPERATION PROCEDURE

Group G: Going to Power

While taking the reactor to power, the operator must be especially watchful for signs of unusual reactor behavior. For example, the reactor must not go critical upon insertion of the safety block alone, nor must the reactor fail to achieve criticality when all control rods are moved to their proper places. If these or other deviations from normal behavior are observed, the operator will return the reactor to its shutdown condition, and he will proceed no further until the causes for the irregular behavior are understood and appropriate corrective measures have been taken.

1. Transcribe a set of readings from the previous Steady-power Log Sheet onto the present Steady-power Log Sheet, using the column provided for this purpose. During the present run compare the readings with the corresponding previous readings, taking special note of any sizable variations.
2. Turn the reactor key switch to RESET and then to REACTOR AND γ-FACILITY ON.
3. Confirm that the selector switch is in the DELAYED CRITICAL mode rather than in the BURST mode.

4. Reset the scram button. This turns off the scram-button light. The following lights on the control panel should now be on:
   a. REACTOR ON.
   b. OPERATION PERMIT.
   c. SOURCE INSERT LIMIT.
   d. FISSION CHAMBER INSERT LIMIT.
   e. MAGNET ENGAGED.
   f. SAFETY BLOCK OUT.
   g. SAFETY BLOCK WITHDRAW LIMIT.
   h. SERVO OFF.
   i. BURST ROD WITHDRAW LIMIT.
   j. REGULATING ROD WITHDRAW LIMIT.
   k. MASS ADJUSTMENT ROD WITHDRAW LIMIT.

   Interlocks prevent further action unless the following conditions exist.
   a. Calibration switch (at reactor) for Log N–period amplifier is on OPERATE.
   b. Period is greater than 5 sec.
   c. Log count-rate-meter calibration switch is on USE.
   d. Either the fission-chamber C.R.M. recorder or the start-up BF_3 C.R.M. recorder shows at least 2 counts/sec.
   e. Source is in inserted limit or Log N recorder reads more than 0.0004 (10^{-5} of full power).
   f. Regulating rod is in withdrawn limit.
   g. Mass-adjustment rod is in withdrawn limit.
   h. The 3-min time-out has been completed.
5. Reset the annunciators. Record any that do not clear, determine why they do not clear, and take appropriate action.
6. Turn the safety-block position switch to INSERT. This dead-man switch must be held on INSERT continuously during insertion. Note that the safety-block position indicator moves down scale, the MAGNET ENGAGED light stays on, the SAFETY BLOCK OUT light goes out, and the SAFETY BLOCK WITHDRAW LIMIT light goes out. As the safety block moves in, observe the increasing counting rates exhibited by the fission-chamber channel and by the BF_3 start-up channel. Pause occasionally during the insertion process and note that the count rate levels off each time.
7. Continue to insert the safety block until the SAFETY BLOCK INSERT LIMIT light comes on. Continue inserting a small fixed distance beyond this reading* to ensure the proper seating of the safety block against its stop. Interlocks prevent future steps involving the regulating rod and the mass-adjustment rod unless the safety block remains firmly in the inserted position.
8. Turn the regulating-rod position switch to INSERT. This is the same type of dead-man switch as that used for the safety block. Observe that the REGULATING ROD WITHDRAW LIMIT light goes out and that the indicator hand moves down scale. Continue to insert until the rod has traveled its full distance and the INSERT LIMIT light comes on, all the while carefully observing the increase in counts on the count-rate-meter channels. (The mass-adjustment rod will not be inserted at the same time. This rod is not interlocked but is administratively controlled.)
9. Before starting to insert the mass-adjustment rod, review the Steady-power Log Sheet and note the position of the rods during the previous level-power operation. The operator must not take the reactor to critical by relying upon dial readings. He should be alert, however, to note any inconsistencies in dial readings. If such should be observed, he will return the reactor to a safely subcritical condition until such deviations are thoroughly understood.
10. Keeping these thoughts in mind, turn the mass-adjustment-rod position switch to INSERT. This is the same type of switch as that used for moving the regulating rod and the safety block. Observe that the MASS ADJUSTMENT ROD WITHDRAW LIMIT light goes out and that the indicator hand moves down scale.
11. Pause occasionally during the rod-insertion process and note that the count rate levels off each time the rod motion is stopped. When the point is reached at which the count rate continues to rise steadily while the rod motion is stopped, criticality has been achieved.
12. Switch off the high-voltage supply to the start-up BF_3 channel when its recorder goes off scale.
13. Level off the reactor at a Log N reading of about 0.001.
14. On all charts write the date, run number, burst number, and time.

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*This is in the order of 0.1 in. and will be set in the field. It represents overtravel of the drive mechanism only and not of the safety block itself.

BURST-OPERATION PROCEDURE

Group H: Preparing for and Firing the Burst

1. Fill in the basic information requested at the top of the Burst Log Sheet.
2. Record the following information describing the target burst.
   a. Name of the Technical Director (or his representative).
   b. Size of the burst desired, relative to an earlier burst (give burst number of earlier related burst).
   c. Reactor period required.
   d. Fission yield desired.
   e. Maximum core temperature expected.
3. Fill in the proper column with detailed information about the earlier related burst.
4. Withdraw the source. Observe that the SOURCE INSERT LIMIT light goes out and the SOURCE WITHDRAW LIMIT light comes on.
5. Leave the regulating rod fully inserted for the remainder of the burst test.
6. Partially withdraw the mass-adjustment rod to lower the reactor power until the two "period" BF_3 channels are counting at rates* suitable for starting careful measurements of reactor period.
7. Move the mass-adjustment rod† to establish a trial reactor period approximating the desired period. (It is assumed here that a positive period is needed.)
8. Record the safety-block and rod positions.
9. Record the reactor-core temperature as indicated on TC 3 (connected to the 0 to 100°C recorder).
10. Record the scaler counts as the reactor rises steadily on its period. Uniform intervals should be used for these counts. After the power has risen a decade or more, plot the counts vs. time on semilog paper, measure the slope of the curve, and compute the reactor period.‡ The alternate method of transcribing the counts directly from the scaler onto semilog graph paper is permissible.
11. Compute the reactor period, determine the corresponding reactivity in cents, and compare with the target values.
12. Record the appropriate "Trial 1" values on the Burst Log Sheet.
13. If the reactor period is not satisfactory or if the power level rises too high (above 0.004 on the Log N, which is approximately 1 watt) before the period computations are completed, withdraw the safety block to let the power drop back to a level suitable for again starting period measurements, re-establish the reactor period (trial 2), and repeat steps 7 through 12. The period is re-established by moving the safety block back to its inserted limit. This may be done by turning the safety-block control handle to the INSERT position while simultaneously operating the SAFETY BLOCK INHIBIT BYPASS switch.
14. When a satisfactory period has been achieved, turn the safety-block switch to WITHDRAW and record the time. The switch does not have to be held. Note that the INSERT LIMIT light goes out and that the safety-block position indicator moves up scale. When the safety block has completed its travel, note that the SAFETY BLOCK OUT and the WITHDRAW LIMIT lights come on and that the MAGNET ENGAGED light remains on.
15. Turn the mode switch to BURST. This prevents further addition of reactivity by means of the mass-adjustment rod (or regulating rod).
16. Observe the decay of the neutron population, waiting at least 20 min to ensure that it has decayed to background level.
17. Accurately read and record TC 3 (0 to 100°C recorder) near the beginning, middle, and end of this waiting period and compare with the temperature recorded during the latest trial period.
18. If the core temperature drifts as much as 2°C in either direction, a new trial reactor period must be established. If a new trial period is needed, take the following steps.
   a. Scram the reactor by turning the scram handle.
   b. Withdraw all the rods.
   c. Insert the source.
   d. Insert the fission chamber.

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*These values, to be established in the field, will be selected to provide adequate range for measuring the period in the linear-response portion of the instrument.

†The mass-adjustment rod is used here because its position can be determined accurately by virtue of its two-handed (coarse and fine) selsyn dial, whereas the regulating rod has only a single-handed (coarse) position indicator.

‡Note that the reactor power continues to rise while the period is being computed. Effort should therefore be made to speed this computation. Reference curves should be plotted in their most useful form, formulas should be precalculated as much as possible, and pencils, straightedges, graph paper, table space, etc., should be prepared for use before they are needed.

e. Turn on the high voltage supplying the start-up BF_3.
f. Change the mode selector switch from BURST to DELAYED CRITICAL.
g. Return to step 4 under Burst-operation Procedure, Group G: Going to Power, and proceed from there through step 18 again and on through the remainder of the Burst-operation Procedure.

19. If the core temperature remains steady and 20 min have passed, turn the safety-block position switch to INSERT and hold it there. Interlocks prevent safety-block insertion unless Sequences "A" and "B" have been completed. Lights on the control panel indicate when Sequences "A" and "B" have been completed. The requirements for completing Sequences "A" and "B" are as follows:
    a. Safety block must have been in its inserted limit.
    b. Power level must have been at least up to 0.0004 on the Log N (10^{-5} of full power).
    c. Safety block must subsequently have been in its withdrawn limit.
    d. Source must be in its withdrawn limit.
    e. NOT AT END OF RUN interlock must be satisfied. This requires that the burst rod be withdrawn and that there be no scram signals.
20. Observe the SAFETY BLOCK INSERT LIMIT light come on, and continue to insert 0.1 in. further to duplicate the reading obtained during the successful trial reactor period.
21. Immediately turn the burst-rod position switch to INSERT. This fires the burst rod in. The switch handle does not have to be held.
22. During the burst, scram circuits are tripped and annunciators are actuated. Push the SCRAM ACKNOWLEDGE button to turn off the audible alarm.
23. Observe that the following indicator lights are now showing at the console.
    a. SCRAM
    b. BURST ROD WITHDRAW LIMIT
    c. SAFETY BLOCK OUT
    d. END OF RUN

    The safety-block drive should be automatically moving toward its withdrawn limit as indicated when its position indicator hand is moving up scale.
24. Initiate group withdrawal by turning the GROUP WITHDRAW switch handle. The handle will remain in the WITHDRAW position. This automatically withdraws the regulating rod and the mass-adjustment rod as indicated when their position-indicator hands are moving up scale.
25. Announce the time of the burst and record the time on the Burst Log Sheet and also on the recorder charts.
26. In the "Burst" section of the Burst Log Sheet, record the maximum core temperatures indicated by "TC 1" and "TC 2." Subtract the initial (before burst) temperature from these values to get the temperature rise.
27. Record the annunciators that were tripped at the time of the burst.
28. Insert the source.
29. Close the solenoid-operated air-block valve.
30. Turn off the reactor key switch and observe the REACTOR OFF light come on.

BURST-OPERATION PROCEDURE

Group I: Postoperation Duties

The following duties are normally performed at the end of each shift.

1. Remove the reactor key from the key switch.
2. Record the three dial readings when the safety block and the rods are in their withdrawn limits.
3. Compute the total running time (hr) and total integrated power (kw-hr) for the complete run, enter the numbers on the Check-out Sheet, and add them to the previously accumulated totals.
4. Transcribe into the permanent, bound HPRR Log Book the basic information concerning the run as suggested by the rubber-stamp blanks to be filled out. Add other data as requested by the Technical Director.
5. Write in the HPRR Log Book a careful description of the run and record any remarks which may be helpful during future runs or which properly belong in this permanent record book.
6. Install the Steady-power Log Sheet in its loose-leaf binder and put the binder and the permanent HPRR Log Book in their proper places.
7. Make a Rad-Safe survey measurement at the bunker door before venturing outside.
8. Lower the reactor to its overnight storage station while an observer with field glasses is stationed at the doorway of the bunker.
9. Check the control-room lights, heaters, air conditioning, etc., to make sure they are in proper stand-by condition.
10. Take care of the reactor, hoist, and elevator keys in the approved manner.

(d) *Maintenance and Alterations*

(i) Definition. Any repair work or one-for-one replacement of parts which is necessary to keep the reactor and its control and safety system operating in the approved manner is defined as maintenance. Other repair work and/or system changes which are designed to change the mode of operation or to improve the operation are defined as alterations.

(ii) Responsibility. Maintenance, both electrical and mechanical, may be initiated by the reactor supervisor, the instrument engineer, or the reactor operator; but normally it will be initiated by the reactor supervisor. All maintenance in the control and safety system will be done or supervised by the instrument engineer. In any case, the work will be described in the permanent HPRR Log Book. No one will work alone around the reactor or its controls, on the tower, or in the bunker. It is permissible, however, for one man to work at the reactor and one in the bunker if they maintain communications.

(iii) Procedures. There are two types of alterations, those for which the changes are already outlined and those which may be found necessary after operations are under way and for which change notices must be written.

The planned alterations will be initiated by the reactor supervisor when directed to do so by the Technical Director. For other alterations the following procedure will be followed by the operating group. The reactor supervisor will request that the ORNL reactor-controls group make a change in the system. This request may be made by phone. The reactor-controls group will initiate a change notice, which must be approved by (1) the head of the reactor-controls group, (2) the individual in the Neutron Physics Division who has the over-all responsibility for operation of the HPRR, and (3) the Technical Director of Operation BREN. If a significant change in operating procedures results because of the revision, the Neutron Physics Division will notify the head of the ORNL Reactor Operations Review Committee. When all necessary approvals have been obtained, the reactor supervisor will be instructed to proceed with the change. He may be notified by phone with a follow-up letter including the changed drawings, if the revision is not overly involved. The reactor supervisor will keep the Technical Director informed concerning the status of the requests and subsequent changes. If operating changes will result, the Technical Director will inform the AEC representative from the Albuquerque Operations Office of the change. This procedure will be followed for all changes at NTS.

(iv) Planned Alterations. Two alterations are planned, namely, converting from 1-kw operation to 10-kw operation and converting from steady-state operation to burst operation.

PLANNED-ALTERATIONS PROCEDURE

1. To convert from 1 kw to 10 kw operation, the following procedure is necessary.
   a. Change the 3.3-megohm resistors in each level-safety preamplifier to 330 kohm.
   b. Check performance with a current generator.
   c. Check performance with neutron level from reactor.
2. To convert from steady-state operation to burst operation, the following procedure is necessary.
   a. Remove the burst-rod hold-down clamp.
   b. Open the high-pressure-tank valve.
   c. Remove the wire connecting terminal 9G to 9H located on the relay-panel terminal block RST9. This change makes the low-air-pressure interlock on the low-pressure tank scram the reactor when the pressure falls to 60 to 65 lb.
   d. Remove the wire connecting terminal 16H to 16N on relay contact K45A of relay K45 located on relay panel. This change makes the burst-rod withdraw-limit interlock operative in the safety-block-insert control circuit. These electrical changes are shown on the Elementary Diagram (NDA Dwg. 12049).
   e. Check the system for proper performance by going through burst operations with the core made several dollars subcritical when the burst rod is inserted.

(v) Minor Maintenance. The reactor core presents a considerable gamma-ray hazard that must be considered each time the reactor is approached closely. The performance of

even minor maintenance at the reactor must be preceded by estimates of the radiation exposure expected, based on dose-rate measurements and estimates of the time expected to be spent at the exposed location. The health-physics aspects of such work will be under the direction of the Rad-Safe Officer.

Generally, the reactor supervisor, the reactor operator, and the instrument engineer will be expected to perform such maintenance. It is important, however, that these people reserve a sizable portion of their total allotted exposure for making later emergency repairs that may require considerable knowledge of the reactor. It is safe, therefore, to assume that certain specific tasks at the reactor may be performed by field personnel relatively unfamiliar with the reactor, acting under instructions from the reactor supervisor.

(vi) Major Maintenance. Work at the reactor which would otherwise result in overexposure can be performed only after first removing the reactor core. The removal of the core is a major task, and the decision to remove the core must not be reached lightly or without a careful consideration of all the facts. Some of the hazards and problems to be expected are as follows:

1. Radiation exposure to those removing and replacing the core.
2. Nonnuclear hazard to personnel working with this heavy core (over 200 lb) on a small platform 31 ft above the ground.
3. Possible damage to the delicate thermocouples attached to the core.
4. Locating a crane (and operators) to lower the core to the ground (if this is the best method).
5. Locating a storage place for the core which is suitable from the standpoints of security, normal radiation hazard, and excursion hazard (e.g., flooding).

The following check list, though not complete, may be helpful if major maintenance makes removal of the core necessary.

MAJOR-MAINTENANCE PROCEDURE

1. Make sure proper Rad-Safe supervision is present.
2. Make sure that the safety block and all three rods are in their withdrawn positions.
3. Make sure that the reactor, the hoist, and the elevator are inoperable. (Carry the keys in your pocket.)
4. Cut off all power to the reactor.
5. Take the reactor-handling tools to the 31-ft-high platform.
6. Remove the hoist-cab walls (sheet-metal dust covers) as necessary.
7. Remove the uranium—molybdenum glory-hole specimen, if present, and put it in a safe place.
8. Remove the safety-block guard.
9. Remove the safety block from the reactor.
10. Take the safety block from the 31-ft-high platform,* place it in a properly marked container, and store it in a prearranged safe location. Rope off the area, post RADIATION HAZARD signs, and request the Technical Director to provide the proper security guard.
11. Collect the safety-block spring and all lock pins and other small parts in a box so they will not get lost. Remember that these nonuranium parts could present both a radiation hazard and a contamination hazard because of induced activity.
12. Move the safety-block drive shaft to its inserted position.
13. Disconnect the thermocouples.
14. Carefully position the hydraulic core jack and raise it lightly against the core.
15. Remove the three bolts that fasten the core-supporting lugs to the reactor frame.
16. Lower the core by lowering the hydraulic jack.
17. Carefully wheel the jack from under the reactor.
18. Remove the core from the jack by lifting it by its mounting lugs.
19. Remove the core from the 31-ft-high platform and store it in the prearranged safe location, making sure that a critical-mass hazard is impossible.

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* Placing the core in the personnel elevator and raising it (remotely) to about 200 ft may be worth consideration as a core-storage method.

