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HAZARDS SUMMARY REPORT

for the

PWAR-3 CRITICAL ASSEMBLY EXPERIMENTS

by

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May 28, 1956

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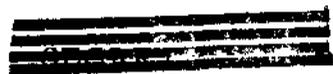
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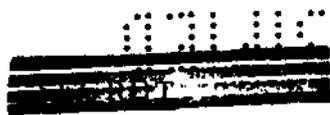
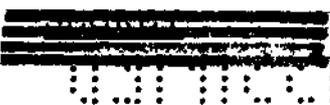


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Francis J. Jankowski, Robert F. Redmond,
Joel W. Chastain, and Sherwood L. Fawcett

Critical experiments are described and the hazards attendant to these experiments are evaluated for a beryllium-moderated, sodium-cooled, solid fuel element aircraft reactor design (PWAR-3).

The proposed critical assembly is composed of beryllium blocks, boron carbide simulated control cylinders, and fuel element assemblies. The fuel element assemblies are composed of strips of stainless steel, aluminum, and uranium foil contained in a stainless steel box. The assembly is arranged with the core axis horizontal, with one-half of the core on a fixed table and the other half on a movable table. Control and safety of the assembly is achieved by inserting or withdrawing certain fuel elements from the assembly.

The proposed critical assembly is not inherently safe against power surges. Accordingly, the possibility of an accident either from malfunctioning of equipment or from operator error is minimized by limiting the amount of reactivity available in any one operation. This is accomplished by interlocking the system so only one set of rods can be operated at a time and so that the movable table and rods can not be moved simultaneously. Furthermore, the speed at which reactivity can be added has been limited. The maximum normal rate of reactivity addition is estimated to be less than 6 cents/sec for the control rods and 4 cents/sec for table closure. In addition, the scram system can eject all the rods in about 0.2 second.

Hazards calculations are made to determine the dosage from direct irradiation, fall-out and inhalation from a radioactive cloud resulting from an accident. The exclusion area is shown to be adequate for even the maximum hypothetical accident.

INTRODUCTION

Pratt-Whitney Aircraft is considering a solid fuel reactor system as part of the aircraft nuclear propulsion project of the AEC. This reactor (known as PWAR-3) will use UO_2 -stainless steel fuel elements with sodium coolant and beryllium oxide or beryllium metal moderator. Because of the high neutron leakage and high fuel loading, the reactor will operate with an intermediate energy neutron spectrum. Control is accomplished with

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rotatable cylinders in the reflector region. These cylinders are loaded over a 120° sector with B₄C rods. The rest of the cylinder contains reflector material.

The physics and engineering analysis carried out by Pratt-Whitney has shown this type of reactor to have considerable promise. Owing to the importance of this project and the magnitude of effort which will be required to complete the development of a power reactor of this type, it is necessary that the physics characteristics of this reactor be verified by experiment at this time. Battelle has the facilities and the technical manpower to undertake these investigations and has been requested by Pratt-Whitney to perform the necessary experiments. A critical assembly is being constructed and will be operated in the Critical Assembly Laboratory at the Battelle West Jefferson Atomic Energy Center.

A Hazards Summary Report was submitted and a presentation made to ACRS in October 1954, which evaluated the Battelle site and the facility. This report presents the hazards considerations pertinent to the critical experiments for the PWAR-3.

Purpose of the Experiments

The purposes of the experiments are the following:

- (1) Determination of critical mass.

The calculation of the critical mass for an epithermal type of reactor requires elaborate multigroup methods. There has been only limited experience and little experimental verification of these calculations. The uncertainty associated with such computations makes a critical experiment mandatory before finalizing the design.

- (2) Investigation of control effectiveness.

The reactor control must give a safe margin of sub-criticality at room temperature at the start of core life and must permit the overriding of peak xenon at the end of life. Consideration will also be given to the effect of malfunctioning of some control elements.

- (3) Determination of the power distribution.

A peak to average power ratio of not more than 1.2 is desired both throughout the core and averaged over

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any one fuel element. The power distribution should meet these requirements at all times during the reactor lifetime and under various control conditions.

(4) Optimization of the reactor design if necessary.

The number, loading, and location of fuel elements, and the number, size, composition, and location of control elements, may be varied to optimize the three items given above.

The critical assembly used to perform these investigations will be operated at room temperature. Although the core, control cylinders, and fuel elements in the proposed power reactor have circular cross-sections, the critical assembly will be constructed of small elements of rectangular cross-sections stacked to simulate the shapes of the proposed reactor. This method of stacking the elements to simulate the given reactor shape will permit variations in loading, composition, and geometry. Control of the critical assembly will be accomplished by withdrawing fuel from the core and by separating the reactor into two equal halves.

This critical assembly will be the first reactor constructed and operated for this solid fuel reactor program. However, there have been several reactors built and operated for other programs from which experience can be drawn for aid in the design and safe operation of this critical assembly. In particular, the PPA at Knolls Atomic Power Laboratory, and the reflector-moderated reactor critical assemblies operated at Oak Ridge National Laboratory have been reviewed in formulating the mechanical designs and experimental procedures for this critical assembly program.

REACTOR AND ITS OPERATION

Site Location and Description

The site of the Critical Assembly Facility is in Madison County, Ohio, fifteen miles west from downtown Columbus. The property faces on the Georgesville-Plain City Road, an improved county highway which is not heavily travelled. The Battelle-owned land is bounded on the south by the Pennsylvania Railroad and on the east by Big Darby Creek which is the boundary between Franklin and Madison Counties. The tract contains a total of 400 acres and extends about 5000 ft. north from the railroad tracks. Figure 1 is a topographic map showing the site in relation to Columbus and the surrounding area.

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The Critical Assembly Laboratory is one of three buildings at the Battelle Atomic Energy Center which is located at the northern end of this property. The location of these buildings with respect to each other and the details of the immediate vicinity are shown in Figure 2. Figure 3 is an aerial photograph of the area and shows all of the Battelle land. The critical assembly laboratory is labelled ZPR. The nearest boundary of the property is over 1200 feet from the building.

The surrounding area is a farming community and is sparsely populated. The closest town is West Jefferson, population 1650, which is about two miles southwest of the laboratory location. The closest building is a barn, 2000 feet northwest of the site and the closest dwelling is 3100 feet to the southwest. During the summer months, a girl scout camp located across the river is inhabited. The site is about 2000 feet from the camp. The estimated total number of residents within the mile radius circle is 54.

The site is located on a level ground having an average elevation of 900 feet above sea level. The average elevation of downtown Columbus is approximately 750 feet. The change in elevation from the site to downtown Columbus takes place by a gradual rise over the 15 mile interval. A flat bottom ravine about 40 feet deep crosses the plot from east to west and is the bed of a small intermittent stream. The Big Darby Creek flows in a broad valley along the eastern boundary of the property and is approximately 50 feet below the elevation of the site.