20. Consider performing the needed reactor maintenance with the three remaining rods in place. If radiation measurements at this stage and repair-time estimates indicate that the radiation level is still too high, proceed to remove the three remaining rods.
21. Removal of these three remaining uranium–molybdenum pieces requires moving the rods to their inserted positions, which, in turn, requires electric (and air) power. At this stage the assembly remaining in the hoist cab is no longer considered a reactor because the core has been removed. Move the three rods to their insert limits, and remove the rods from the reactor.
22. Store the three rods in the prearranged safe location.
23. Perform the major repair that required the core removal.
24. Make a complete inspection of all the reactor machinery and electrical components located at the hoist cab. This is a rare opportunity to make a deliberate detailed study of rectifiers, ion chambers, preamplifier-cabinet items, limit switches, rod drives, etc., without the presence of the gamma-radiation field.
25. When it is ascertained that all this work is finished, restore the reactor circuitry to its normal condition.
26. While the core is still off, proceed through the formal reactor-operating procedures, for both steady-state and burst operations, to demonstrate that the controls are working properly in every respect.
27. Reinstall the core and its auxiliaries, following roughly the disassembly procedure in the reverse order.

3.2.5 Storage

At the termination of each operating day, the reactor will be shut down as described in Sec. 3.2.4. The mass-adjustment rod, the control rod, and the safety block will be in the withdrawn positions, and the neutron source will be in the inserted position. Leaving the neutron source in the inserted position is an additional safety measure because this procedure will prevent step insertion of excess reactivity before a sustained fission chain would initiate. The personnel hazard associated with the neutron source is low. (A polonium–beryllium source of 1.0×10^5 neutrons/sec total emission yields approximately 0.01 millirad/hr at 1 meter.)

At the end of each operating day, the protective covers will be left in place, enclosing the reactor hoist car. The car will be positioned at an elevation at which the tower has been reinforced for safe storage in the event of unfavorable weather conditions, and the hoist controls will be locked. The key will be under the control of the Technical Director. Access to the tower will be controlled by the Technical Director and will be physically restricted by locked personnel-elevator controls and barricaded ladder. Storage of the reactor in a partly elevated position will minimize the fission-product gamma-ray-exposure dose rate in the area around the base of the tower.

3.2.6 Qualifications of Reactor Operators

A minimum of two persons must be in the control room for operation of the HPRR. One of these persons must be a reactor supervisor. The second person, a reactor operator or trainee, will report to the reactor supervisor and will be qualified to shut down the reactor in the event of sudden incapacity of the supervisor.

The person responsible for the assembly, operation, and disassembly of the reactor at NTS will be L. W. Gilley. Mr. Gilley, an associate physicist at ORNL, has an M.S. degree in physics, has attended the Oak Ridge School of Reactor Technology, and has worked for the past nine years at the ORNL Critical Experiments Facility. In particular, he has participated in the preliminary critical experiments as well as the critical experiments, calibration experiments, and burst experiments conducted with the reactor at the Critical Experiments Facility.

Mr. Gilley will normally be the reactor supervisor. Assisting Mr. Gilley as reactor operator and also qualified to serve as reactor supervisor in the event of Mr. Gilley's absence will be D. R. Ward. Mr. Ward, a development engineer at ORNL, has a B.S. degree in engineering, has attended the Oak Ridge School of Reactor Technology, and has operated reactors for the past $2\frac{1}{2}$ years at the ORNL Tower Shielding Facility. He has participated in the preliminary testing of the HPRR at the ORNL Critical Experiments Facility.

Qualified as reactor operator will be J. R. Hill. Mr. Hill, an engineer in the ORNL Instrumentation and Controls Division, has an M.S. degree in nuclear engineering and has participated in the preliminary testing of the HPRR with particular emphasis on the instrumentation.

Operation BREN is not to be considered a training program for reactor operators; however, if the reactor supervisor decides that the reactor operation will be sufficiently routine and trouble free, he may check out assistant operators as needed. (Burst operation will require the presence of Mr. Gilley and also Mr. Ward or Mr. Hill.) Persons who may be trained as assistant reactor operators include J. A. Auxier, J. S. Cheka, F. F. Haywood, F. W. Sanders, and J. H. Thorngate, all of the ORNL Health Physics Division. All these persons have M.S. or B.S. degrees in physics and have had several years of contact with reactor operation.

The assistant operator may operate the reactor in the steady-state mode in the presence of a reactor supervisor. An individual will be qualified as an assistant when he can demonstrate a knowledge of the reactor shutdown procedures as outlined in Sec. 3.2.4(b), Steady-state-operation Procedure, Group F: Shutting Down. The reactor supervisor will indicate in the permanent HPRR Log Book when an individual is qualified as an assistant operator.

3.3 THE 1200-CURIE Co^{60} GAMMA-RAY SOURCE

A 1200-curie Co^{60} gamma-ray source (Figs. 3.18 and 3.19) is to be used in Operation BREN for the purpose of making measurements of dose from a gamma source positioned at a height of up to 1500 ft above the air–ground interface.

The source facility will be placed on the tower hoist car after the HPRR has been removed. The source will be positioned in two phases. In the first phase the hoist car will be raised to the scheduled height; in the second phase the source will be raised from the shield (Fig. 3.20) by operating controls in the bunker control room.

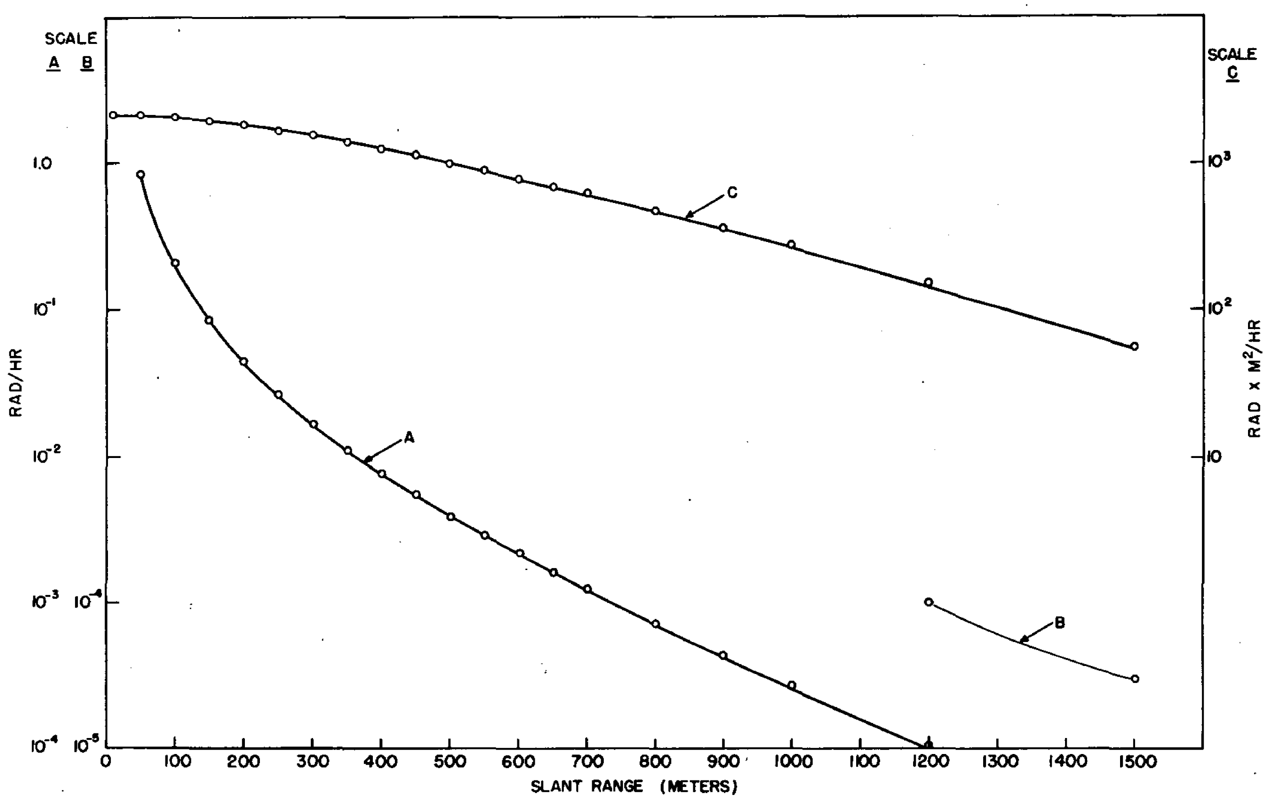

Fig. 3.18—Dose rate vs. distance for 1200-curie Co^{60} gamma-ray source.

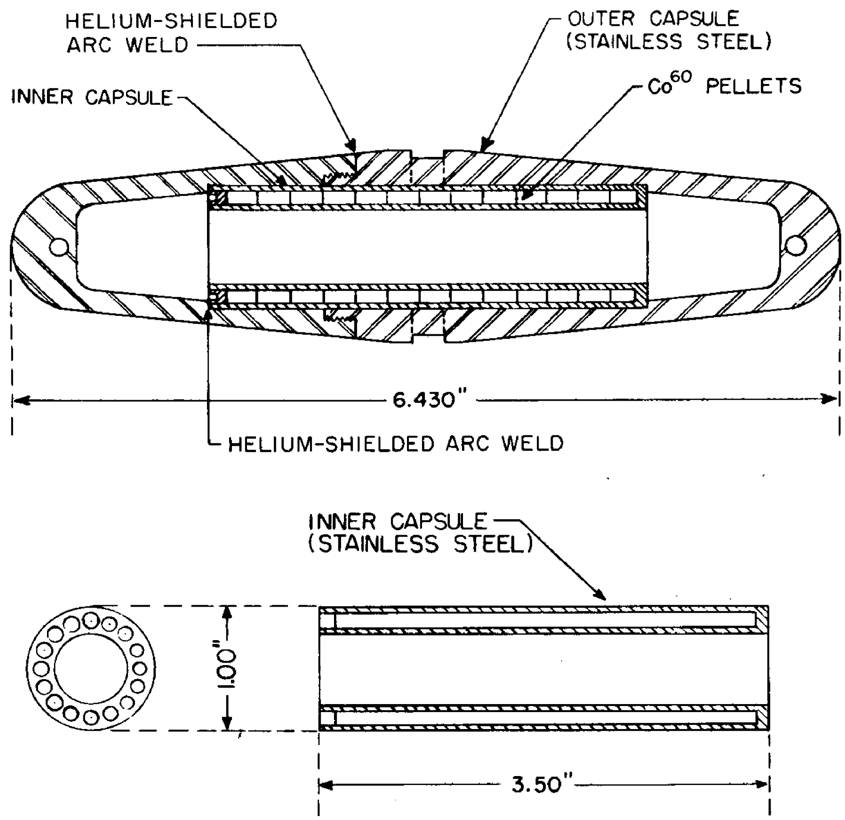

Fig. 3.19 — Co^{60}-source container.

The source-utilization schedule will be compatible with the requirements of Program 1, and other Program Directors will be requested to conform to this schedule.

3.3.1 Description

The gamma-source facility (Fig. 3.21) will consist of the following components:

1. A 3300-lb lead source container.
2. A 1200-curie Co^{60} source connected by a short steel cable to the bottom of the removable plug in top of shield.
3. Structural support for lifting mechanism and source suspension while source is out of the shield.
4. Gear-head motor (24 volts direct current, 5.5 amp, 9000 rpm, 1500–1 gear reduction, 72 watts).
5. Magnetic clutch–brake assembly, which, while energized, will provide coupling to gear-motor shaft for lifting source from shield. When motor is not running, the shaft is held immobile by an internal brake. This property allows the clutch–brake to hold the source suspended. When the clutch is purposely de-energized or when power to the clutch fails, the source and plug return slowly to the shield by gravity.
6. Microswitches, positioned so as to limit the upward and the downward travel. There are three switches in each position; dual tandem switches control the motor, and the third switch controls the limit lights.
7. Lifting cable and sheave.
8. Electrical circuits to furnish power to the clutch–brake, the gear motor, the limit lights, the siren alarm, and a warning light near the tower base.

Safety Features. The gamma-source facility is engineered and built to be fail safe. The features that ensure this safety are itemized as follows:

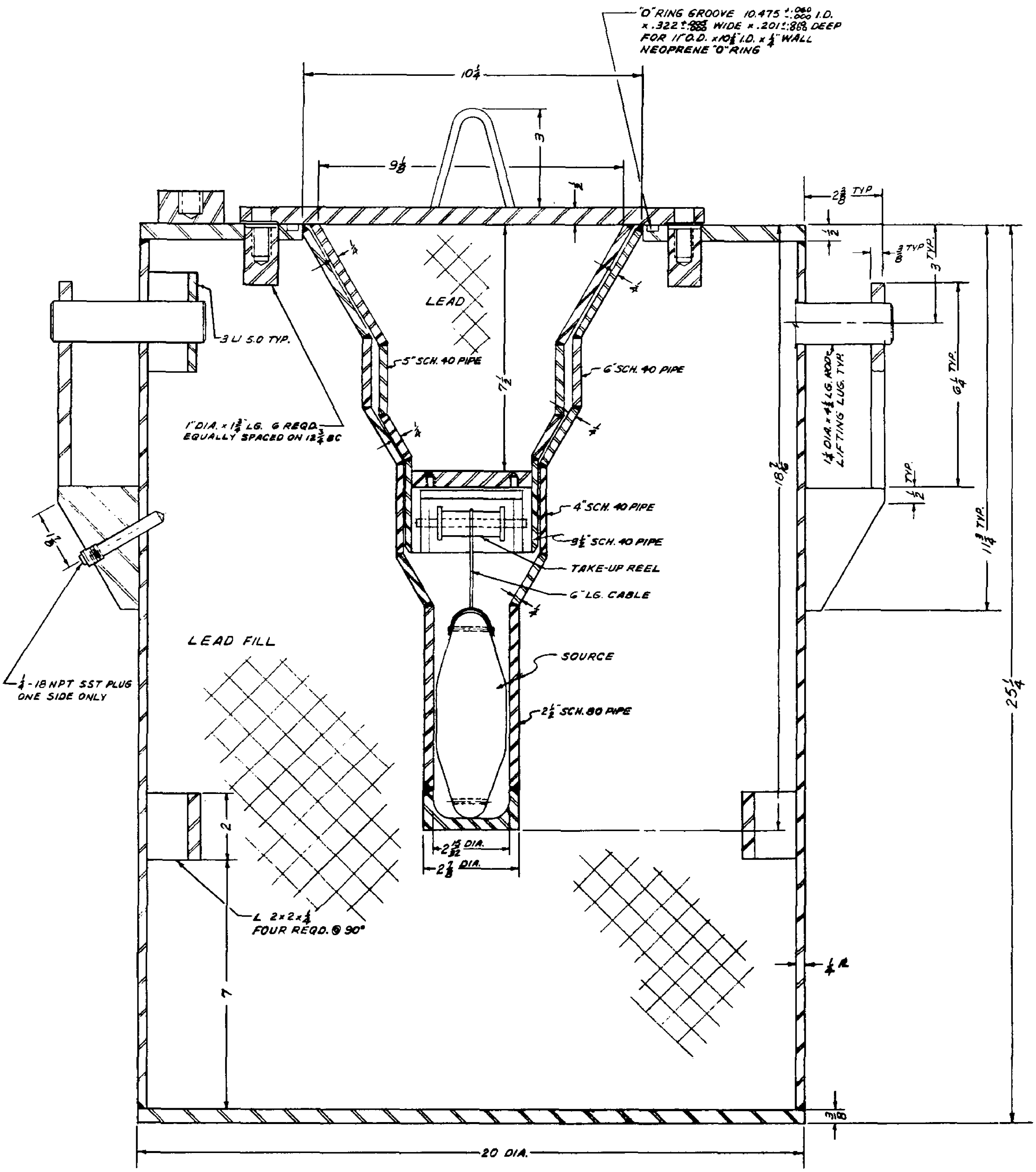

Fig. 3.20—Co^{60}-source shield. (All dimensions are in inches.)

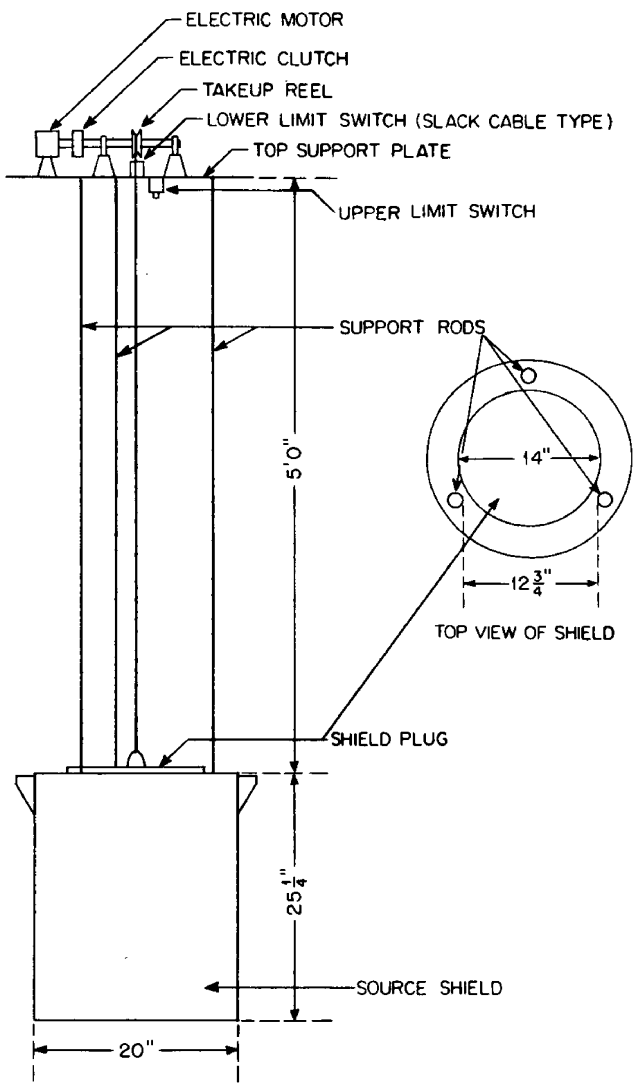

Fig. 3.21—Co^{60}-source lifting mechanism.

1. Six limit switches are used, three to limit the upward travel of the source and three to limit the downward travel.

For the upper limit, two single-pole, single-throw, normally closed switches control the motor, and one single-pole, single-throw, normally closed switch controls the limit light. The two switches for the motor are connected in series, and this offers a double safety. There is a fuse in the RAISE side of the motor circuit. If both of the motor upper-limit switches should fail, the plug would stop on reaching the base plate that supports the lifting mechanism because the current in the motor circuit would increase and blow the fuse, thus stopping the motor.

For the lower limit, two single-pole single-throw, normally open series connected switches control the motor, and a single-pole double-throw, normally closed switch controls the light. These will stop the motor when the cable is slack and will turn on a lower-limit light.

2. If electric power to the motor or clutch–brake fails, the clutch will become de-energized, and the source will be returned to the shield by gravity.

3. The motor contains an internal magnetic brake that does not allow the gear-head shaft to turn unless the motor is running. As long as the clutch–brake is energized, the source is held in the desired position.

4. The plug, of tapered stepped design, is guided by the lifting-mechanism base-plate support rods, which are sufficiently close together to prevent excessive misalignment of the plug and source.

3.3.2 Shipping Methods

The 1200-curie Co^{60} gamma-ray source was transported from ORNL to NTS and will be returned to ORNL by trailer truck. The source is contained in a shield approved by the Bureau of Explosives. This facility and other equipment shipped at the same time comprised a full truck load, and the entire shipment was loaded and unloaded within the regulated areas of ORNL and NTS.

3.3.3 Installation

After removal of the HPRR, complete with relay rack, from the source hoist car, the Co^{60} source and shield will be placed in the lower compartment of the car. Several of the conductors in the HPRR control cables will be used to provide electrical power for operation of the source-raising device.

3.3.4 Operating Procedures

The operator of the gamma facility will follow each step of the procedure given in Fig. 3.22 and will initial each accompanying blank.

3.3.5 Storage

At the termination of each operating day, the 1200-curie Co^{60} gamma-ray source will be returned to its shield as instructed in the check list (Fig. 3.22), and the hoist car will be moved to its lowest position. The shield plug will be secured by passing a chain through the lifting eye and the main-shield lifting lugs and fastening with a padlock. The key will be controlled by the Technical Director.

After the source shield is locked, the protective covers will be replaced, and the hoist car will be placed in its storage position as described in Sec. 3.2.5.

3.4 CALIBRATION SOURCES

The relatively small radiation sources used by personnel of the several programs in the calibration of instruments will not be under the direct control of the Technical Director. However, they must be registered with the Operation BREN Rad-Safe Officer and with the NTS Rad-Safe Officer, and they must be transported and stored according to applicable regulations.

REFERENCES

1. W. E. Kinney and J. T. Mihalczo, *ORNL Fast Burst Reactor—Critical Experiments and Calculations*, USAEC Report CF-61-8-71, Oak Ridge National Laboratory, August 1961.
2. G. R. Keepin, T. F. Wimett, and R. K. Zeigler, Delayed Neutrons from Fissionable Isotopes of Uranium, Plutonium, and Thorium, *Phys. Rev., 107:* 1044 (1957).

Date________________________

Expected duration of exposure________________________

Time source was raised________________________

____ 1. Initiate area clearance, which shall be concluded by H–15 min.

____ 2. Physical inspection of the gamma-source facility will be made by the Rad-Safe Officer or his representative and shall be concluded by H–15 min.
- ____ a. Confirm locking chain over plug (source cannot be removed).
- ____ b. Gear-motor mounts tight.
- ____ c. Microswitch mounts tight.
- ____ d. Lifting-cable connections disconnected from plug.
- ____ e. Clutch–brake pressure-plate bolts tight.
- ____ f. All a-c and d-c connections secure.

____ 3. The operator in the bunker control room will be requested to assist in the following duties and verify by intercommunication:
- ____ a. Clutch switch is in DE-ENERGIZE position.
- ____ b. Motor switch is in OFF position.
- ____ c. Turn power and warning light on.
- ____ d. Energize clutch.
- ____ e. Turn motor switch to RAISE.
- ____ f. Energize upper-limit switch. The motor stops and the upper-limit light turns on.
- ____ g. Energize the lower-limit microswitch. This will allow the motor to move toward the lower limit.
- ____ h. Turn motor switch to LOWER.
- ____ i. De-energize the lower-limit microswitch. The motor will stop and the lower-limit light will turn on.
- ____ j. Turn off clutch.
- ____ k. Turn power and warning light off.
- ____ l. Remove locking chain from top plug (source can be removed).
- ____ m. Reattach lifting cable to plug.

____ 4. The person making the inspection will at this time return to the bunker and inform the Technical Director or his representative that the gamma-source facility is ready for operation. At H–15 min, the tower siren will sound.

____ 5. Place the hoist car at its designated elevation. The Technical Director or his representative will authorize the removal of the source.

____ 6. The operator will:
- ____ a. Turn power and warning light on.
- ____ b. Energize clutch.
- ____ c. Turn motor switch to RAISE. The source will then be lifted out of the lead shield. The lower-limit light will go out. The source is at its designated height when the upper-limit light turns on.
- ____ d. Turn motor switch to OFF.
- ____ e. Note time:________________________.

____ 7. Personnel in the experimental area will be notified that the source is out. They will again be notified when the source has been returned to the shield.

____ 8. At the end of the exposure, the operator will:
- ____ a. Note time:________________________.
- ____ b. Turn motor switch to LOWER. Upper-limit light turns off.
- ____ c. When lower-limit light turns on, turn motor switch to OFF.
- ____ d. De-energize clutch.
- ____ e. Turn power and warning light off.

____ 9. The Rad-Safe Officer or his representative will monitor the background outside the control-room bunker.

____10. Operator will return hoist car to base of tower if the Technical Director desires.

____11. After hoist car descends, area will be monitored by Rad-Safe Officer or his representative.

____12. Technical Director or his representative will authorize lifting of access restrictions to the tower area.

Fig. 3.22—Check list for operation of Co^{60} gamma facility.

Chapter 4

OBJECTIVES

4.1 GENERAL OBJECTIVES

The experiments for which Operation BREN was initiated have been organized as Program 1 and will be carried out by the ORNL Health Physics Division. Program 2 will be conducted by ORNL Health Physics Division personnel after completion of all the experiments of Program 1.

Since the needs of Program 1 are basic to Operation BREN, scheduling of all operational events, to include timing of experiments, reactor power levels, working hours, etc., will be adjusted to the requirements of Program 1. Selected experiments proposed by other organizations for which the radiation fields available during Operation BREN can be used on a noninterference basis can be incorporated into Operation BREN.

Program 1 is part of a study, begun in 1956, aimed at determining the radiation doses received by the survivors of the nuclear bombings of Hiroshima and Nagasaki, Japan. Neutron and gamma-ray shielding factors applicable to many of the structures in these Japanese cities will be determined in the course of the experiments. In addition, much of the work is of fundamental nature and is applicable to general radiation-physics problems.

4.2 SPECIFIC OBJECTIVES

The specific objectives of Program 1 are listed in detail, and less detailed descriptions of the objectives of other programs are included in this section.

4.2.1 Program 1: Spectra, Distribution, and Attenuation of Mixed Radiation

The specific objectives and experiments of Program 1 are distributed among four projects, denoted Projects 1.1, 1.2, 1.3, and 1.4. It is expected that there will be a maximum of cooperation and sharing of facilities among these projects. The objectives of these four projects are discussed in the following sections.

(a) *Project 1.1: Gamma-ray Dose Measurements in Houses.* Project 1.1 consists in the measurement of gamma-ray doses (1) in and around facsimiles of Japanese houses and (2) in air unperturbed by shielding at the same distance from the radiation source as the houses.

(b) *Project 1.2: Energy and Angular Measurements Made with Collimators.* The scope of Project 1.2 is defined as follows:

1. Measurement of gamma-ray dose as a function of angle of incidence.
2. Measurement of neutron dose as a function of angle of incidence.
3. Measurement of the spectrum of neutrons leaking from the reactor during steady-power-level operation.

4. Measurement of the spectrum of gamma rays due to operation of the reactor at a steady power level.

5. Measurement of the spectrum of gamma rays emitted by short-lived fission products as a function of time after operation of the reactor in the burst mode.

6. Measurement of the spectrum of gamma rays emitted by a point Co^{60} source as a function of angle of incidence.

7. Measurement of the gamma-ray dose from a point Co^{60} source as a function of angle of incidence.

(c) *Project 1.3: Neutron Dose Measurements in Houses.* Project 1.3 consists in the measurement of neutron doses (1) in and around facsimiles of Japanese houses and (2) in air unperturbed by shielding at the same distance from the radiation source as the houses.

(d) *Project 1.4: Energy and Dose Measurements as Functions of Distance and Height.* The scope of Project 1.4 is defined as follows:

1. Measurement of gamma-ray dose as a function of horizontal distance from the reactor tower and distance above or below the air–ground interface.

2. Measurement of neutron dose as a function of horizontal distance from the reactor tower and distance above or below the air–ground interface.

3. Measurement of neutron spectrum as a function of horizontal distance from the reactor tower and distance above or below the air–ground interface.

4.2.2 Program 2: Gamma-ray Measurements, Spectra, and Distribution

Program 2 is an extension of aerial radiometric studies performed at NTS in 1960 over the CETO Extended Source Calibration Area (ESCA). The ESCA will be rebuilt, with one corner at the base of the Operation BREN tower. Radiation detectors will be installed in the hoist car and located at elevations between the ground and the top of the tower. The results will permit comparison with 1960 data for use in further development of aerial radiometric equipment and calibration thereof over a large area of uniform contamination.

4.2.3 Program 3: Evaluation of Radiation Shields

The general objectives of Program 3 are to evaluate the shielding characteristics of various materials in various configurations exposed to mixed fields of neutrons and gamma rays. This study will extend a program of evaluation conducted at ORNL.

4.2.4 Program 4: Neutron Field, Spectrum, and Depth–Dose Studies

The general objectives of Program 4 are to measure the spectrum, flux, and dose rates of neutrons from an unshielded reactor supported on a tower. Measurements will be made at ground level to a horizontal distance of 1500 yd from a point under the reactor. Foil-activation and counting techniques will be used in addition to BF_3 and polyethylene-lined proportional counters and nuclear-emission film. Depth–dose measurements will be made in phantoms.

4.2.5 Program 5: Neutron Field and Induced-activity Measurements

The general objectives of Program 5 are (1) to determine some of the characteristics of the neutron distribution in soil (e.g., fast-neutron to thermal-neutron ratios) which result from operation of an unshielded reactor, mounted on a tower, at several heights above the ground as a function of horizontal distance from a point directly under the reactor and as a function of depth in soil and (2) to determine the induced radioactivity in samples of selected materials as a function of the same parameters.

4.2.6 Program 6: Radiation Effects on Electronic Components

The general objectives of Program 6 are to determine the effects of mixed fields of neutrons and gamma rays on electronic components of interest in missile systems. This includes the effects that pulses of radiation (i.e., dose rate), integrated exposure, and the electromagnetic pulse associated with a nuclear pulse have on the performance of these electronic components.

4.2.7 Program 7: Comparative Dosimetry Study with Miniature Ionization Chambers

The experimental objective of Program 7 is to obtain total radiation dose as a function of depth with the use of tissue-equivalent phantoms. Tissue-equivalent ionization chambers, nuclear emulsions, and films submerged in the phantoms will be used to measure total radiation dose.

Chapter 5

OPERATING PROCEDURES

5.1 GENERAL OPERATING PROCEDURES

The ORNL Health Physics Research Reactor will be mounted on the source hoist car on a 1500-ft tower in Area 4 at NTS (Figs. 5.1 and 5.2). The reactor will be the source of neutrons and gamma rays and will be operated at a power level of up to about 10 kw. During the late phases of the study, the reactor will be pulsed (about 10^{17} fissions per pulse), and the spectrum of the fission-product gamma rays will be studied as a function of time after burst. Several facsimiles of Japanese houses of the type used during Operation Hardtack (Phase II) will be used in the studies.

Collimated detectors will be used to study the angular dependence of the radiation field relative to the line between the detector and the reactor.

At the conclusion of measurements of mixed radiation, the HPRR will be replaced by a 1200-curie Co^{60} gamma-ray source for study of the gamma-ray field from a point source.

An area-clearance sweep will be concluded by 15 min before operation of the reactor (or the 1200-curie Co^{60} gamma-ray source) is initiated, and roads will all be barricaded by flashers and signs. An exclusion area of approximately 750 yd in radius, centered on the tower, will be marked off, and personnel will not be allowed above ground within this circle during reactor operation. A siren alarm at the tower will sound 15 min before operation commences.

Radiation-detector stations on the tower, at the bunker, and in the experimental area will be monitored continually during operation. Regular and special health-physics (Rad-Safe) monitoring of all participating personnel at NTS will be required throughout the period during which the experiments are in progress.

After shutdown, personnel will not be permitted within 500 yd of the reactor tower until the Technical Director or his representative designates at the conclusion of the health-physics survey. More proximal limits of access will be set, depending on radiological conditions, and access to an area in which the dose rate exceeds 30 mrem/hr will require specific approval of the Technical Director or the Deputy Technical Director, who will consult with the Rad-Safe Officer, Mr. Haywood.

All mechanical and experimental operations on the tower will be shut down if one of the following conditions exists.

1. The average wind velocity at any elevation exceeds (or is predicted to exceed within 4 hr) 30 mph.
2. The velocity of gusts of wind at any elevation exceeds (or is predicted to exceed within 4 hr) 40 mph.
3. Rain or snow is falling (or is predicted to fall within 4 hr).
4. Showers are visible within, or on the ridges surrounding, Yucca Flat.

No personnel will be allowed on the tower if one of the following conditions exists.

1. The average wind velocity at any elevation exceeds (or is predicted to exceed within 4 hr) 20 mph.

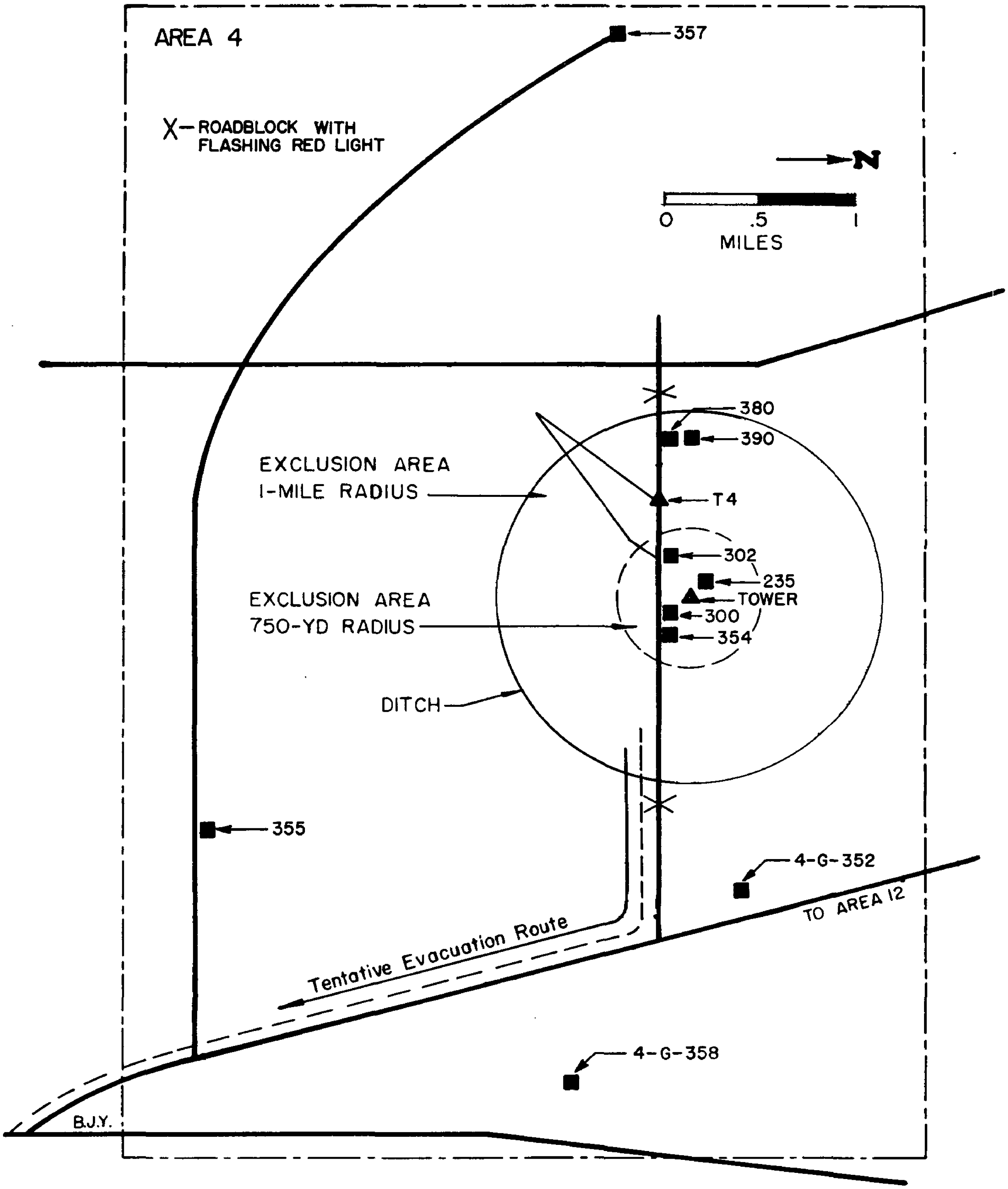

Fig. 5.1—NTS Area 4.