Critical Assembly Building

The Critical Assembly Facility is a building having 10,500 square feet of floor space. The building contains the reactor assembly room, a control room, a vault, a counting room, an instrument laboratory, a shop, and rooms which may be used as offices or laboratory space. The first and second floor plans are shown in Figure 4. The building is constructed of concrete block faced with brick, with a structural steel frame except for the storage vault and assembly room. The storage vault is constructed entirely of reinforced poured concrete. The assembly room walls are two feet thick up to 26 feet. The wall above this and the roof are Q panel aluminum siding. All of this is supported by a heavy structural steel frame.

The arrangement of the assembly room, control room, and vault, and the associated stairwell, forms an area which can be shut off from the rest of the building.

The power provided for the installation is 300 KVA. Other utilities are compressed air, demineralized water, natural gas, and a 3-inch water supply main. Battery-powered emergency lights are provided in the assembly and control rooms.

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FIGURE 1. TOPOGRAPHICAL MAP

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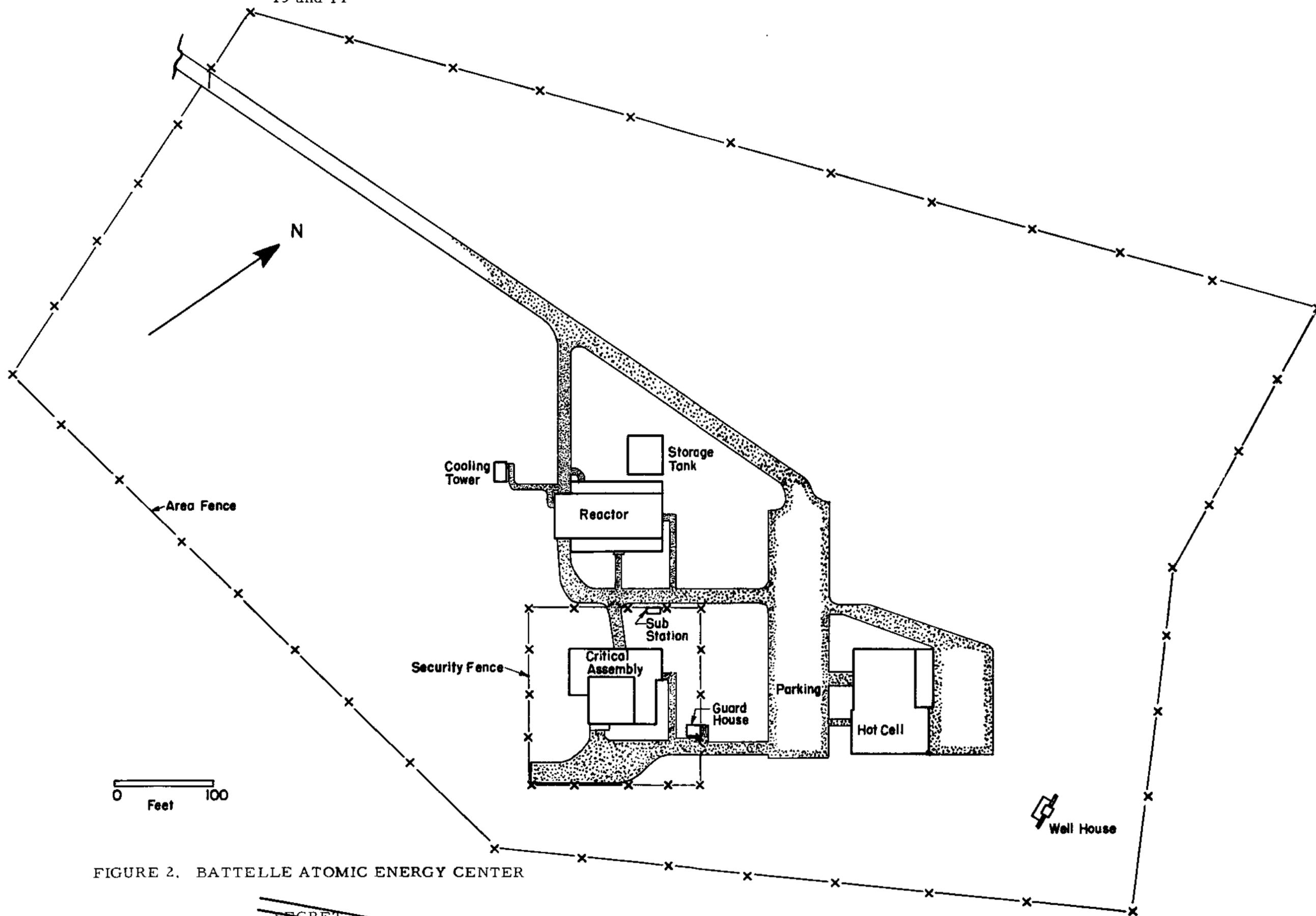


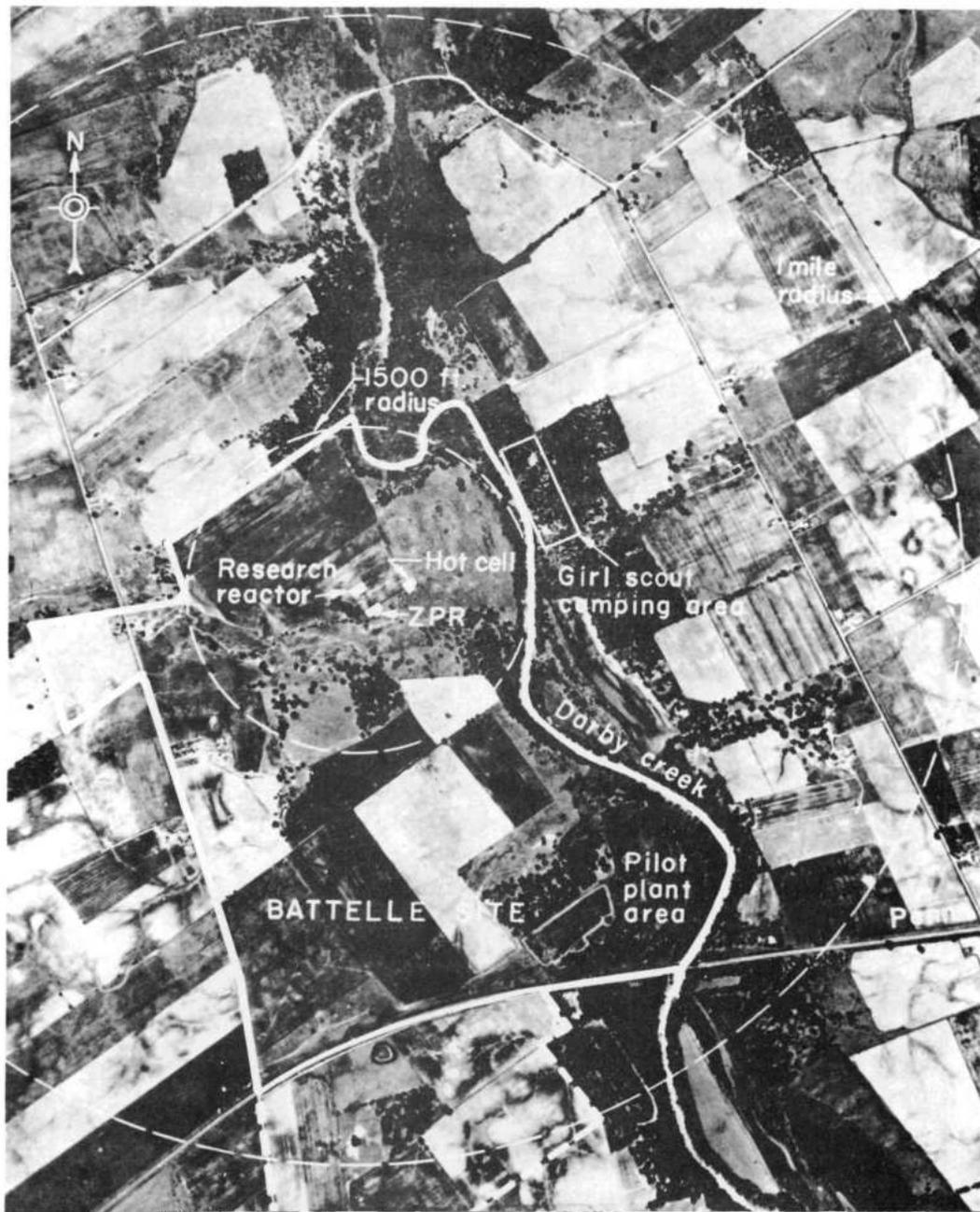
FIGURE 2. BATTELLE ATOMIC ENERGY CENTER

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FIGURE 3. AERIAL PHOTOGRAPH OF BATTELLE ATOMIC ENERGY CENTER

The building is heated by forced hot-water heat. The temperature is controlled by thermostats in the assembly room and by valves on the individual heaters in other locations. The control room, the instrument laboratory, and the counting room are air-conditioned.

Assembly Room

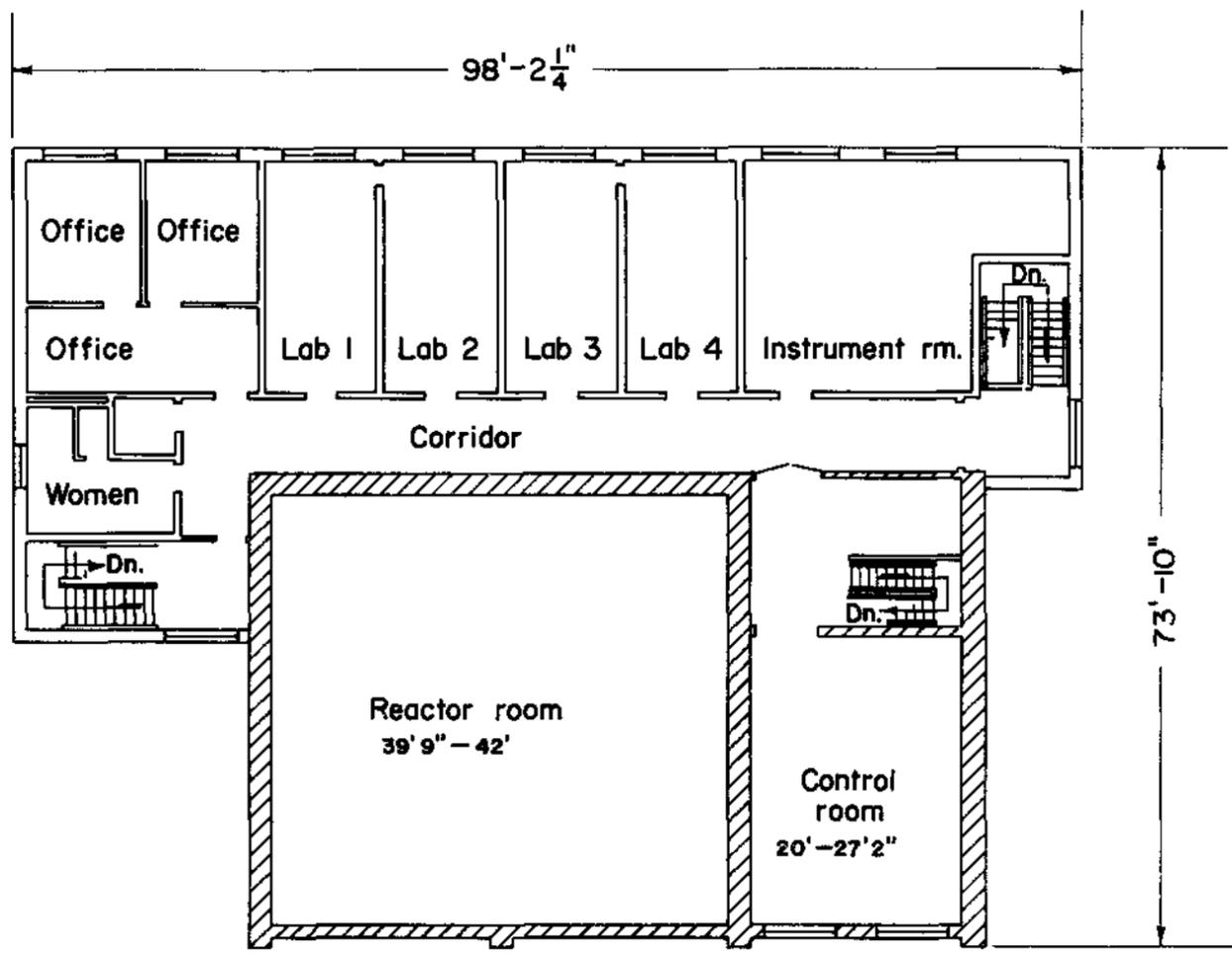
The reactor assembly room is approximately 40 foot square and 50 foot high. The three walls facing into the building are of reinforced poured concrete two foot thick. The other wall is 1 foot thick poured concrete. This concrete extends to a height of 26 feet which is enough to shield personnel in the inhabited portion of the building. Above this height the walls are constructed of Q-panel aluminum metal siding on the outside and steel sheets on the inside. The Q-panels have caulking material between them. The steel plates have lead tapes sealing the seams. This type of construction furnishes a tight enclosure so that activity from a possible accident will be contained or will be released very slowly to the atmosphere if the building is not damaged by the accident. The siding above the solid concrete wall is supported by structural steel framework which also supports a ten-ton crane. The roof is built-up of asphalt and gravel over heavy building paper which is supported by a Q-panel metal deck.

A large exhaust fan located in a penthouse above the office area supplies fresh air to the room. The exhaust is through the ceiling. Both the air inlet and the exhaust opening are equipped with louvers and a solenoid-operated sheet metal plate to make the openings tight when the fan is not in operation.