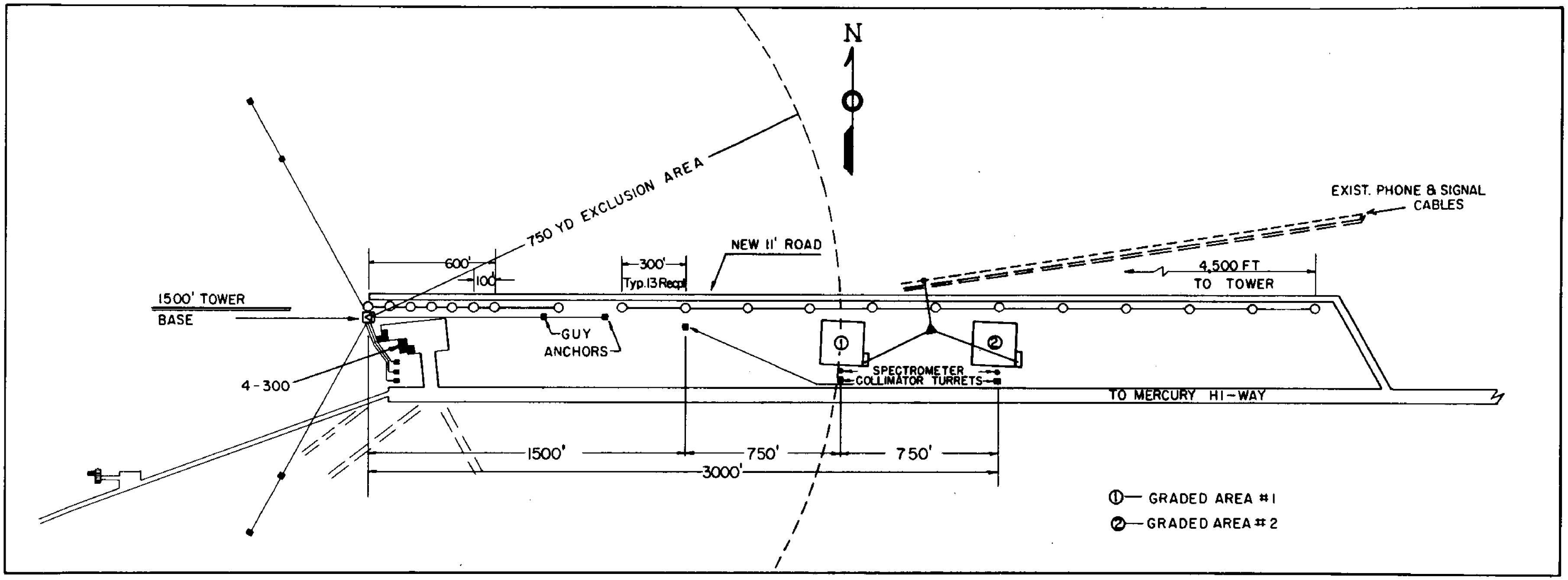

Fig. 5.2—Program 1 operational area.

2. The velocity of gusts of wind at any elevation exceeds (or is predicted to exceed within 4 hr) 30 mph.
3. Rain or snow is falling (or is predicted to fall within 4 hr) or the surface of the tower is wet.
4. Showers are visible within, or on the ridges surrounding, Yucca Flat.

5.2 PROGRAM 1: SPECTRA, DISTRIBUTION, AND ATTENUATION OF MIXED RADIATION

5.2.1 Project 1.1: Gamma-ray Dose Measurements in Houses

Three radiation-equivalent facsimiles of Japanese houses will be built, one each of types A, B, and C as used in Operation Hardtack (Phase II).[1] Initially, they will be positioned 750 yd from the tower. In addition, three wall sections will be built for use in certain house arrays. All these structures will be movable.

Measurements of both gamma and neutron radiation will be made in the houses above points laid out on the floor in a grid similar to that used in Operation Hardtack (Phase II). The grids for all houses are shown in Figs. 5.3 to 5.6.

Direct comparison can be made with neutron measurements in houses that were similarly placed in Operation Hardtack (Phase II). The neutron data from Operation BREN will serve as an extension of, and a supplement to, information obtained during Operations Hardtack (Phase II) and Plumbbob.[2] In addition, the gamma information, which was not obtainable in Operation Hardtack (Phase II), will be added to the pattern.

One house will be constructed concurrently with experimentation so that attenuation of radiation by roof and walls can be determined separately. Measurements will be made in houses placed both singly and in groups so that attenuation by an individual house can be determined as well as attenuation due to shielding by another structure or group of structures.

The proposed house configurations are shown in Figs. 5.7 to 5.14.

As a pattern emerges, it may become unnecessary to measure all stations in houses that are parts of groups.

If radiation intensity is sufficient to make measurement feasible, several of these configurations will be repeated at 1000 yd.

Instruments to be used include "Radsan" proportional counters to measure neutron dose,[3–5] "Phil" Geiger-Mueller (G-M) dosimeters to measure gamma dose,[6] and a "long counter" consisting of a BF_3 tube encased in paraffin. The long counter, which will be operated at a fixed location, will be used for normalizing the readings of other instruments to a "standard" reactor power. The proportional counters and G-M counters will be mounted in pairs on a stand so that the array can be positioned over any of the floor grid points in the houses or at specific locations outside.

Special power supplies and preamplifiers will be provided to enable the instruments to operate with 250-ft cables. The readout equipment will be located in a van type instrument trailer in a revetment behind the experimental area.

5.2.2 Project 1.2: Energy and Angular Measurements Made with Collimators

In the development of methods for calculating the shielding afforded by typical Japanese residences, it is necessary to obtain information on the fraction of neutron and gamma-ray dose which is incident at a point as a function of the angle of incidence. It is also necessary to have knowledge of the energy distribution of neutrons and gamma rays as a function of angle of incidence. Some data have been obtained during weapons tests,[1,2] but more are needed to complete the development of methods of calculating shielding factors. Measurements made during Operation BREN will extend and refine knowledge of neutron energy and angular distributions and will provide urgently needed data on gamma-ray energy and angular distributions.

Instruments to be used to measure neutron and gamma-ray dose during steady-power-level operation will be Radsan[3–5] proportional counters and Phil[6] G-M dosimeters, respectively. The Phil dosimeters will also be used to measure gamma-ray dose from the Co^{60} source.

(Text continues on page 79.)

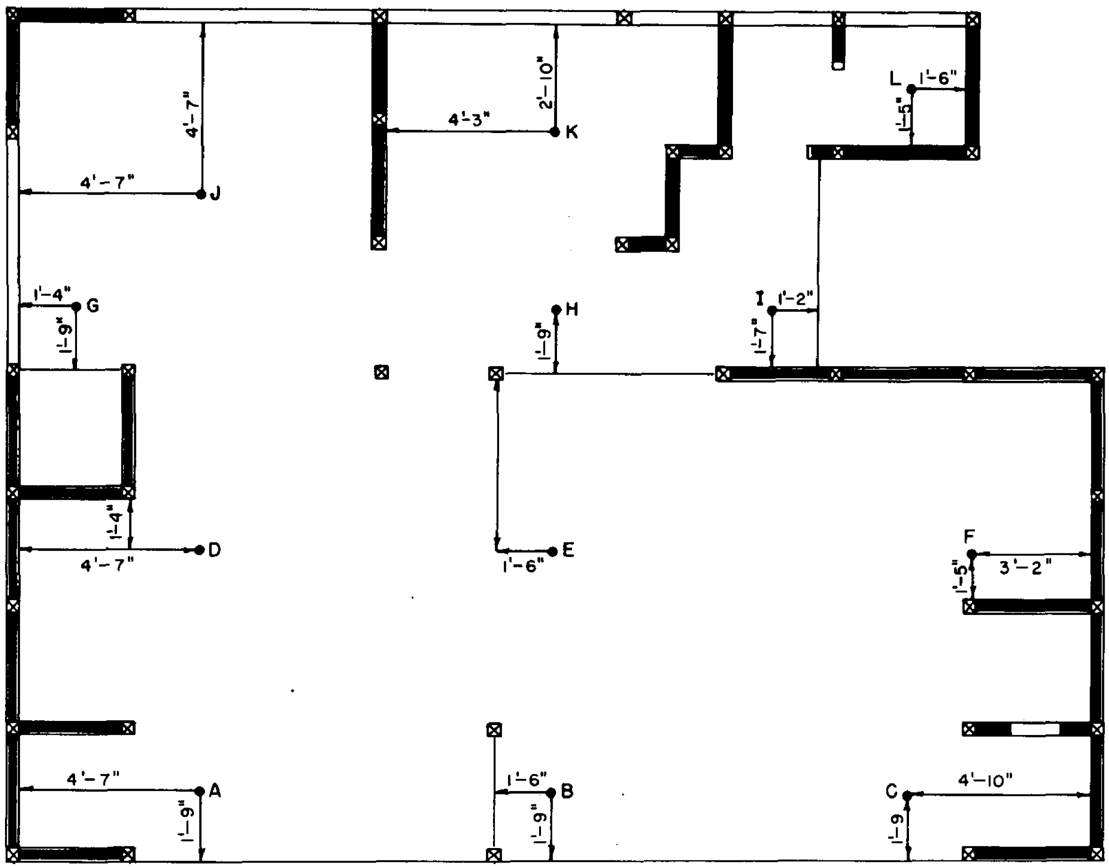

Fig. 5.3—Plan of type A Japanese-house facsimile.

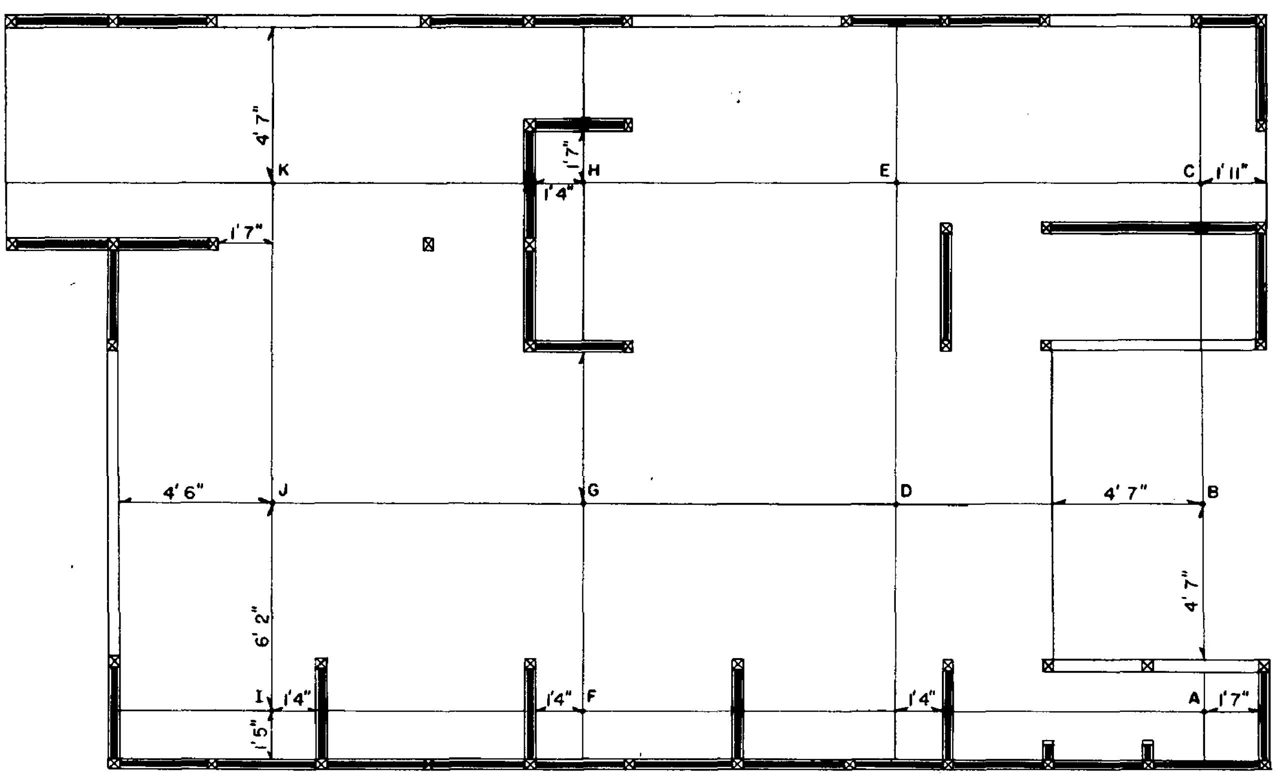

Fig. 5.4—First-floor plan of type B Japanese-house facsimile.

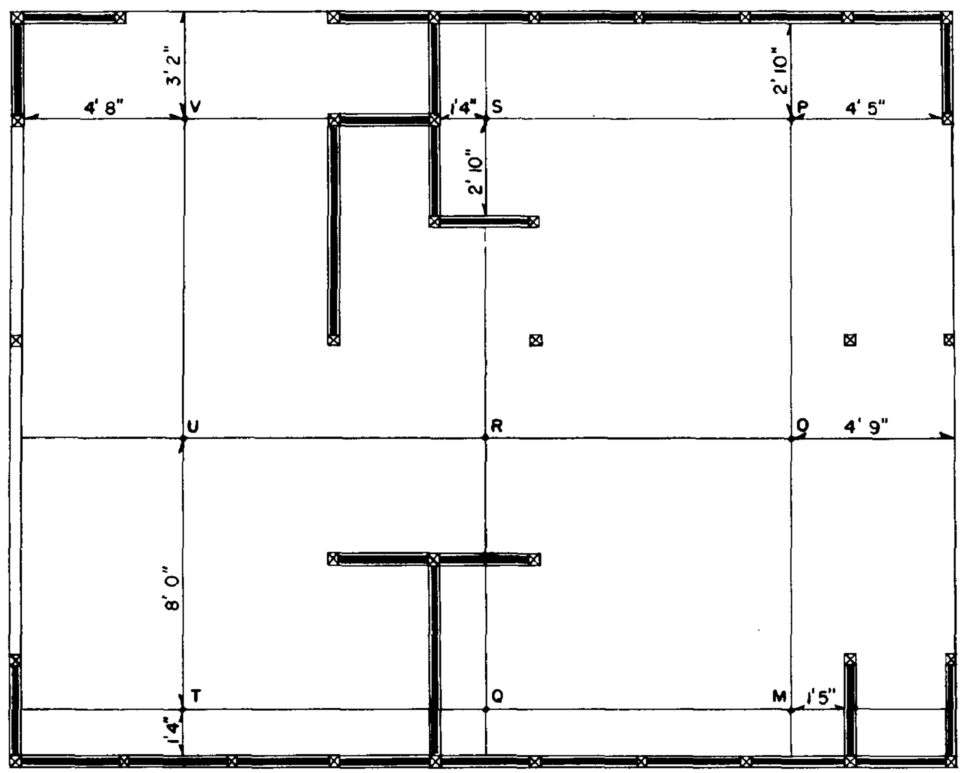

Fig. 5.5—Second-floor plan of type B Japanese-house facsimile.

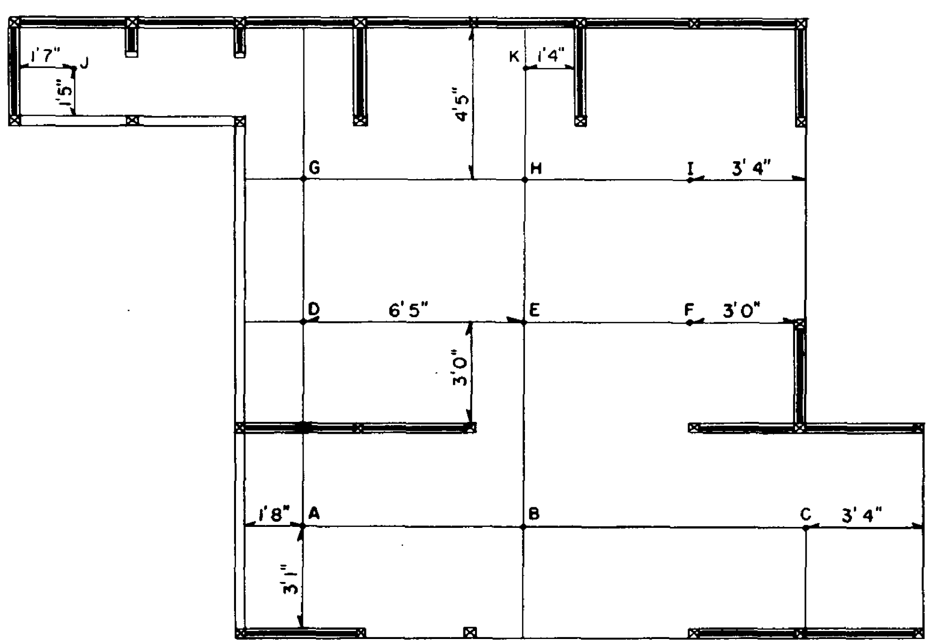

Fig. 5.6—Plan of type C Japanese-house facsimile.

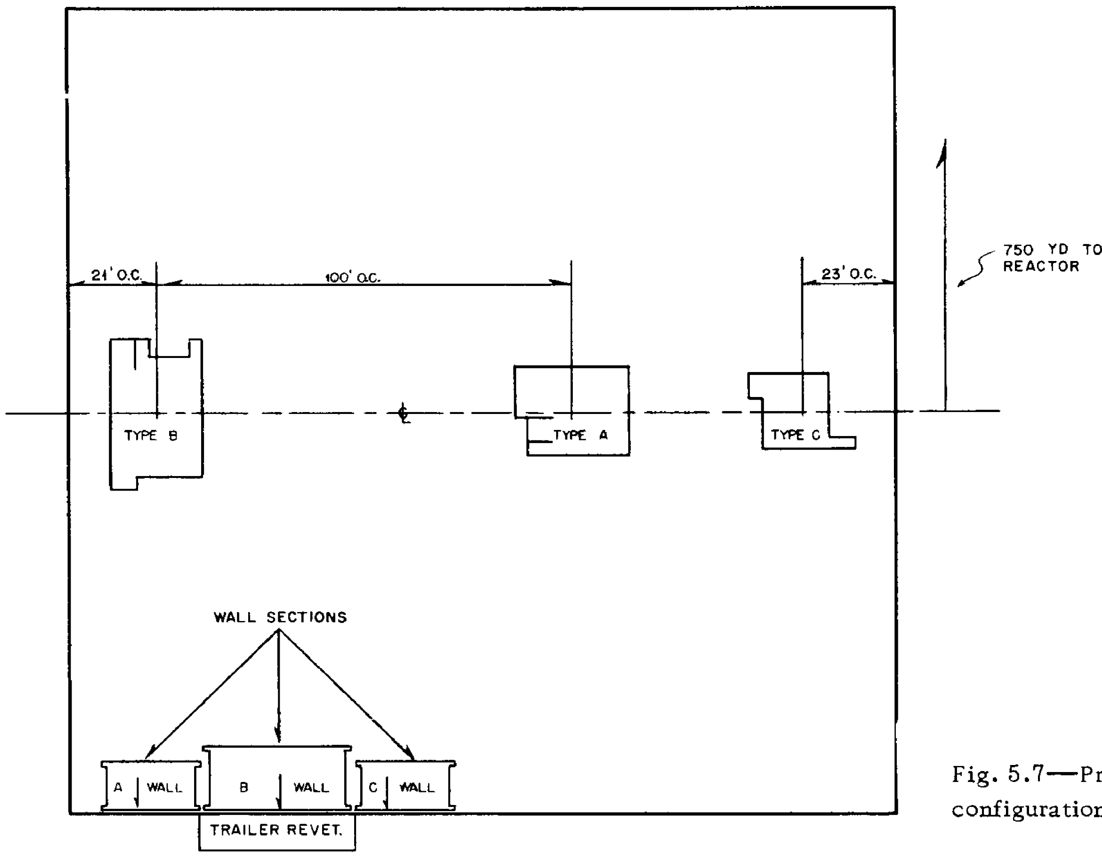

Fig. 5.7—Proposed house configuration 1.