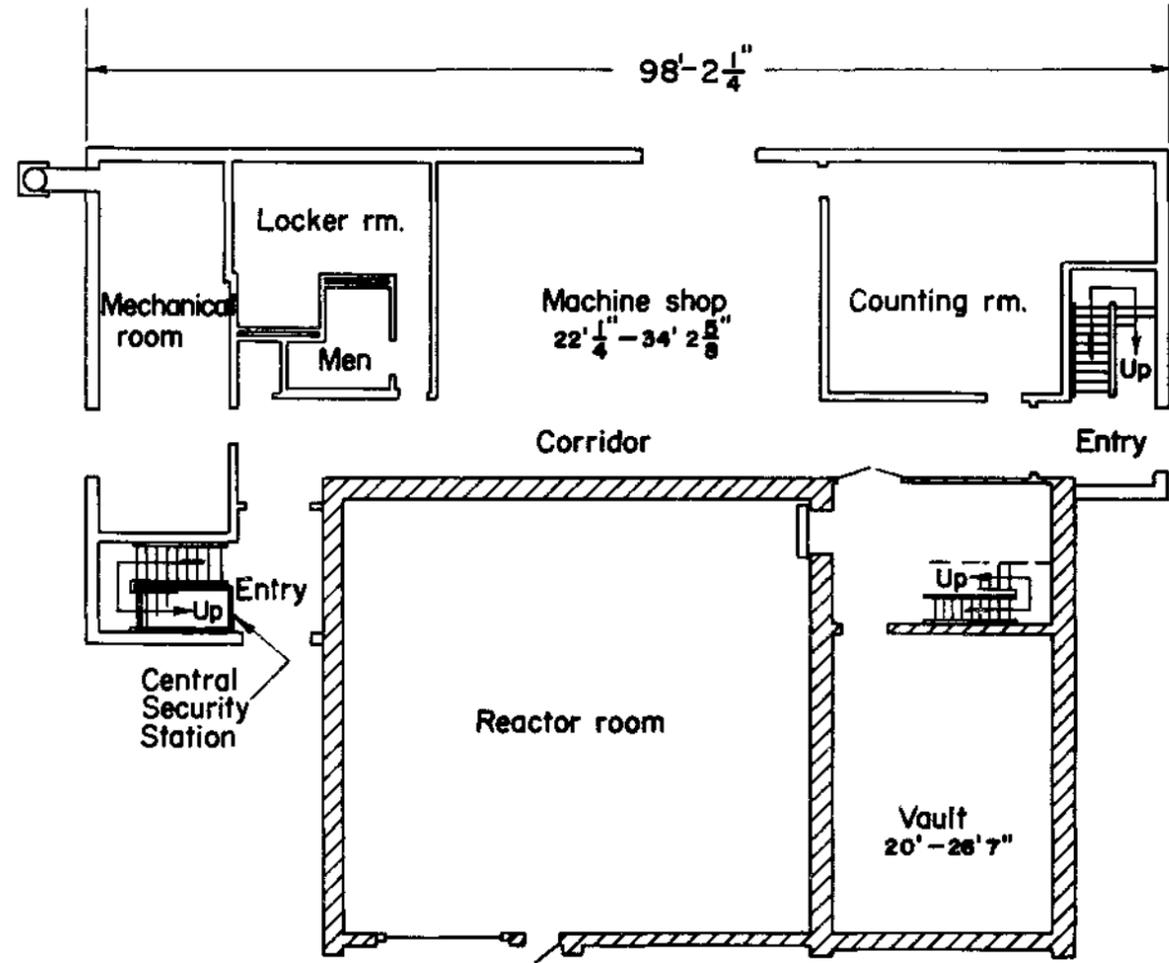
The assembly room has a single entrance into the remaining part of the building which can be closed off with a steel and concrete door having gamma shielding properties equivalent to the two-foot wall. In addition to this door there is a personnel entrance and a truck entrance to the outside which will be kept closed and locked during reactor operations.

There are a number of 4 in. conduit openings between the assembly room and control room for the passage of control wiring. These openings are at a 45° angle above head height to prevent radiation streaming from reaching personnel. The openings which are not in use will be fitted with shielding plugs. The ones in use will be filled with a material to make them reasonably gas tight.

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Second Floor Plan



First Floor Plan

Critical assembly area 

FIGURE 4. FLOOR PLAN OF CRITICAL ASSEMBLY BUILDING

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Control Room

The control room is located on the second floor adjacent to the assembly room. It is approximately 20 feet wide and 27 feet long. Services in this room include air conditioning and a dehumidifier. The control and research instrumentation are located along one wall.

Storage Vault

The storage vault is on the first floor beneath the control room and is approximately the same size, that is, 20 by 27 feet. The walls, floor, and ceiling are of reinforced concrete construction. Two walls, the floor, and ceiling are one-foot thick reinforced concrete. The remaining two walls are two-foot thick concrete. The steel vault-type door is equipped with a four tumbler manipulation-proof combination lock.

Counting Room

The counting room is located on the first floor so that the use of heavy shielding will not be a problem and so that foils of short half life can be removed from the assembly room to the counting room with a minimum delay. The room is approximately 21 feet square. No special wall construction is provided and shielding will be provided for individual detectors as required. There is a grounded bus bar running around the room. Air conditioning is provided to maintain the air at 75 F and 50 per cent relative humidity.

Office and Laboratories

Approximately 750 square feet of floor space is provided on the first floor for a machine shop. Equipment in this shop includes a power hacksaw, a band saw, a lathe, a drill press, a milling machine, a number of benches, and an assortment of hand tools. The machine shop has a truck entrance for bringing in bulky equipment.

An electronic laboratory for repairing instruments and for building up and modifying the instruments as required is located on the second floor above the counting room. This laboratory is air conditioned.

Seven additional rooms on the second floor provide space for offices, for a conference room, and for laboratory space should it be required.

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Description of Reactor Assemblies

The proposed power reactor for which these studies are being made is described briefly below. The critical assembly to be used in these studies, along with the associated controls and instrumentation is described in greater detail.

Power Reactor

The reactor design for which these studies are being made has a core which is a right circular cylinder 30 in. in diameter and 30 in. high. The fuel elements are uranium oxide dispersed in stainless steel wafers approximately 1 in. in diameter. These wafers are stacked in stainless steel tubes 30 in. long. Four small diameter tubes run through the wafers to provide coolant passages. In the present design there are between 200 and 250 of these fuel elements. The fuel elements and the moderator are cooled by molten sodium. The moderator is beryllium oxide or beryllium. The reactor power is controlled by 6 to 10 cylinders located around the periphery of the core. These cylinders are six to eight in. in diameter, having a 120-degree sector filled with boron carbide elements and the remainder filled with reflector material. The reflector regions between the control elements also contain beryllium oxide or beryllium.

The core and the control cylinders are housed in stainless steel shells. Stainless steel headers at the bottom and the top of the core provide for the introduction and removal of the sodium coolant. The gamma shield immediately adjacent to the reflector is closely packed tungsten carbide spheres.

Some of the desired characteristics of the reactor are that the peak to average power ratio over the core region be less than 1.2 and that the loading be between 60 and 80 kilograms of uranium-235. The control cylinders should provide approximately 4 per cent shutdown in the new cold condition and permit full control over the entire lifetime of the reactor. This should include overriding peak xenon at the end of the useful core life.

The critical assembly work will attempt to evaluate these factors and optimize them if necessary.

Critical Assembly

The core for the critical assembly experiment will be constructed with its axis horizontal and will be divided into two equal parts by a plane

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perpendicular to the axis. Each half of the core will be loaded onto a table-like structure. One table will be stationary while the other will be movable in a horizontal direction.

Aluminum grid tubes nominally 3 in. by 3 in. by 36 in. long will be stacked on the beds of both the movable and fixed tables. The simulated fuel elements and reflector pieces will be placed in positions in these tubes. Each table will have 576 tubes stacked in a 24 by 24 tube array as shown in Figure 5. The tubes will be held rigidly by clamps as shown. When the tables are run together, the tubes form a "honeycombed" cube which is 6 ft. on a side.

The critical assembly fuel elements will be made up of strips of uranium, and stainless steel, representing the matrix and cladding material, aluminum, representing the sodium and possibly other materials representing fission product poisoning. These strips will be stacked in 1 in. by 1 in. by 20 in. long stainless steel boxes as shown in Figure 6. The boxes are equipped with covers to form a closed container for the fuel elements in the mock-up which will simulate closely the elements in the reactor design.

The strips of material will occupy only 15 in. of the box length. The other 5 in. will be filled with a block of aluminum to mock-up the sodium in the header at the inlet or outlet to the fuel elements. Thus, a 20 in. long box simulates one-half (15 in.) of a reactor fuel element and the sodium in the header immediately above or below it. A small number of boxes will be made with dimensions of 1/2 in. by 1 in. by 20 in. to give flexibility in duplicating the reactor loadings. A typical loading cross-section is shown in Figure 7.

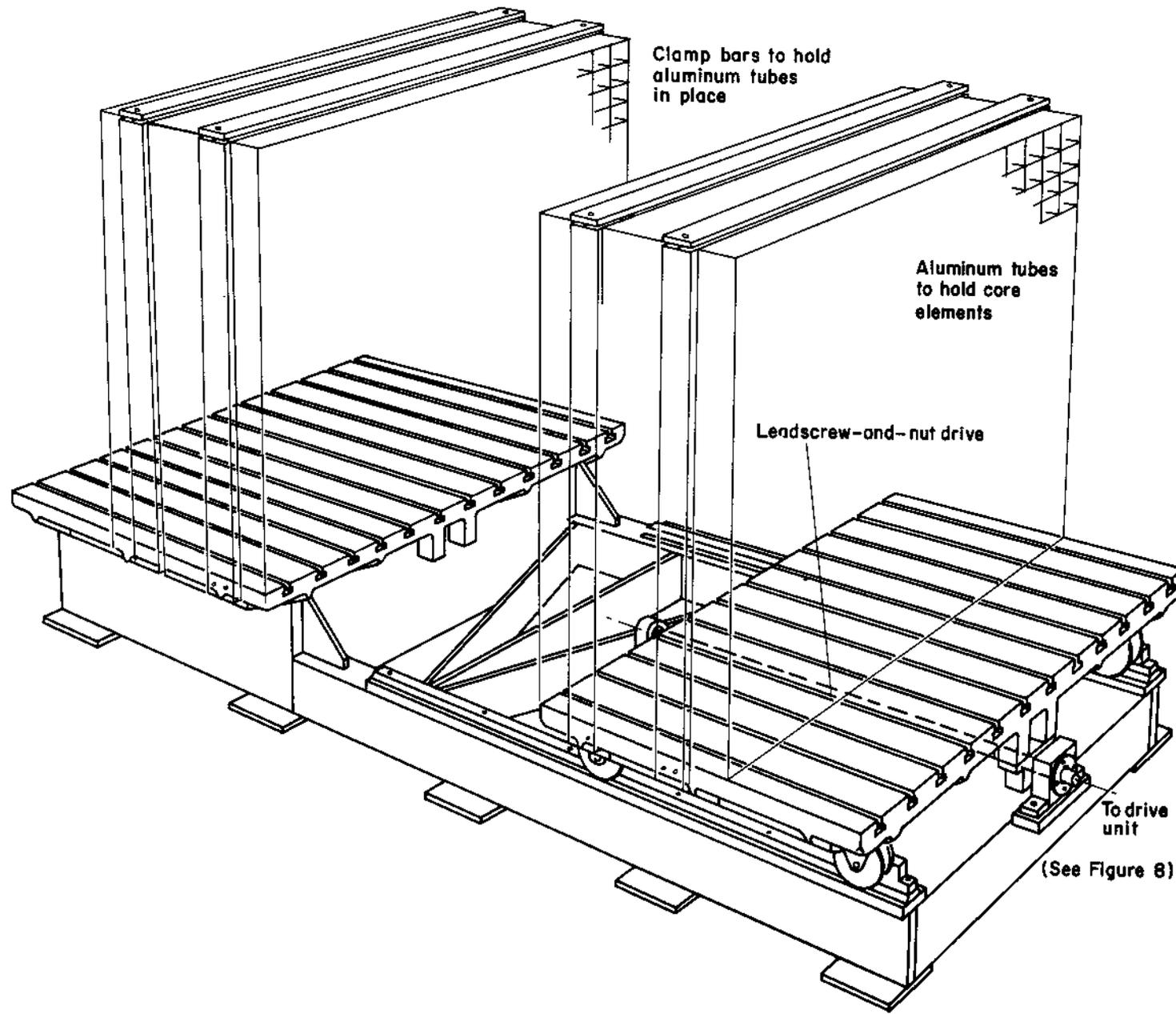
One hundred kilograms of uranium having an enrichment of about 93 per cent in the isotope 235 is being rolled into metallic foil to be used as fuel. The foil will be cut into various sizes as shown in Table 1.

TABLE 1. METAL STRIPS FOR CRITICAL ASSEMBLY FUEL

No. of Strips	Thickness	Width	Length
5830	0.004 in.	0.882 in.	15.000 in.
600	0.003 in.	0.882 in.	15.000 in.
600	0.002 in.	0.882 in.	15.000 in.
40	0.001 in.	0.882 in.	15.000 in.

The beryllium moderator blocks will be cut on a one-inch modulus, with many of them greater than the one inch size so as to decrease the total number of blocks which must be handled. A few of the blocks will be only one-half inch thick to permit greater freedom in locating the fuel elements.

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Clamp bars to hold aluminum tubes in place

Aluminum tubes to hold core elements

Leadscrew-and-nut drive

To drive unit

(See Figure 8)