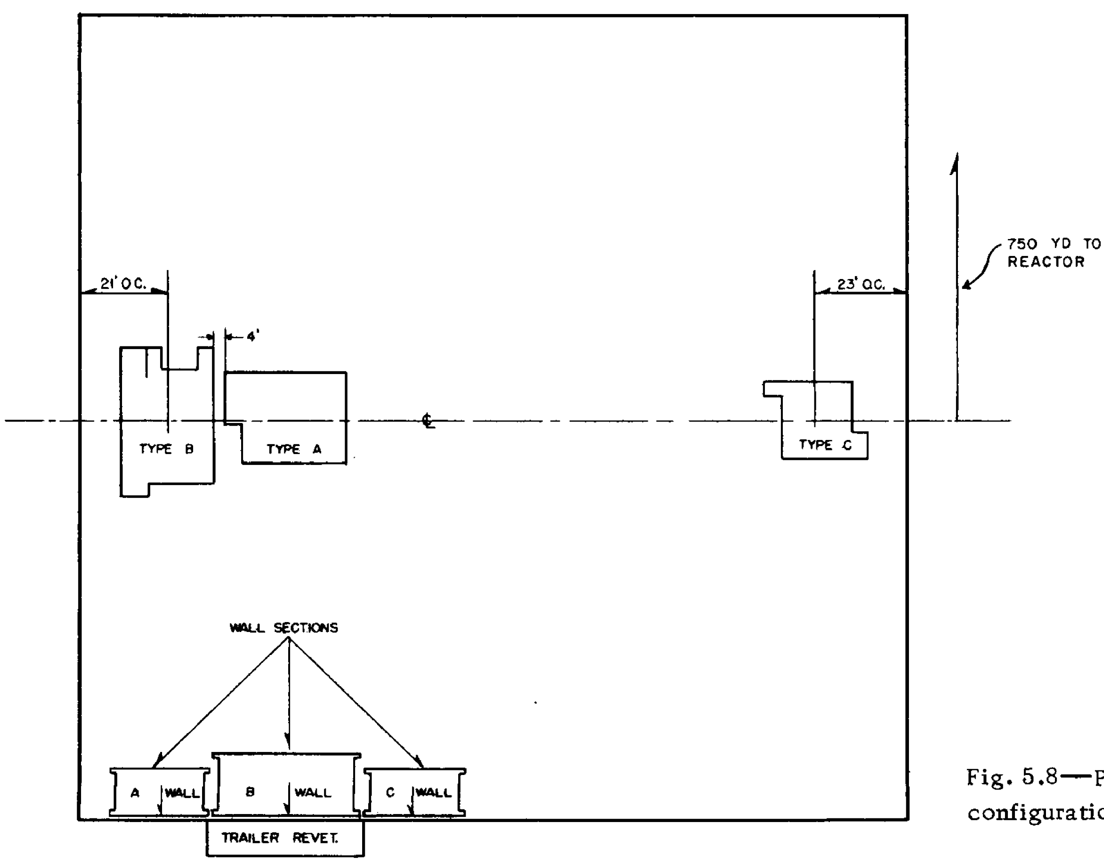

Fig. 5.8—Proposed house configuration 2.

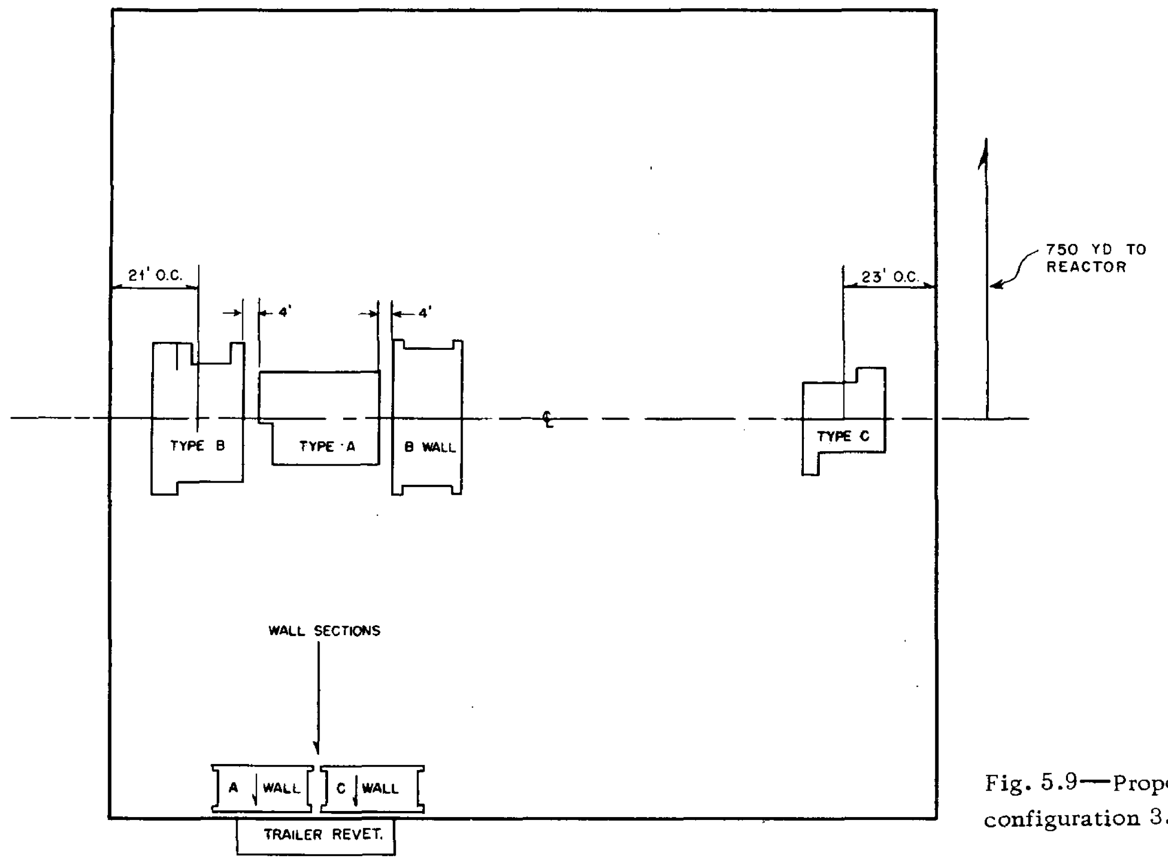

Fig. 5.9—Proposed house configuration 3.

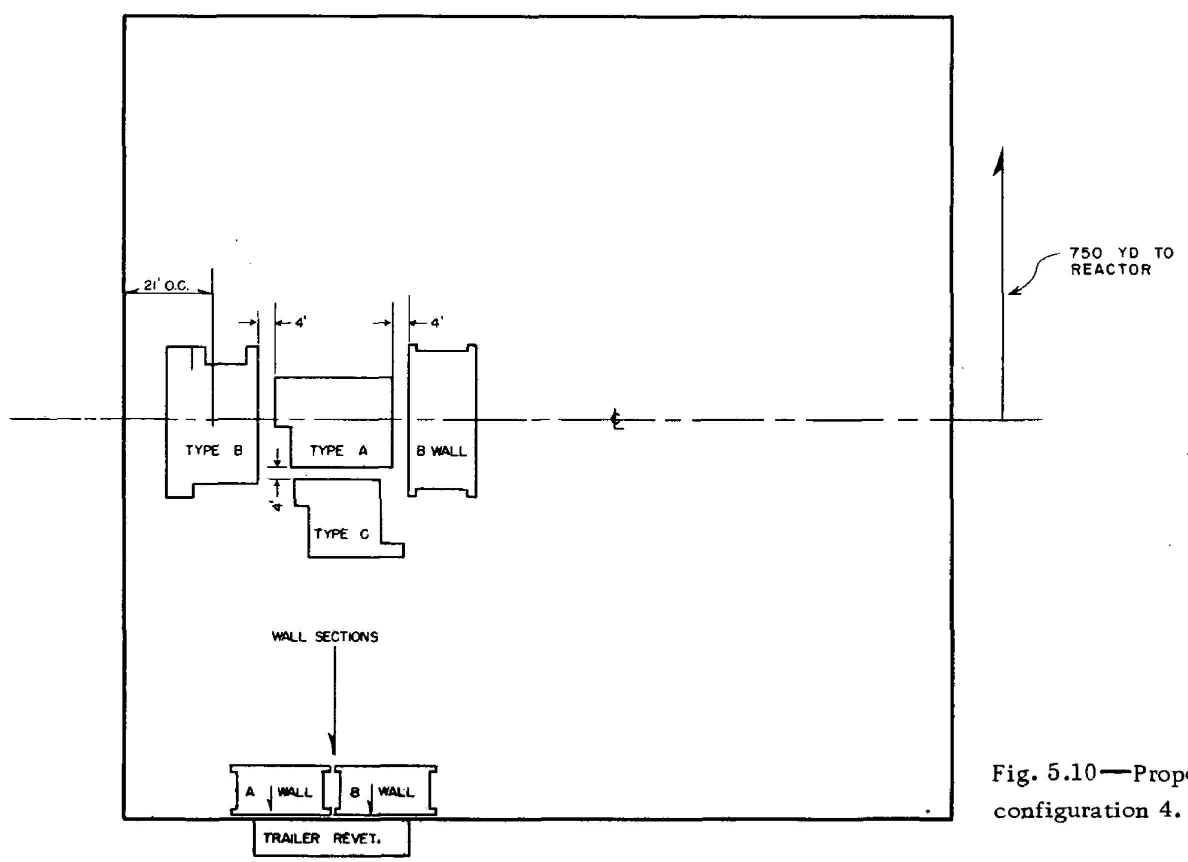

Fig. 5.10—Proposed house configuration 4.

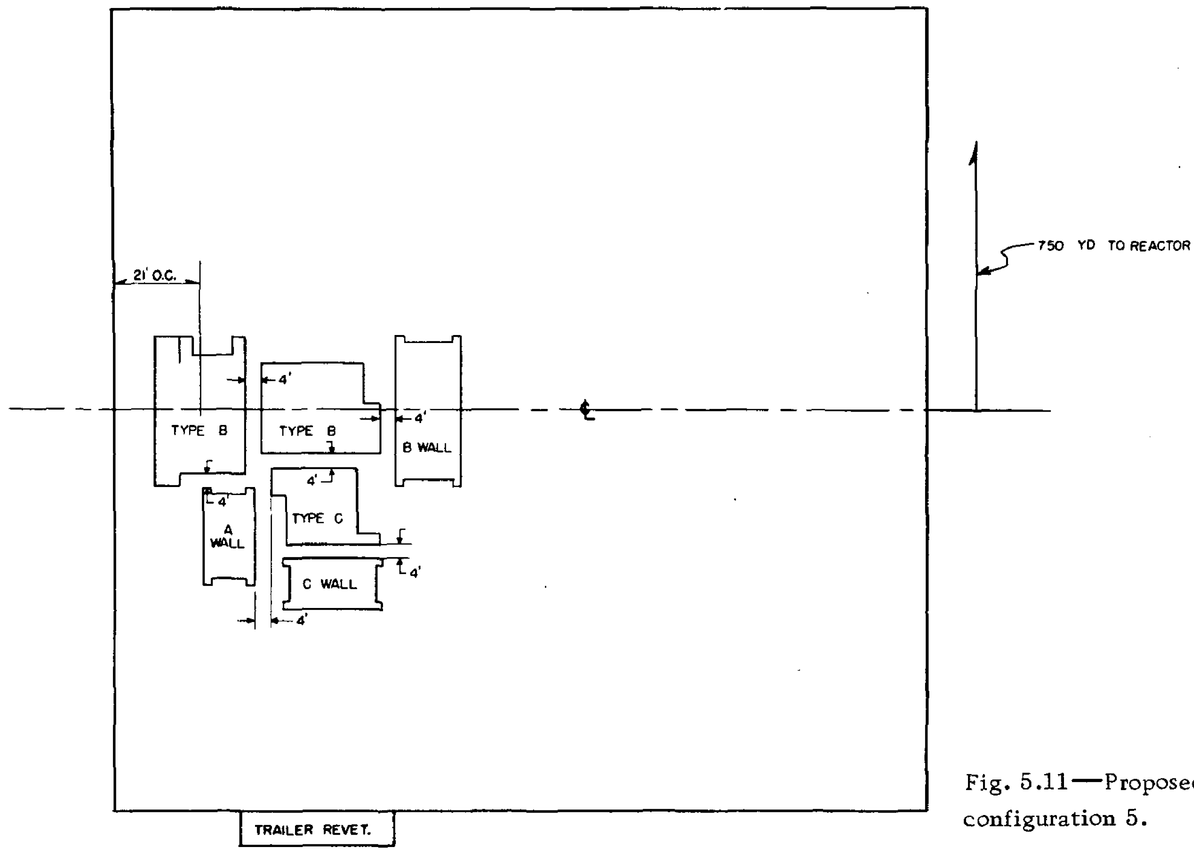

Fig. 5.11—Proposed house configuration 5.

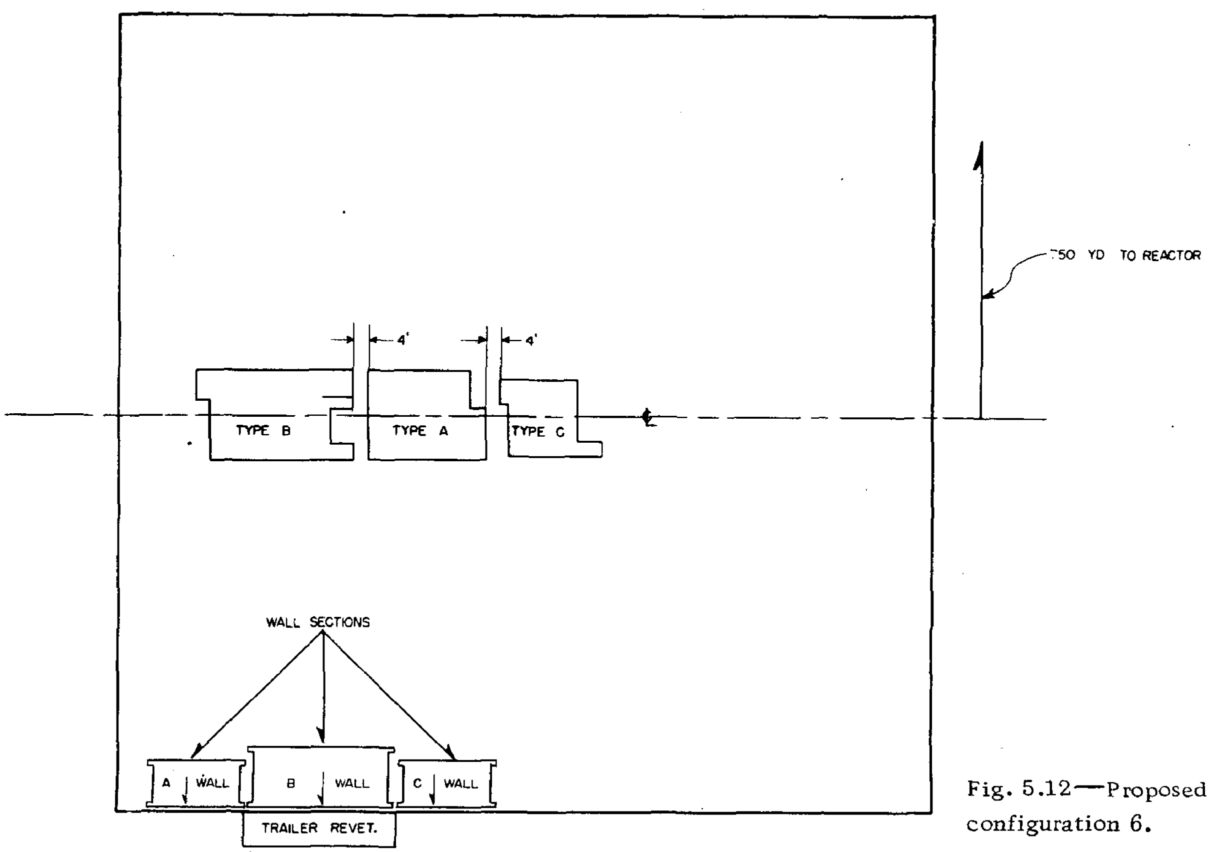

Fig. 5.12—Proposed house configuration 6.