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FIGURE 5. REACTOR TABLE

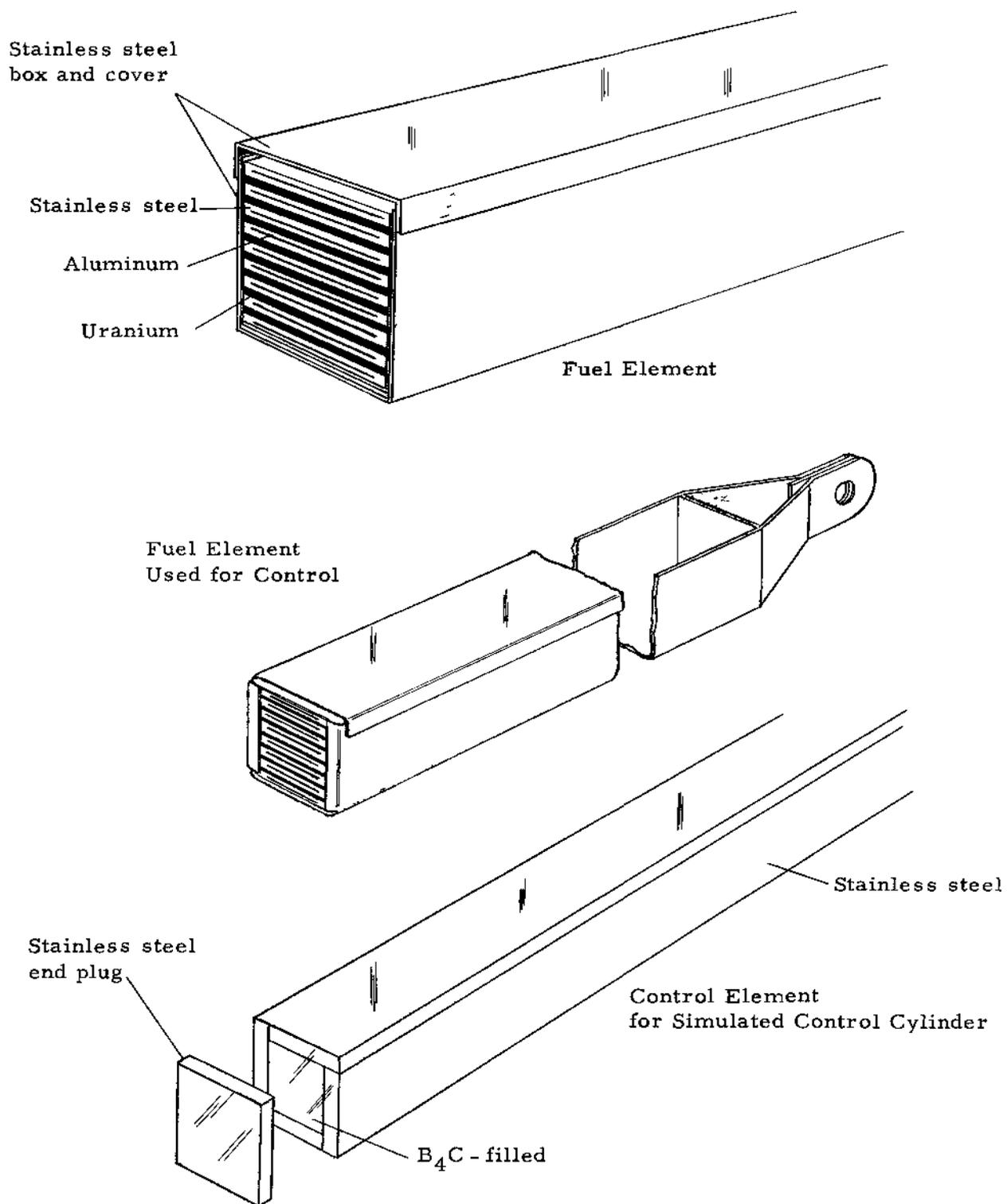


FIGURE 6. CRITICAL ASSEMBLY CORE ELEMENTS



Control Drum Outline

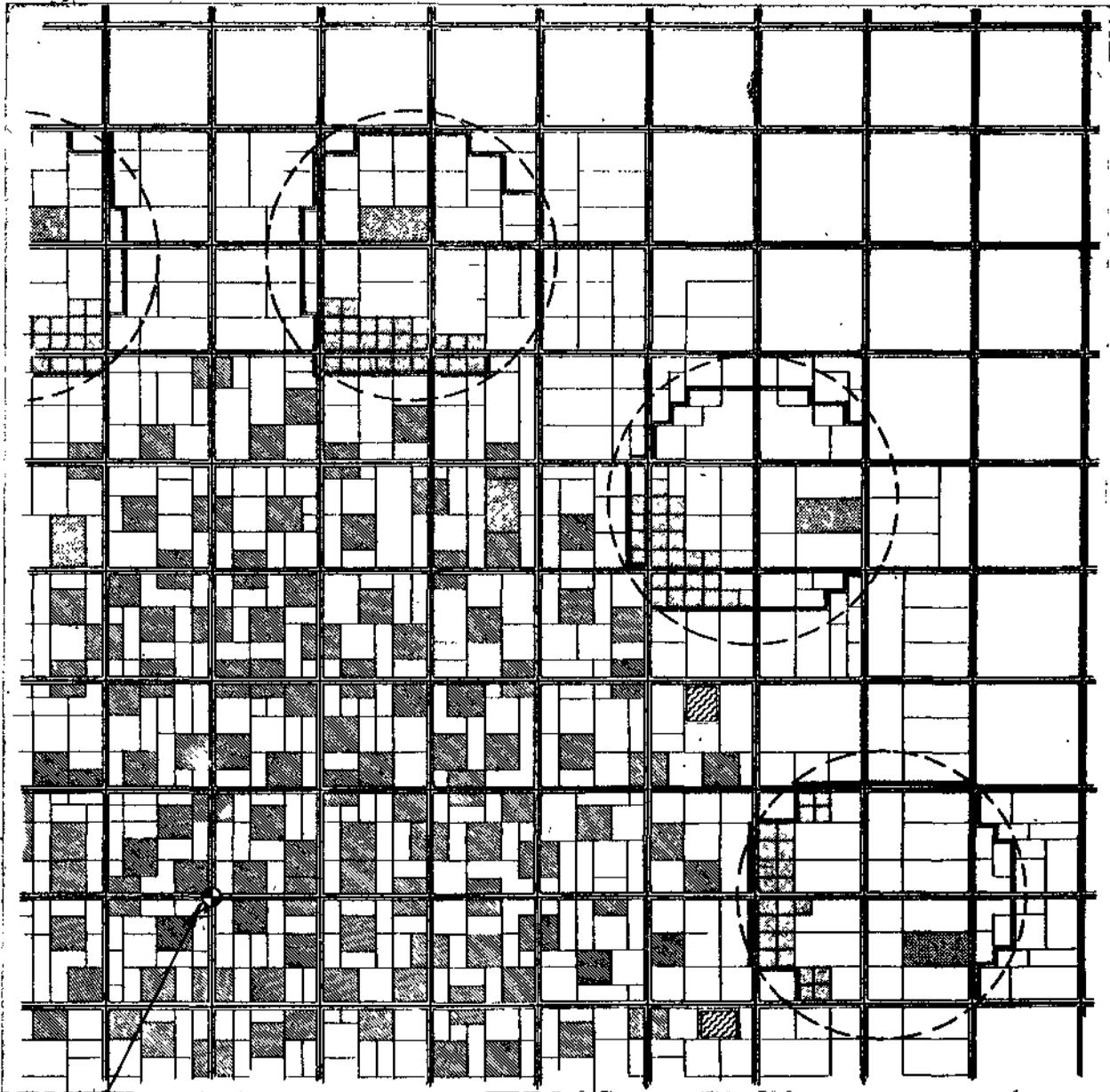
B₄C-filled Control Element

Moderator

Fuel Element

Control Fuel Element

Grid Tubes



Central axis of core

FIGURE 7. SECTION OF TYPICAL CORE CONFIGURATION

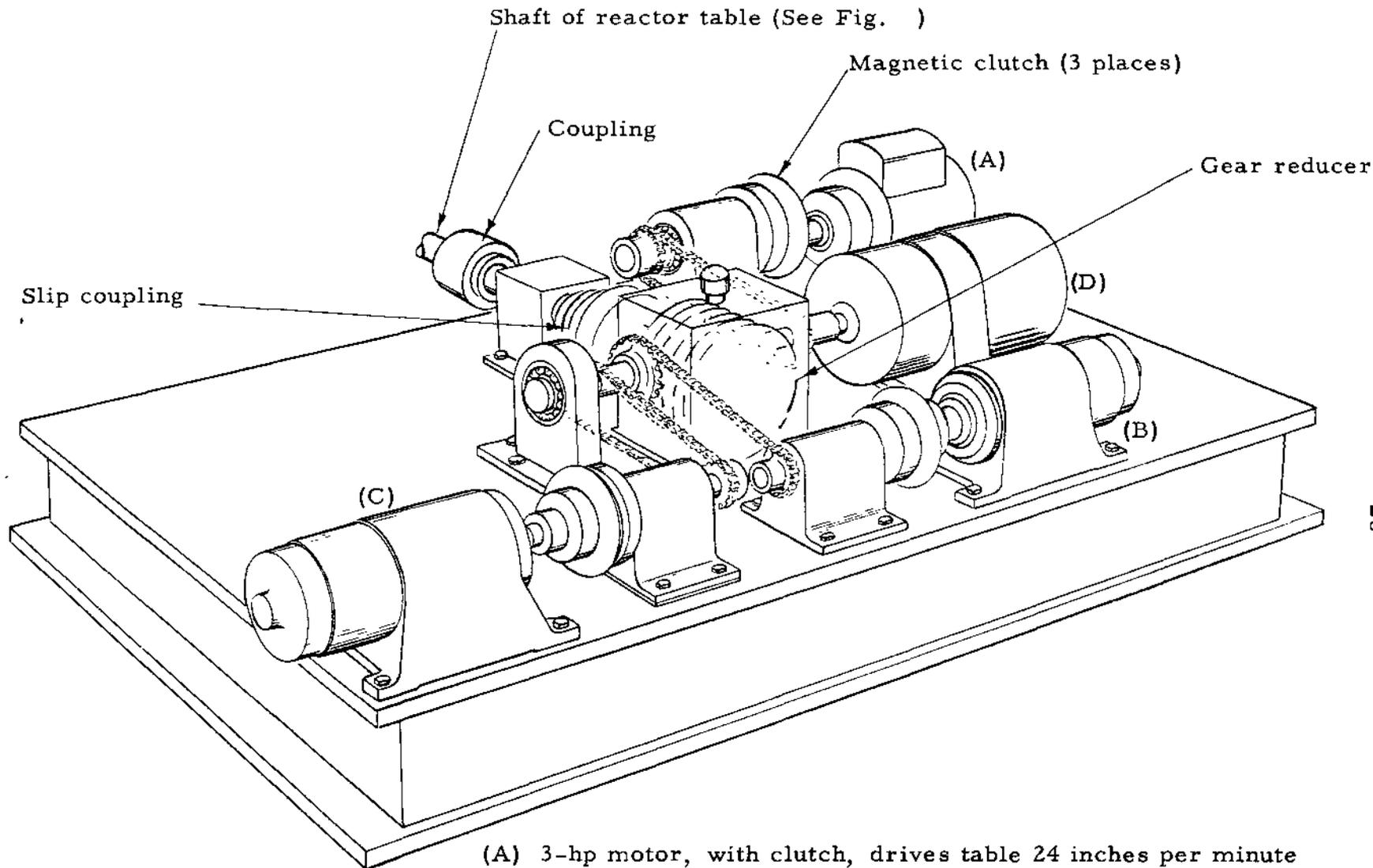
A few of the blocks will be cut slightly undersize. This will permit insertion of special square cross-section tubes to act as guides for the removable fuel elements used for control. It will also permit stainless steel strips to be inserted to simulate the core and control cylinder shells.

The elements in the simulated control cylinders will be made of stainless steel bars welded to form a square tube approximately 1 x 1 x 20 inches (see Figure 6). The central opening in the tube will be filled with boron carbide to a density of approximately 0.85 g/cm³. These simulated control tubes will fill approximately a 120 degree sector of simulated control cylinder. Rotation of the cylinder in the reactor will be simulated in the critical assembly by rearranging by hand the position of the boron-carbide control elements.

The critical assembly will actually be controlled by withdrawing certain fuel elements from the assembly. These fuel elements (shown in Figure 6) will be made up in a manner similar to the others except that the stainless steel boxes will be 36 in. long with means at one end for attaching the driving mechanism. The 36 in. length will permit the control element box to extend through the half of the core in which it is located. There will also be provisions to prevent the strips from sliding in either direction and for holding the lid tightly on the box. These provisions will not be made in the other fuel boxes. The control fuel elements will slide in stainless steel tubes which will allow more than 0.080 inch clearance between the control element and the tube. The stainless steel guide tubes will be attached to the aluminum grid tubes by brackets which will position the guide tube end and will prevent longitudinal motion of the guide tube.