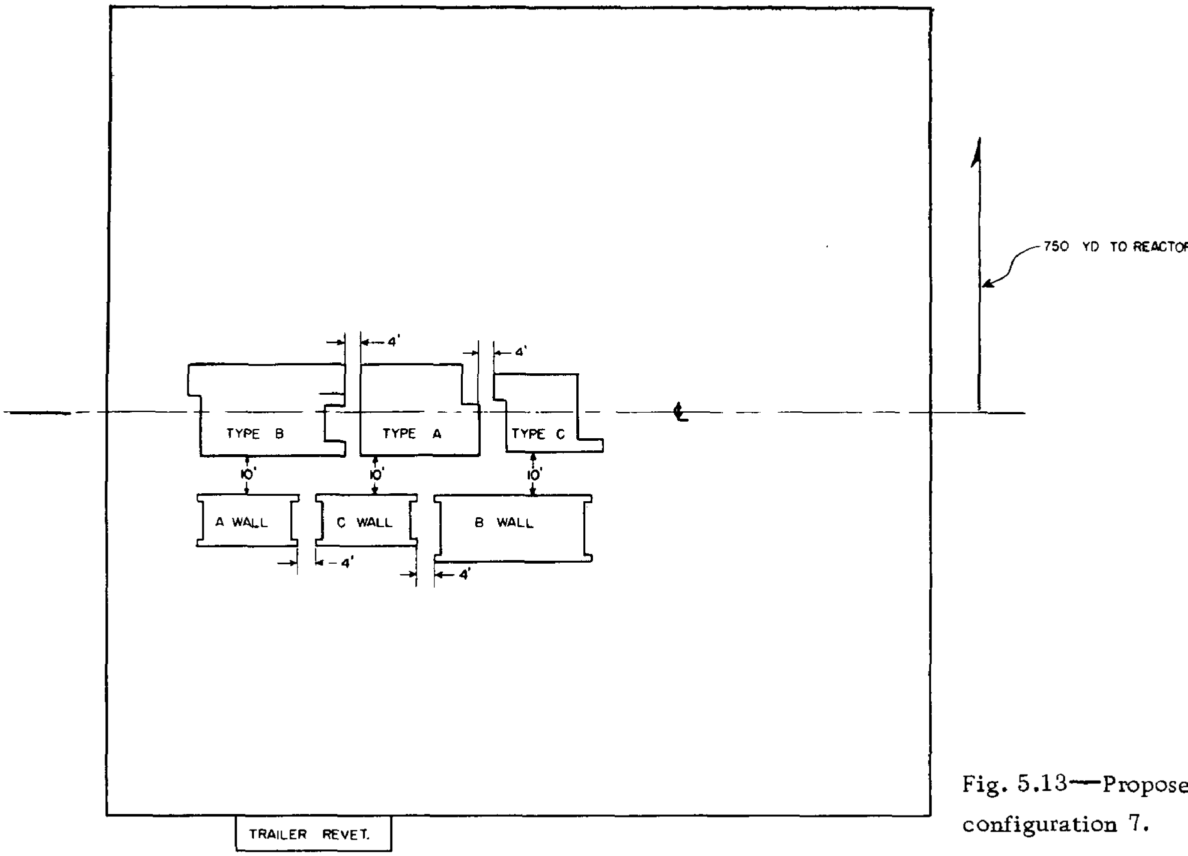

Fig. 5.13—Proposed house configuration 7.

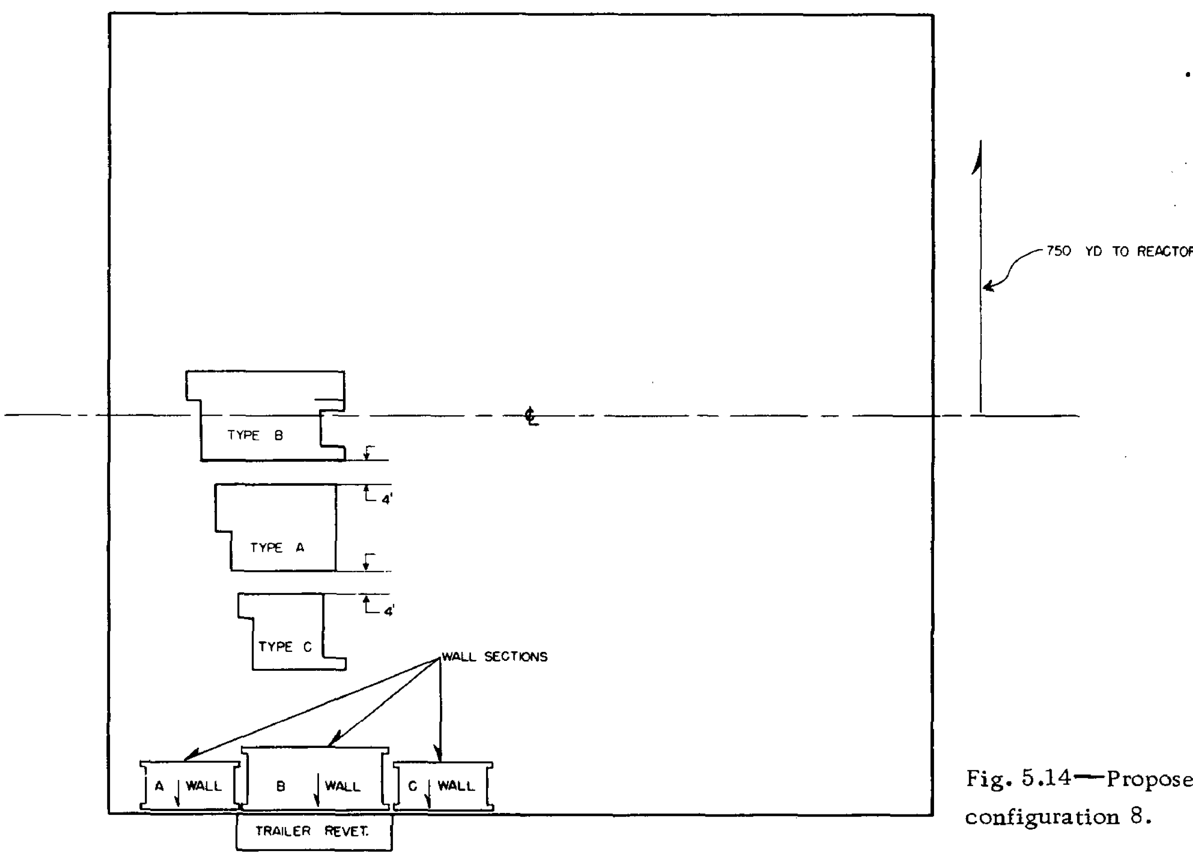

Fig. 5.14—Proposed house configuration 8.

A double-crystal proton-recoil "telescope" type spectrometer[7] and a magnetically analyzing Compton spectrometer[8] will be used to measure neutron and gamma-ray energies, respectively, during steady-power operation. A semiconductor-diode neutron spectrometer[9] and a silicon-diode fission threshold counter[10] will also be used to measure fast-neutron energy spectrum. A 3- by 3-in. NaI-crystal scintillation spectrometer will be placed in a collimator, shielded during a burst, uncovered as soon as the neutron population decreases, and operated to measure the spectrum of gamma rays from short-lived fission products as a function of time after burst. It will also be used to measure the spectrum of gamma rays from the Co^{60} source.

A long counter will be used to normalize all measurements of radiation from the reactor to "standard" reactor power.

5.2.3 Project 1.3: Neutron Dose Measurements in Houses

The operating procedures of Project 1.3 are included in Sec. 5.2.1.

5.2.4 Project 1.4: Energy and Dose Measurements as Functions of Distance and Height

Project 1.4 is designed to determine the effects of the air–ground interface on the distribution of gamma-ray and neutron dose and of neutron energy by means of the following experiments.

1. Measurement of gamma-ray dose as a function of horizontal distance from the reactor tower and distance above or below the air–ground interface.
2. Measurement of neutron dose as a function of horizontal distance from the reactor tower and distance above or below the air–ground interface.
3. Measurement of neutron spectrum as a function of horizontal distance from the reactor tower and distance above or below the air–ground interface.

The measurements above ground are to be made in a vertical plane extending horizontally from the base of the reactor tower eastward for 4500 ft and vertically from ground level to 500 ft above the ground. Detectors for these measurements are to be positioned in the plane by means of helium-filled balloons. The instrument package will be fastened to the balloon center guy and will be lifted by the balloon.

The experimental data will be normalized to "standard" reactor power by using the long counter in the experimental area, which will be operated continuously.

Measurements of neutron spectrum will be made in the same plane but at fewer positions.

The Radsan[3–5] will be used for measuring fast-neutron dose, the Phil[6] for gamma dose, and silicon-diode fission threshold counters for fast-neutron spectrum.[10]

Underground measurements will be made by putting the instruments in a protective cover and placing them underground to depths of about 1 ft.

Balloon Safety (Prepared by J. H. Knight). The balloons must be prevented from breaking tie-downs and floating off into the airways. This will be prevented as follows: The balloons will be fully inflated at the 4241-ft elevation. This will ensure, in the unlikely event that one does break free from its moorings, that it will rise at approximately 1700 ft/min until it reaches 10,000 ft above mean sea level, at which time it will burst. This rapid rate of ascent will assure that the balloon will fall within the test site air-control area before winds could blow the balloon outside the area.

5.3 PROGRAMS 2 TO 7

Detailed information for Operation BREN Programs 2 to 7 will be published in the reports for the various programs.

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Chapter 6

SAFETY

6.1 HAZARDS DUE TO OPERATION OF THE ORNL HEALTH PHYSICS RESEARCH REACTOR

Several types of abnormalities could occur during routine operations. Such accidents can generally be placed into one of two major categories. The first type of accident could result from a failure of the safety mechanisms or controls to scram the reactor, and the second, from the addition of excess reactivity at an inopportune moment, which could result in excessive yields.

6.1.1 Hazards During Steady-state Mode

During operation in the steady-state mode, the reactor assembly is first brought to delayed critical in the presence of a source. The regulating rod is then inserted until it reaches a position at which a positive period is obtained.

Reactivity can be added manually or automatically to maintain the desired steady-power level.

Overinsertion of the regulating rod can result in the inadvertent raising of the reactor power level at a rate equivalent to the reactivity addition of approximately 25 cents/min to a maximum of approximately 80 cents above delayed critical.

In the event that all the safety channels (period, level safety, and temperature) should fail, the thermal expansion of the core would limit the average core temperature to a value that would cause no damage to the core. However, in the event that, in addition to failure of the safety channels, reactivity were added with the mass-adjustment rod or by movement of a reflector, the resulting temperature rise could cause damage to the core; this combination of events, however, is considered unlikely.

During steady-state operation inadvertent addition of a reflector will not result in an excessive safety hazard because the reactor would be shut down by the period, level safety, or temperature scram systems.

Reactor operation will result in little radiation hazard to personnel operating the control console, which will be installed in an underground bunker protected by 4 ft of concrete and 7 ft of compacted earth capped with a layer of asphalt. A dose-attenuation factor of more than 100,000 is conservative for such a structure. Operation of the reactor at its lowest elevation at 10 kw would result in a dose rate of less than 0.1 millirad/hr.

6.1.2 Hazards During Burst Mode

(a) *Control Failure.* Failure of the control mechanisms to scram the reactor after conclusion of a burst would leave the reactor above delayed critical. After a burst of 1×10^{17} fissions, the reactor would attain an average temperature[1] of approximately 1080°C within approximately 13 sec. The temperature would then slowly decrease to an equilibrium temperature of approximately 544°C after a long period of time. Local maximum temperatures in excess of the alloy melting point of approximately 1180°C would probably occur.

If such an accident should occur in the open after extensive operation,* the core could contain fission products equivalent to approximately 5.52 curies of I^{131} (see Ref. 2, which is included in this report as Appendix A). In the postulated meltdown the fission products actually released would be a fraction of those present and volatile at the attained temperature.[3] It has been demonstrated repeatedly that a person's internal exposure from inhalation of iodine isotopes accumulated during the period of extensive reactor operation[4] is more severe than his external exposure to the xenon and krypton or even to the total fission products (if their release is assumed).

The total equivalent I^{131} present at meltdown (about 5.52 curies) thus represents an excess of the maximum inventory available for release in a meltdown case. However, if this total were to be released, the estimated resultant intake by a person at a distance of 1000 m from the reactor would be, at 3.6×10^{-2} μc inhaled per curie released, 0.2 μc (see Ref. 5, which is included in this report as Appendix B). The total resultant internal exposure to the person's thyroid, 295 mrem, is a pessimistic estimate and should be reduced, both because total melting of the reactor core is improbable and because total release of the iodine inventory cannot occur.

(b) *Excessive Yield.* It is possible to calculate a theoretical yield for the step or ramp addition of reactivity at various points in the pulse-operation procedure. The hazardous part of the cycle is during the time interval between delayed-critical operation and the transition to the burst mode. An external source is not available during this interval.

The energy release from an accident involving the (ramp) addition of a large amount of excess reactivity, in the absence of an external source of neutrons, would depend upon how far the reactivity insertion had progressed up to the time when the first persistent fission chain was initiated.

The "wait-time"[6] before a persistent chain is initiated is considered the potentially most hazardous interval. This wait-time varies because of the statistical nature of the fission and neutron transport processes in the assembly.

It can be shown that, if the excess reactivity were added before the insertion of the safety block, the relatively slow movement of the safety block would result in an average accident of a slow rise in a delayed-critical type of ramp rather than a sharp pulse.[6]

It can be shown that, if excess reactivity were added as a ramp or step addition after insertion of the safety block, before, during, or after insertion of the burst rod, theoretical yields in excess of 10^{18} fissions could be attained,[6] depending on the amount of reactivity added. If an increase of 2 dollars in reactivity should occur, an excursion of approximately 10^{20} fissions could result on the firing of the burst rod.

It is important to note that this hypothetical excursion would require the undetected addition of reactivity equivalent to the moving of a 10 by 10 by 1 in. sheet of plexiglas from 2 in. away from the assembly to ½ in. away.

A steel tube that is physically attached to the bottom of the core surrounds the volume into which the safety block is ejected. This system would effectively prevent the addition of excess reactivity due to a fall of the reactor to the ground [see Sec. 3.2.1(a)].

The "maximum credible" accident as estimated by Nuclear Development Corporation of America (NDA)[1] is one in which the assembly falls onto concrete, with all the fuel instantly assembling into the configuration, giving maximum reactivity. NDA estimated that, under these conditions, approximately 65 cents of reactivity above prompt critical would be added, yielding a burst of approximately 1.2×10^{19} fissions.

It is of interest for purposes of comparison with weapons detonations that, with the use of a conversion of approximately 1.4×10^{23} fissions = 1 kt, a burst of 1.2×10^{19} fissions would release an energy equivalent to approximately 200 lb of high explosive.

*For this calculation it was assumed that the reactor operated at a level of 1 kw for 2 hr each day for an infinite number of days before the accident.

Although the probability of occurrence is extremely low, it is useful to evaluate the effects of excessive burst yields. Previous studies[7] have shown that a burst of 10^{19} fissions representing approximately 312 Mw-sec of energy would presumably volatilize at least part of the reactor core.

The fission-product inventory formed was shown to be equivalent to 70.06 curies of I^{131} at 1 min after the burst and 95.71 curies at 1.12 hr after the burst.[2] The inventory is not greater than 95 curies at any time subsequent to 1.12 hr. All estimates of downwind exposures and of the thyroid doses are based on the greater of the preceding values.

An adult at 1000 m from the reactor could have an inhalation intake of 3.5 μc I^{131} equivalent, which would result in a thyroid exposure of 5.19 rem. This exposure does not exceed the permissible quarterly adult-ingestion intake.[5]

It should be noted that the estimated bone exposure due to strontium isotopes is less than the thyroid exposure from iodine isotopes[2] by a factor of over 20.

Estimates made of external exposure due to submersion in the cloud of fission products created during an accident[5] indicate that, if attrition by deposition and condensation in transit is neglected and if a wind speed of 5 m/sec is assumed, the earliest arrival of the airborne released material at a distance of 1000 m is 200 sec, which would lead to an external exposure not greater than 15 rem. The combined prompt neutron and gamma-ray dose that could be received by personnel in the control bunker should such a burst of 10^{19} fissions occur at ground level at the base of the tower would be less than 2 millirads.

There are several factors that tend to reduce the hazard of the maximum credible accident. The reactor will be rigidly fastened into the hoist car before fueling, and therefore the likelihood of the reactor's falling out of the car is very remote. The 18-ft-high car moves on rails in a "shaft" enclosed by horizontal angle beams at 10-ft intervals; it is therefore unlikely that the car could leave the shaft. In addition to safety devices that are a part of the hoist system, the reactor control cables and pulleys would have a cushioning effect so that, if the hoist car and reactor should plunge down the shaft, they would be decelerated before reaching the ground. Since the critical assembly is suspended from above, it would be further slowed as the bottom of the car collapsed. Lowering the rate of addition of reactivity tends to reduce the magnitude of a burst.

On the basis of the mechanical safety measures built into the design and the administrative controls available, it can be stated that the possibility of an incident that would result in hazards to the population is extremely small.

6.1.3 Other Hazards

(a) *Flood.* A severe flood at the location of the test site is extremely unlikely, and the location of the reactor on the tower at a minimum possible height of 25 ft further precludes flooding. When not on the tower, the core will be disassembled to a safe configuration.

(b) *Weather.* The tower has been designed to withstand winds of greater than 110 mph.

(c) *Fire.* All structural and accessory materials are noncombustible, with the exception of the electronics insulating materials. The electronics are located in a separate compartment and are appropriately fused.

(d) *Sabotage.* It is conceivable that a clever saboteur could circumvent the security measures contemplated for the site; therefore a 24-hr guard surveillance will be required. If a saboteur were thoroughly familiar with the control system, with its detailed layout and functions, he could, by passing the interlocks, cause a combination of operations that would seriously disrupt the utility of the reactor assembly. It is, however, exceedingly unlikely that he could cause an excessive burst that would endanger lives other than his own.

6.2 HAZARDS DUE TO OPERATION OF THE 1200-CURIE Co^{60} GAMMA-RAY SOURCE

The gamma facility, to be used subsequent to operation of the HPRR, is designed to be operated so that there will be no significant radiation hazard to personnel on-site or off-site.

6.2.1 General Description

The lifting device and storage shield are described in Sec. 3.3. All Co^{60} is doubly encapsulated in stainless steel 347. The primary container, a hollow cylinder with 16 evenly spaced holes around its rim, is sealed by a helium-shielded arc weld. The secondary container is a sturdy stainless-steel helium-shielded arc-welded capsule (see Fig. 3.19).

The construction of the source capsule is such that it would survive, without rupturing, a drop of 1500 ft to the desert floor. The capsule will also absorb, without damage, all shock incurred during shipment.

6.2.2 Source Suspension

A small hole ($^5/_{32}$ in.) is drilled through each end of the solid tip of the capsule. This allows the capsule to be attached by a steel cable to the bottom of the source shield plug.

As explained in Sec. 3.3, the source is suspended approximately 3 ft above the top of the shield by mechanical lifting of the shield plug. The source capsule is subject to sway, but the guide rods will hold the plug sufficiently in place to allow the capsule and plug to return to their seats in the shield without delay. The opening in the shield is in narrow step reductions so that it offers a funnel for the source. The doors of the hoist car prevent entrance to, and blockage of, the shield by extraneous objects, such as birds.

6.2.3 Radiation Hazards

The fail-safe characteristics of the source facility are discussed in Sec. 3.3.1.

When the source is seated in its shield, the external dose rate at 1 ft from the center of the shield (nearest possible point of measurement) is approximately 60 millirads/hr, and the dose rate at 1 meter is approximately 10 millirads/hr.

During exposure of the source, unprotected personnel will not be closer than 325 yd to the tower without authorization by the Rad-Safe Officer. The dose rate at this distance would be approximately 30 millirads/hr if the source were exposed at the lowest possible elevation.

Personnel inside the control bunker will not be subjected to a significant amount of radiation during any phase of the exposure.

The source facility will be operated at 500-, 1125-, or 1500-ft elevation most of the time. The dose rates corresponding to these distances are 86, 13, and 3 millirads/hr, respectively.

Dose rates at the base of the tower will be lower because of shielding by the source shield and the tower structure. When the source is operated in its lowest position, the dose rate in the control bunker will be less than 0.05 millirad/hr.

Surface contamination of the shield will be monitored on a routine basis with the use of smear techniques.

6.3 RADIATION-SAFETY PLAN

6.3.1 Monitoring

The radiation monitoring program for Operation BREN will be executed under the direction of the Rad-Safe Officer. The program will consist in routine and operational monitoring.

Routine monitoring will include the following procedure.

1. Surface-contamination monitoring on the source hoist car and in the bunker control house.
2. Monitoring of external radiation hazards.
3. Personnel monitoring.

Operational activities that will require radiation monitoring are as follows:

1. Installation of the HPRR on the tower.
2. Maintenance operations on the HPRR.
3. Removal of the HPRR from the source hoist car.
4. Installation, maintenance, and removal of the Co^{60} facility on the source hoist car.
5. Radiation emergencies.

The following Rad-Safe instrumentation is to be used in routine and operational monitoring.

1. Cutie Pie (measures beta–gamma dose rates of 0 to 10 rads/hr).
2. G-M survey meter (measures low-level beta–gamma dose rates, beta–gamma contamination on work surfaces, floor, personnel, clothing, and equipment).
3. Portable alpha scintillation counter (measures alpha contamination on work surfaces, floor, personnel, clothing, and equipment).
4. Fast-neutron survey meter (measures fast-neutron dose rate from delayed neutrons, from spontaneous fission, and from polonium–beryllium neutron source during operations that require personnel to be on the source hoist car).
5. Thermal-neutron survey meter (measures thermal-neutron dose rate from the sources listed in item 4).
6. (a) Personnel badge dosimeter (measures exposure to beta, gamma, and neutron radiation over extended length of time).
   (b) Orientation belt (for those near the reactor).
7. Pocket dosimeter (measures gamma dose; results are tabulated either weekly or daily).
8. Air sampler (to take air samples periodically in the bunker and routinely during maintenance operations on the HPRR).
9. Long counter (to be used in experimental area for neutron background and data normalization).
10. Scintillation detector (for gamma background in bunker).

For an effective radiation monitoring program, there must be established operational levels of exposure hazard, both external and internal, which should not be exceeded. These contamination levels to be observed in Operation BREN are given in Table 6.1.

Periodic samples will be taken of the air inside the control bunker. The maximum permissible concentrations in air (MPC_a) for a 40-hr work week for U^{235} and Sr^{90} as given in Na-

Table 6.1 — MAXIMUM OPERATIONAL LEVELS OF RADIATION FOR OPERATION BREN

Location of radiation	Type of radiation	Direct reading on surface	Amount transferrable (smear)
		Work Areas and Equipment	
Source hoist car	Alpha	2000 dis/min/100 cm^2	400 dis/min/100 cm^2
	Beta–gamma	Not detectable owing to high dose rate from fission products in reactor core	1.0 millirad/hr
Control bunker	Alpha	300 dis/min/100 cm^2	30 dis/min/100 cm^2
	Beta–gamma	0.25 millirad/hr	0.25 millirad/hr
Maintenance equipment	Alpha	300 dis/min/100 cm^2	30 dis/min/100 cm^2
	Beta–gamma	0.25 millirad/hr	0.25 millirad/hr
		Personal Clothing and Shoes	
Clothing	Alpha	300 dis/min/100 cm^2	
	Beta–gamma	0.25 millirad/hr	
Shoes	Alpha	300 dis/min/100 cm^2	
	Beta–gamma	0.6 millirad/hr	
		Skin Surfaces	
General body	Alpha	150 dis/min/100 cm^2	
	Beta–gamma	<0.1 millrad/hr	
Hands and feet	Alpha	150 dis/min/100 cm^2	
	Beta–gamma	<0.3 millirad/hr	

tional Bureau of Standards Handbook 69 will apply for this area (Table 6.2). If the air activity exceeds the values given in Table 6.2, operations will be suspended until appropriate measures can be taken to control the air supply.

Routine surveys will be made in each of the program areas once each month. Additional surveys will be made on request by the Program Directors or at the discretion of the Rad-Safe Officer.

Table 6.2 — MPC_a INSIDE CONTROL BUNKER (40-HR WEEK)

Type of radiation	MPC_a, $\mu c/cm^3$
Alpha	5×10^{-10}
Beta	5×10^{-10}

Table 6.3 — PROBABLE GAMMA-RAY DOSE RATES AT 5 FT FROM CENTER OF CORE[8]

Cooling time, hr	Gamma radiation, rem/hr	Maximum working time permitted,* min
	Operation at 1 Kw for 5 Hr	
0.5	5.0	3.6
6.0	0.90	20
12.0	0.50	36
	Operation at 1 Kw for 10 Hr	
0.5	6.50	2.8
6.0	1.50	12
12	0.85	21
	Burst Operation of 10^{17} Fissions	
0.5	30.0	0.6
4.0	2.0	9

*The time required for a person working at a distance of 5 ft from the core to receive the maximum permissible dose for a single exposure (300 mrem).

The exposure limit for Operation BREN will be 3 rem per quarter. The weekly accumulated dose shall not exceed 300 mrem. During all major or minor maintenance operations on the HPRR, the maximum single exposure to any person will be 300 mrem (except in case of emergency). However, in no case will the accumulated exposure at NTS combined with previous occupational exposure exceed the maximum permissible dose (in roentgens equivalent man) of $5(N - 18)$ for any person, where N is the age of the person exposed.

The probable gamma-ray dose rates at 5 ft from the center of the core caused by fission-product buildup during operation of the HPRR are given in Table 6.3. It is evident from the data in Table 6.3 that shielding of the core during maintenance operations may prove to be a necessary precaution. In any event, carefully considered plans will be executed to reduce the chance of overexposure.

Additional provisions of the routine operating procedure are as follows:

1. Each user group participating in Operation BREN will use NTS personnel monitoring as required in NTSO-SOP, Chapter 0524.
2. ORNL employees will also wear the ORNL standard film badge, with an additional fast-neutron film packet, to keep ORNL personnel records up-to-date.

3. Groups other than those from ORNL will comply with the requirements of NTS in addition to those of the respective parent organizations.
4. Bioassay techniques will be employed in determining the body background of personnel prior to arrival at NTS.
5. All parent organizations of personnel participating in Operation BREN are required to submit records showing the amount of radiation their personnel have received through occupational exposure.
6. Each Program Director will receive information describing the radiation hazard in his area at all times when the reactor or gamma source is in use.
7. Program Directors will register with the Rad-Safe Officer all radioactive materials used in relation to their programs.

6.3.2 Emergency Procedure

As previously discussed, a conservative estimate of the maximum credible accident is a burst of not more than 10^{20} fissions at ground level. It has been estimated that the dose due to prompt gamma and neutron radiation from a maximum credible accident would be approximately 1.3 rads and less than 0.02 rad at the 750-yd station and inside the control bunker, respectively.

Nuclear-accident dosimetry will be provided by complete threshold-detector units located on top of the bunker and on top of Program 1 mobile laboratory. A secondary station will be located on the source hoist car and will be changed at least every 10 kw-hr of HPRR operation. Counting facilities for threshold-detector-unit foils and blood Na^{24} will be located in the Program 1 support trailer.

A tentative emergency procedure that will ensure safety to personnel of Operation BREN is necessary. The action that would be taken if there were evidence of an accident due to operation of the HPRR is outlined as follows:

1. A signal would be given by the Radiation Source Officer to evacuate the experimental area. Personnel would take two potassium iodide tablets, put on face masks with charcoal-loaded canisters, and move to the B–J–Y.
2. Proper reports would be channeled to OFO-NTS through the Technical Director and CETO.
3. Under the direction of the Radiation Source Officer, personnel in the 4-300 bunker would immediately seal off the entrance to the control room, take two potassium iodide tablets, and put on face masks. They would remain in the bunker until the order for leaving had been issued by the Technical Director.
4. Individual doses would be estimated from results of processing badge dosimeters, reading pocket dosimeters, and analyzing accident dosimeters, including Na^{24} in the blood.
5. Cleanup of contamination resulting from a reactor accident would follow normal NTS procedures. Exposure provisions of National Bureau of Standards Handbooks 59 and 69 would apply. Emergency services of NTS support contractor would be requested.

REFERENCES

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3. G. E. Creek, W. J. Martin, and G. W. Parker, *Experiments on the Release of Fission Products from Molten Reactor Fuels*, USAEC Report ORNL-2616, Oak Ridge National Laboratory, July 22, 1959.
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8. G. Breidenbach, D. Burgreen, E. Gahan, B. Minushkin, L. Rutland, H. Soodak, and A. Stathoplos, *Preliminary Design of the ORNL Fast Burst Reactor,* USAEC Report NDA-2136-1, Nuclear Development Corp. of America, July 30, 1960.

Chapter 7

SCHEDULE

7.1 TENTATIVE OPERATIONAL SCHEDULE

D-Day is the date the 1500-ft tower is completed and turned over to Operation BREN. The following operational schedule is based on this date.

D − 2 weeks	Reactor, trailers, and key personnel of Program 1 arrive at NTS. Commence control-room installation (subject to agreement with construction contractor), experimental instrument installation, and check-out.
D-Day	Commence installation and check-out of reactor.
D + 2 weeks	Commence experiments.
D + 10 weeks	Finish measurements of mixed radiation fields. Remove reactor and substitute Co^{60} source.
D + 14 weeks	Complete measurements of gamma-ray field from Co^{60} source. Complete interim reports. Leave NTS.

7.2 WORK HOURS

The hours of operation will be varied according to the needs of Program 1 experiments, construction schedules, required maintenance, and weather conditions. It is assumed that some periods of operation will be at night, on week ends, or on holidays. It is expected that during Operation BREN an extended work week will be in effect for project personnel only. The longest period of operation will probably be 12 hr during any 24 hr, for not more than two consecutive days. It is probable that five consecutive days of shorter operating periods will be a maximum. Time requirements for sufficient processing of data to maintain adequate program planning precludes longer continual operation.

Chapter 8

REPORTS

8.1 INITIAL REPORTS

To facilitate planning of the support requirements and of the technical program of Operation BREN, all Program Directors of approved or tentative programs are required to submit initial status reports, using the normal CETO form, to the Director, CETO, or to the Technical Director, Operation BREN.

The Rad-Safe Officer, Operation BREN, will request specific information concerning the radiation history of personnel of all programs before the arrival of personnel at NTS.

8.2 PROGRESS REPORTS

Progress reports will be submitted to the Director, CETO, through the Technical Director, Operation BREN, on a flexible schedule to be established in the field. The purpose of these reports is to apprise the Director of the progress and needs of the programs in regard to administrative, support, and technical matters.

8.3 INTERIM REPORTS

Each Program Director will submit to the Director, CETO, through the Technical Director, Operation BREN, an interim report by project which will describe the experiments performed. This report will include objectives, apparatus, techniques, data, and preliminary conclusions. A draft of each report must be submitted before departure from NTS at the conclusion of the operation.

8.4 FINAL REPORTS

Each Program Director will submit a final report by project to the Director, CETO, through the Technical Director, Operation BREN, for publication as a CEX report. A draft of each report must be submitted within six months of the conclusion of Operation BREN.

Appendix A

X-847

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

July 18, 1960

To: Fast Burst Reactor File

Re: FBR Hazards–Estimates of Fission Product Inventories and Exposures

As part of a preliminary study of FBR hazards, calculations have been made to estimate the probable maximum amounts of fission products present under certain stipulated conditions. Particular attention has been given to those isotopes of special interest in the calculation of internal exposures following inhalation of air-borne fission products and of submersion exposures.

For the case of a postulated meltdown of part of the reactor core, the modes of operation of the reactor were examined to determine which would represent the most severe, hence controlling, hazard. These modes of operation are tabulated in terms of power level and frequency to give a comparable basis for choice:

(1) 2 hr/da for 5 da/wk at 1 kw = 10 kwh/wk

(2) 6 hr/da for 5 da/wk at 5 watts = 150 whr/wk

(3) 2 bursts/da for 5 da/wk at 10^{17} fiss/burst = 8.7 kwh/wk

The conversion in case (3) above is based on 3.2×10^{10} fiss/sec = 1 watt[(1)] or 0.87 kilowatt hour per burst of 10^{17} fissions. The foregoing shows that the worst time for a meltdown to occur would be at the end of a 2-hour run at 1 kilowatt after a long sequence of previous daily runs of like operation.

In a postulated meltdown the fission products released would be a fraction of those present and volatile at the attained temperature. Numerous experiments[(2)] have shown the most important of these to be isotopes of iodine and of the noble gases Xe and Kr. For the case where no selective means are employed to reduce the concentrations of I isotopes, as in the case here if a meltdown were to occur out-of-doors, the internal exposure from the inhalation of iodine isotopes accumulated in reactor operation has repeatedly been shown[(3)] to be much more severe than the external exposure to the Xe and Kr, or (even) to the total fission products if their release is assumed. For this reason,

(1) From ORNL-2127, Part 1, Vol. 1, "U-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time," by J. O. Blomeke and M. F. Todd, Aug. 1957.

(2) In particular, see ORNL-2616, "Experiments on the Release of Fission Products from Molten Reactor Fuels," G. E. Creek, W. J. Martin, and G. W. Parker, Dec. 1958.

(3) For example, in "Reactors, Hazard vs Power Level," by T. J. Burnett, Aug. 1956, Nucl. Sci. and Engr., Vol. 2, No. 3, May 1957, pp. 382-393.

attention has been confined herein to the calculation of the maximum inventories of iodine isotopes present under the most severe of the normal operations cited.

For simplicity of calculation, a more restrictive case was assumed, i.e., operation for 7 da/wk. Here the reactor is considered to have been operated for 2 hours and shut down 22 hours, with this daily cycle having been repeated for approximately infinite time (as far as I isotope content is concerned). In the case where the I is of substantially longer half life than its Te precourser, (as with I^{131}) it is simpler and gives a larger I inventory to assume as present the fraction of I which would be held up. After continuous operation for time t the activity built up will be $A_s (1 - e^{-\lambda t})$, where A_s is the saturation activity and λ the decay constant for the isotope. If shutdown then occurs, the activity present at time T thereafter is $A(t,T) = A_s (1 - e^{-\lambda t}) e^{-\lambda T}$. With repeated operating cycles as in the assumptions stated above, the total present will be:

$A_s (1 - e^{-\lambda 2}) = A(2)$ for the previous 2 hr,

plus: $A(2)\, e^{-\lambda 24}$ for the 1st preceding 24 hr period,

plus: $A(2)\, (e^{-\lambda 24})^2$ for the 2nd preceding 24 hr period,

plus . . . $A(2)\, (e^{-\lambda 24})^n$ for the nth preceding 24 hr period.

This may be written as:

$$\Sigma\, A(2,24) = A(2)\, (1 + e^{-\lambda 24} + (e^{-\lambda 24})^2 + \ldots + (e^{-\lambda 24})^n + \ldots) \tag{1}$$

which is a geometric progression with $e^{-\lambda 24}$ (<1) as the common ratio, hence:

$$\Sigma\, A(2,24) = \frac{A_s (1 - e^{-\lambda 2})}{(1 - e^{-\lambda 24})} \tag{2}$$

At a power level of 1 kilowatt, the number of atoms of an isotope produced per second is:

$$N_i = 3.2 \times 10^{13} \text{ (fiss/sec/kw)} \times Y_i \text{ (chain yield)} \tag{3}$$

At equilibrium, the saturated activity is N_i dis/sec, hence:

$$A_{si} = \frac{3.2 \times 10^{13}\ Y_i}{3.7 \times 10^{10}}\ \text{curies} \tag{4}$$

In the case where the I has a shorter (or nearly the same) half life than its Te parent, as with I^{132} (and I^{134}), the daughter growth from its parent must be considered. After operation for time t at a production rate of N_1^0 atoms of parent per unit time, the accumulated activity of the parent (denoted by subscript 1) and of the daughter (subscript 2) are:

$$\Sigma\ A_1(t) = N_1^0\ (1 - e^{-\lambda_1 t}) \tag{5}$$

$$\Sigma\ A_2(t) = \frac{N_1^0\ \lambda_1\lambda_2}{\lambda_2 - \lambda_1} \left\{ \frac{1 - e^{-\lambda_1 t}}{\lambda_1} - \frac{1 - e^{-\lambda_2 t}}{\lambda_2} \right\} \tag{6}$$

The remaining daughter activity after shutdown for time T following operation for time t is the sum of two components, that growing from the parent present at shutdown:

$$\Sigma\ A_2(p) = \frac{N_1^0\ \lambda_2\ (1 - e^{-\lambda_1 t})}{(\lambda_2 - \lambda_1)}\ (e^{-\lambda_1 T} - e^{-\lambda_2 T}) \tag{7}$$

and that remaining from the daughter present at shutdown:

$$\Sigma\ A_2(d) = \frac{N_1^0\ \lambda_1\lambda_2}{(\lambda_2 - \lambda_1)} \left\{ \frac{1 - e^{-\lambda_1 t}}{\lambda_1} - \frac{1 - e^{-\lambda_2 t}}{\lambda_2} \right\} e^{-\lambda_2 T} \tag{8}$$

After repeated cycles of operation for 2 hr (t) each 24 hr (T) the total daughter activity present in curies is:

$$\Sigma\ A_2(t,T) = \frac{N_1^0}{3.7 \times 10^{10}} \left\{ \frac{\lambda_2}{(\lambda_2 - \lambda_1)} \cdot (1 - e^{-\lambda_1 2}) \left[\frac{1}{(1 - e^{-\lambda_1 24})} - \frac{1}{(1 - e^{-\lambda_2 24})} \right] + \left[\frac{\lambda_2}{(\lambda_2 - \lambda_1)}\ (1 - e^{-\lambda_1 2}) - \frac{\lambda_1}{(\lambda_2 - \lambda_1)}\ (1 - e^{-\lambda_2 2}) \right] \frac{1}{(1 - e^{-\lambda_2 24})} \right\} \tag{9}$$

For convenience in calculation, it may be pointed out that:

$$\frac{\lambda_2}{\lambda_2 - \lambda_1} = \frac{t_1}{t_1 - t_2} \qquad \text{and} \qquad \frac{\lambda_1}{\lambda_2 - \lambda_1} = \frac{t_2}{t_1 - t_2} \tag{10}$$

in which t_1 and t_2 are the respective half lives of the parent and daughter, in this case: Te and I. The total curies present of I^{131}, I^{133}, and I^{135} have been calculated using equation (2) and for I^{132} and I^{134} equation (9) was used. The results, together with pertinent constants, are tabulated in Table II to follow.