Each of the tables in the core support structure will have a machined flat surface, 5 ft. by 6 ft., and will have T-slots on 6-inch centers for bolting down the parts of the core. The movable table will be supported on four wheels, rolling on a track which will also guide the motion of the table. The total table travel will be five feet.

The movable table will be driven by a lead screw with the nut attached to the table. The driving force will be supplied by four different motors, only one of which will operate at a time. The drive unit is shown in Figure 8. A 3 horsepower motor, operating through a magnetic clutch, will drive the table at 24 inches a minute through the first 3-1/2 feet of travel. A 3/4 horsepower motor, also operating through a magnetic clutch, will drive the table at six inches per minute for the next 16 inches. The final closure will be by another 3/4 horsepower motor, operating through a magnetic clutch, which will drive the table at one-half inch per minute. The tables will be driven apart by a 3 horsepower motor connected directly to the gear reduction box. This arrangement will separate the tables at 40 inches per minute.



- (A) 3-hp motor, with clutch, drives table 24 inches per minute
- (B) 3/4-hp motor, with clutch, drives table 6 inches per minute
- (C) 3/4-hp motor, with clutch, drives table 1/2 inch per minute
- (D) 3-hp motor, with no clutch, drives table 40 inches per minute

FIGURE 8. TABLE-DRIVE UNIT

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A friction clutch will be used to connect the driving unit to the lead screw. This will permit the movable table to be driven firmly against the stationary table to insure a reproducible position and will also prevent the shearing of pins or damaging of gears in any case of over-driving.

Control of Critical Assembly

The critical assembly will be controlled by a group of safety and control rods, which will move fuel elements out of and into the core, and by separating the reactor tables. A system of electrical controls and interlocks and nuclear and mechanical instrumentation will tell the operator the configuration and reactivity condition of the reactor at any instant and will prevent unsafe operation of the assembly. A remotely-controlled neutron source in each core half will be present for loading measurements, start-up, and experimental use.