In order to sum their effects, it is convenient to express the inhalation intake of other iodine isotopes in terms of the amount of I^{131} to which they are equivalent. The basis here used is the comparison of total internal exposure (D) to the thyroid subsequent to a single inhalation intake. For I^{131} the total single intake exposure may be calculated using the formula(3):

$$D = 73.8 \text{ fa I } \Sigma\ E_i(\text{RBE})\ N\ T_i/m \tag{11}$$

Evaluating this expression for I^{131} with the constants from the 1959 revision of ICRP(4) where fa = 0.23, Σ E(RBE)N = 0.23 mev, T = 7.6 da, and m = 20 g., we get, with an intake, I, of 1 μc:

$$D_{I^{131}} = 1.484, \text{ or } 1.5 \text{ rem per inhaled } \mu c\ I^{131}$$

For other iodine isotopes the differences lie only in the values for $\Sigma\ E_i$(RBE), the mev/dis dissipated in the thyroid, and for T_i, the effective half life in the thyroid, hence:

$$\frac{D_i}{D_{I^{131}}} = \frac{\Sigma\ E_i \cdot T_i}{\Sigma\ E_{I^{131}} \cdot T_{I^{131}}} \tag{12}$$

represents the equivalence of other iodine isotopes in terms of I^{131}. Table I, on the following page, gives the values used and the resultant equivalence factors.

(4) Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959) published as Vol. 3 Health Physics, June 1960. Pergamon Press.

TABLE I

EQUIVALENCE OF IODINE ISOTOPES TO I^{131}

Isotope	ΣE_i	T_i	$\Sigma E_i \cdot T_i$	$D_i/D_{I^{131}}$
I^{131}	0.23	7.6	1.748	1.000
I^{132}	0.65	0.097	0.063	0.036
I^{133}	0.54	0.87	0.470	0.269
I^{134}	0.82	0.036	0.030	0.017
I^{135}	0.52	0.28	0.146	0.083

The above factors have been applied to give the equivalent curies of I^{131} listed in the last column of Table II.

TABLE II

TOTAL EQUIVALENT I^{131} INVENTORY FOR MELTDOWN

Isotope	Half life	Yield %	A_s curies	Σ A(2,24) curies	Equiv. I^{131} curies
I^{131}	8.05 d	2.9	25.1	2.19	2.19
Te^{132}	77 h	4.4	38.1		
I^{132}	2.4 h	4.4		3.07	0.11
I^{133}	20.8 h	6.5	56.2	6.58	1.77
Te^{134}	44 m	7.6	65.8		
I^{134}	52.5 m	7.6		33.94	0.58
I^{135}	6.68 h	5.9	51.0	10.43	0.87
Total, as equivalent I^{131} curies					5.52

The above represents in excess of the maximum inventory of I^{131} equivalent available for release in the meltdown case, rather than the total actually released. If this total were to be released, the estimated resultant intake would, at 3.6×10^{-2} μc inhaled per curie released(5), amount to 0.2 μc with a total resultant internal exposure to the thyroid of 295 mrem.

(5) Letter, T. J. Burnett to FBR File, "Dilution and Inhalation Intake Estimates," July 13, 1960.

The above estimate of exposure should be reduced both because total melting of the reactor core is improbable and because total release of the iodine inventory cannot occur.

For the case of a postulated excessive burst the estimated maximum burst of 10^{19} fissions represents 312 megawatt-seconds or 3415 BTU of energy which it is presumed would volatilize at least part of the reactor core. The fission product inventory to be evaluated here is that formed in the burst. A preliminary estimate using data given in ADC-65[6] showed that considerably more I^{131} equivalent would be present than in the more probable meltdown case. More careful estimates were made using data given in USNRDL-456[7] which is a revision of the earlier ADC-65 report. Where two values were given, the larger estimate was used. The results for iodine isotopes are given in Table III:

TABLE III

TOTAL EQUIVALENT I^{131} INVENTORY FOR MAXIMUM BURST

	t = 1 min		t = 1.12 hr	
Isotope	Curies	Equiv. I^{131}	Curies	Equiv. I^{131}
I^{131}	.05	.05	4.38	4.38
I^{132}	43.3	1.56	38.1	1.37
I^{133}	37.0	9.96	102.5	27.57
I^{134}	1301	22.13	1730	29.41
I^{135}	438	36.36	397.3	32.98
Total		70.06		95.71

The above values are those present at the times stated after the burst and are not worse at any time subsequent to 1.12 hr. These times are considered to bracket any times of exposure interest downwind and it will be noted that the I inventory variation is not great as these isotopes grow from Te parents formed in the burst, then subsequently decay.

It would seem adequately conservative to base any estimate of thyroid dose on the greater of the above values. Such an estimated[5] intake of 3.5 μc I^{131} equivalent would

(6) "Simultaneous Slow Neutron Fission of U^{235} Atoms, I. Individual and Total Rates of Decay of the Fission Products," Apr. 1949, H. F. Hunter and N. E. Ballou.

(7) "Calculated Activities and Abundances of U^{235} Fission Products," Aug. 1956, R. C. Bolles and N. E. Ballou.

result in a total thyroid exposure of 5.19 rem which does not exceed the permissible quarterly adult occupational exposure. However, it should be pointed out that the ICRP has endorsed the "emergency" levels proposed by the British Medical Research Council[8] which stipulate the following maximum permissible daily ingestion intakes and maximum permissible total ingestion intakes for different ages:

TABLE IV

MAXIMUM PERMISSIBLE INGESTION INTAKES VS AGE

Age	Thyroid mass (g)	Daily intake μc	Total intake μc
Up to 6 mo	1.8	0.06	0.65
6 mo–3 yr	3.4	0.11	1.20
3 yr–10 yr	9.2	0.30	3.40
Adult	25	1.3	15.0

Although the above values are based on ingestion rather than inhalation intakes, and the breathing rate data is lacking on which to calculate comparable limits vs age for inhalation, these limits suggest a measure of concern that inhalation thyroid doses for exposed children might exceed the 25 rem limit on which these British values are based. In addition to the internal exposure to the thyroid from inhaled iodine, this organ would be subject to external exposure from the fission product cloud. For the burst case, such external exposure is of different relative magnitude (compared to internal exposure) than with reactor operation for extended periods.

Before considering the external exposure, it is of interest to examine the amounts of strontium isotopes present following an excessive burst. Because of varying delays in the growth from precoursers, the maximum values for the different Sr isotopes occur at different times[7]. In Table V these maxima are listed together with the times at which they occur or for which they persist. By basing the calculations on the maximal values of Sr isotopes, the results will be conservative and the determination of values for a number of different times can be avoided. Table V gives also the values of total bone exposure per inhaled μc of the different Sr isotopes as calculated using equation (11) and the constants from the latest ICRP report[4]. In addition the inhalation intakes have been estimated[5] in order to yield exposures from individual Sr isotopes for addition.

(8) "Maximum Permissible Dietary Contamination after the Accidental Release of Radioactive Material from a Nuclear Reactor," Apr. 1959, Brit. Med. Jour., pp. 967-969.

TABLE V

SR ISOTOPE INVENTORY FOR MAXIMUM BURST

Isotope	Times of maximum	Maximum curies	Exposure per μc (mrem)	Est. μc inhaled	Est. maximum exposure
Sr^{89}	3.52 hr	1.65	418	0.0594	24.8
Sr^{90}	31.2 m to 9.82 d	0.0152	44,550	5.47×10^{-4}	24.4
Sr^{91}	1.12 hr	265	7.44	9.54	71.0
Sr^{92}	2.15 m to 3.15 m	1103	2.59	39.7	102.8
Total (mrem)					223.0

The foregoing tabulation shows the bone exposure from Sr isotopes to be less than the thyroid exposure from I isotopes by a factor of over 20.

For estimates of the external exposure it is necessary to consider the inventory of total fission products at varying times subsequent to the postulated excessive burst. The formula used for estimating the total external exposure[3], assuming submersion in a cloud of large dimensions is:

$$Z = 0.26\ E_s \frac{X}{Q} \text{ rem/curie released} \qquad (13)$$

where X/Q is the number of curie sec/m^3 per curie released and E_s is the effective energy for submersion exposure in mev/dis. For evaluation of the above, the estimated value for X/Q in this case is taken to be 7.16×10^{-5} c. sec/m^3 per curie released[5], and E_s is arbitrarily assumed to be 1.5 mev/dis although it is recognized that this energy varies with time after burst. Instead of a term in equation (13) to account for the decay as a function of time, hence distance downwind, Table VI gives a number of times which are felt to include those of probable interest. Total fission products are calculated from data in USNRDL-456[7] and listed as the quantities of interest on the assumption that the excessive burst could result in total vaporization. Attrition by deposition and condensation, in transit to the point of exposure, although probably considerable, has been neglected.

TABLE VI

TOTAL FISSION PRODUCTS IN MAXIMUM BURST

Time (t) after burst	Curies at t	Submersion exposure (rad)
19 sec	7.44×10^6	207.7
1 min	1.70×10^6	47.5
3.15 min	5.16×10^5	14.4
9.92 min	1.69×10^5	4.72
31.2 min	6.27×10^4	1.75
1.12 hr	2.87×10^4	0.80

With the assumed wind speed, $\bar{u}$, at 5 m/sec, the earliest arrival of the air-borne released material at a distance of 1000 meters is 200 sec which would lead to an external exposure not greater than 15 rem. A smaller value for wind speed, present in the denominator of the equation for X/Q, would give a larger value for curie sec/m^3 exposure(5) but the additional delay in arrival time would offset this by the decay of activity in transit. It would not be unreasonable to anticipate more favorable estimates for X/Q by Gifford, et al., as a result of the observed behavior of released smoke at this site.

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Appendix B

X-847

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

July 13, 1960

To: Fast Burst Reactor File

Re: FBR Hazards—Dilution and Inhalation Intake Estimates

As part of a preliminary study of FBR hazards made for the purpose of presenting data to the Hazards Evaluation Branch (and, if necessary, to the Advisory Committee on Reactor Safeguards), calculations have been made to determine the dilution factors for use in estimating inhalation intake and submersion exposures potentially resulting from conceivable accidents to this reactor. The two circumstances possibly resulting in a release of airborne fission products are a meltdown of part of the core or an excessive burst partially vaporizing the core. Tests have shown the fuel alloy (90% U^{235}, 10% Mo) not to be pyrophoric.

The maximum elevation at which meltdown could occur is 1017 feet, or 50 feet above the concrete pad, in free air in the daytime. No operations outside the reactor building will be permitted except in daytime and when not raining. The excessive burst postulated (10^{19} fissions or 312 megawatt-seconds) can occur only after dropping the reactor onto the concrete pad surface, elevation 967′. The nearest point of unrestricted access is the north bank of the Clinch River, impounded behind the proposed Melton Hill Dam to an elevation of 800′, at a distance of 1000 meters. The nearest dwelling presently existing south of the Clinch River is at a distance of 1500 meters.

The meteorology of the site was discussed with F. A. Gifford and W. M. Culkowski of the U. S. Weather Bureau, Oak Ridge, Tennessee, on June 28, 1960, in a meeting with M. I. Lundin and T. J. Burnett. Generally pertinent data on the Copper Ridge and Melton Valley areas has been presented in ORO-99(1) and more specific data in ORNL-1550(2) for the TSF(3) and in ORO-196(4) for the EGCR. It was considered that the shielded hollow in the folds of Copper Ridge proposed as the site for this reactor would provide considerable shelter from winds aloft and that the relatively slow drift of airborne material down the winding creek bed would give considerable opportunity for impingement and

(1) "A Meteorological Survey of the Oak Ridge Area," May 1953, ORO-99.

(2) "The Tower Shielding Facility Safeguard Report," June 1953, ORNL-1550 (Del.).

(3) See also: "Climatology and Meteorology of the TSF Site," Nov. 1958, by W. M. Culkowski, U. S. Weather Bureau, Oak Ridge, Tennessee.

(4) "Experimental Gas-Cooled Reactor, Preliminary Hazards Summary Report," May 1959, ORO-196 with Supplement.

deposition on the foliage as well as entrapment and delay in the heavily wooded slopes. A series of direct observations are planned by the Weather Bureau Office personnel using smoke pots to provide more precise estimates of dilution. However, in view of the relatively small amounts (from preliminary calculations) of fission products probably released, it was recommended than an adequately conservative over estimate of resultant concentrations could be made using one of the Sutton diffusion equations[5] (simplified by ignoring the exponential term):

$$\frac{X}{Q} = \frac{2}{\pi \bar{u} C^2 x^{2-n}}$$

"Typical" values suggested by Gifford were: $n = 0.25$, $C^2 = 10^{-2}$, and $\bar{u} = 5$ meters/second. For a distance $x = 1000$ meters, these values give:

$$\frac{X}{Q} = 7.2 \times 10^{-5} \text{ curie sec/m}^3 \text{ per curie released}$$

If a conservative value of 30 liters/min. (5×10^{-4} m^3/sec.) is taken for the breathing rate assuming a condition of exertion, the resultant inhalation intake is 3.6×10^{-2} μc per curie released.

No credit is taken for the difference in elevation of 160 to 210 feet between the point of release and the nearest point of unrestricted access. The concentration at more distant points would be correspondingly less. No credit is taken either for deposition on foliage with consequent reduction of concentration at the river's edge. Rainout is not evaluated because of the operational restriction against use of the reactor outdoors in rainy weather. Cleanup is provided for the reactor building ventilation, hence the case of an enclosed accident is not evaluated on the basis that it would be much less severe than those considered. Local deposition is not estimated because this is a controlled area. Protection of reactor operating staff against external air contamination will be provided by filtration of intake air, shutdown of control building ventilation, or immediate departure of personnel during the interval required for transit of contaminated air from the shielded reactor valley to the control building.

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(5) "Meteorology and Atomic Energy," July 1955, U. S. Department of Commerce, Weather Bureau. Chapter 4, "An Outline of Atmospheric Diffusion Theories," Eq. 4.50.

CIVIL EFFECTS TEST OPERATIONS REPORT SERIES (CEX)

Through its Division of Biology and Medicine and Civil Effects Test Operations Office, the Atomic Energy Commission conducts certain technical tests, exercises, surveys, and research directed primarily toward practical applications of nuclear effects information and toward encouraging better technical, professional, and public understanding and utilization of the vast body of facts useful in the design of countermeasures against weapons effects. The activities carried out in these studies do not require nuclear detonations.

A complete listing of all the studies now underway is impossible in the space available here. However, the following is a list of all reports available from studies that have been completed. All reports listed are available from the Office of Technical Services, Department of Commerce, Washington 25, D. C., at the prices indicated.

CEX-57.1 ($0.75) The Radiological Assessment and Recovery of Contaminated Areas, Carl F. Miller, September 1960.

CEX-58.1 ($2.75) Experimental Evaluation of the Radiation Protection Afforded by Residential Structures Against Distributed Sources, J. A. Auxier, J. O. Buchanan, C. Eisenhauer, and H. E. Menker, January 1959.

CEX-58.2 ($0.75) The Scattering of Thermal Radiation into Open Underground Shelters, T. P. Davis, N. D. Miller, T. S. Ely, J. A. Basso, and H. E. Pearse, October 1959.

CEX-58.7 ($0.50) AEC Group Shelter, AEC Facilities Division, Holmes & Narver, Inc., June 1960.

CEX-58.8 ($1.00) Comparative Nuclear Effects of Biomedical Interest, Clayton S. White, I. Gerald Bowen, Donald R. Richmond, and Robert L. Corsbie, January 1961.

CEX-58.9 ($1.25) A Model Designed to Predict the Motion of Objects Translated by Classical Blast Waves, I. Gerald Bowen, Ray W. Albright, E. Royce Fletcher, and Clayton S. White, June 1961.

CEX-59.1 ($0.60) An Experimental Evaluation of the Radiation Protection Afforded by a Large Modern Concrete Office Building, J. F. Batter, Jr., A. L. Kaplan, and E. T. Clarke, January 1960.

CEX-59.4 ($1.25) Aerial Radiological Monitoring System. I. Theoretical Analysis, Design, and Operation of a Revised System, R. F. Merian, J. G. Lackey, and J. E. Hand, February 1961.

CEX-59.7C ($0.50) Methods and Techniques of Fallout Studies Using a Particulate Simulant, William Lee and Henry Borella, February 1962.

CEX-59.13 ($0.50) Experimental Evaluation of the Radiation Protection Afforded by Typical Oak Ridge Homes Against Distributed Sources, T. D. Strickler and J. A. Auxier, April 1960.

CEX-59.14 ($1.75) Determinations of Aerodynamic-drag Parameters of Small Irregular Objects by Means of Drop Tests, E. P. Fletcher, R. W. Albright, V. C. Goldizen, and I. G. Bowen, October 1961.

CEX-60.1 ($1.75) Evaluation of the Fallout Protection Afforded by Brookhaven National Laboratory Medical Research Center, H. Borella, Z. Burson, and J. Jacovitch, February 1961.

CEX-60.5 ($0.50) Experimental Evaluation of the Fallout-radiation Protection Afforded by a Southwestern Residence, Z. Burson, D. Parry, and H. Borella, February 1962.

CEX-60.6 ($1.00) Experimental Evaluation of the Radiation Protection Provided by an Earth-covered Shelter, Z. Burson and H. Borella, February 1962.

CEX-62.01 ($0.50) Technical Concept—Operation Bren, J. A. Auxier, F. W. Sanders, F. F. Haywood, J. H. Thorngate, and J. S. Cheka, January 1962.