Mechanical Control

To obtain the necessary degree of control, it will be necessary to drive approximately 10 per cent or about 25 of the fuel elements. To keep the number of drive units to a reasonable number, the fuel elements being driven as safety rods will be ganged mechanically so that from three to five are driven together.

The type of drive unit which will be employed is shown in Figure 9. This figure shows the principles of operation. The drive units may differ from this description in minor details. The drive will use an air cylinder for inserting the fuel elements and a spring for withdrawing and scram. The construction of the cylinder will be such that the exhaust port will be reduced in the last few inches of the 18-inch safety rod travel, thus producing a cushioning effect to prevent the rods from stopping too suddenly.

The control or regulating rod drives will use DC motors and rack and pinion units for moving the element. For the present, only one control element is planned for each core half. If more control is found necessary, up to five fuel elements can be connected to one control rod drive. The control rod drive is connected to the control fuel element in a manner similar to that used for the safety rods. However, the control rods have no scram springs.

The position of both safety rods and control rods will be given by Selsyn indicators geared through rack and pinions to the driving shafts. This method is shown in Figure 9. The safety rods will have but one Selsyn which will give the position of the rod to the nearest 0.2 in. The

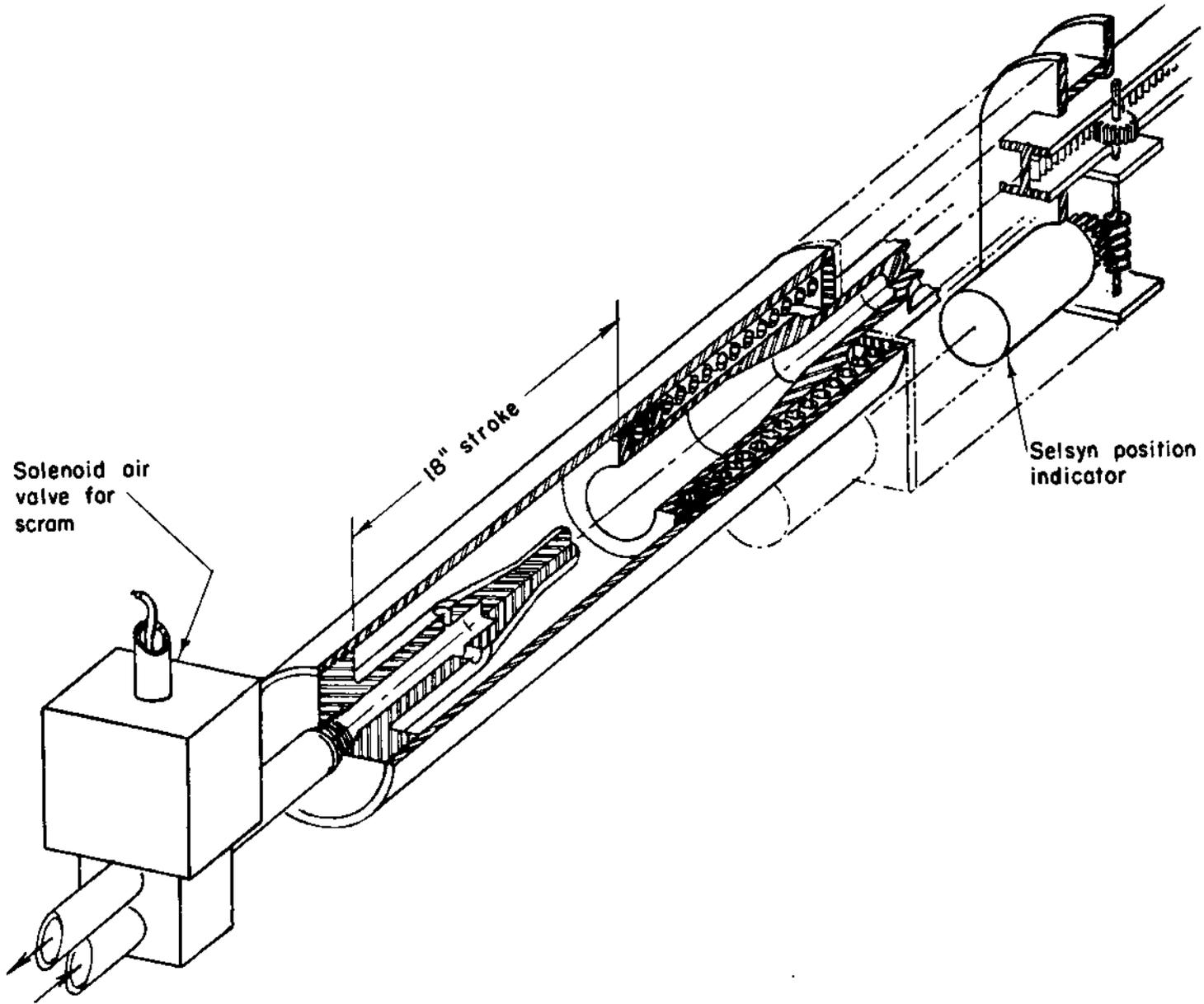


FIGURE 9. SAFETY ROD DRIVE UNIT

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control rod drives will have double Selsyns, one giving coarse position and one giving fine position, so that they may be positioned to the nearest 0.01 in.

There will be a neutron start-up source in each reactor half which may be withdrawn into shielded containers during the critical reactor operations. Each source will be in a capsule approximately 0.225-in. diameter and 1-in. long with a ball on top to permit connecting to the driving unit through a ball and socket connection. The sources will enter the reactor from the reflector end in stainless steel tubes which will penetrate the core approximately eight inches.

The source will be pushed into the core and pulled out by means of a flexible cable. This cable will have a spirally wound wire on it and will be driven by a tooth gear so that the driving gear and cable have the appearance of a worm and worm gear, but will act like a rack and pinion unit. The driving power will be from a geared DC motor. Limit switches operating off the end of the flexible cable will prevent the overdriving of the source in either direction. An interlock switch which will be operated over the last 7 inches of source travel will permit the operation of the rod drive and table drive when the neutron flux is not above a prescribed level.

Figure 10 shows the frame which will be mounted on the tables behind the stacked grid tubes to support the safety and control rod drives and the source drive and container. This sketch shows five safety rod drive units in position with a perforated disc connected to one to which a number of movable fuel elements of the type shown in Figure 6 may be attached. On top of the frame is the shielded source container with the source drive. Control rod drives are not shown.

Electrical Controls

The means of operating and controlling the safety rods is shown in Figure 11. The power supply leads go through a reversing switch, a selector switch, and then through rectifiers to two solenoid controlled air valves. The rectifiers permit the operation of but one solenoid valve at a time depending on the polarity of the system. The two solenoid valves permit the slow operation of the safety rod in either direction and permit positioning of the rod at intermediate positions. The air for driving the rod goes through a filter, a pressure reducing valve, and lubricator, the controlling solenoid, a ball check valve, and a needle valve which provides an orifice for limiting speed of rod travel. A rupture disc type of safety valve is included to prevent serious overpressure on the cylinder, which would delay scramming action.

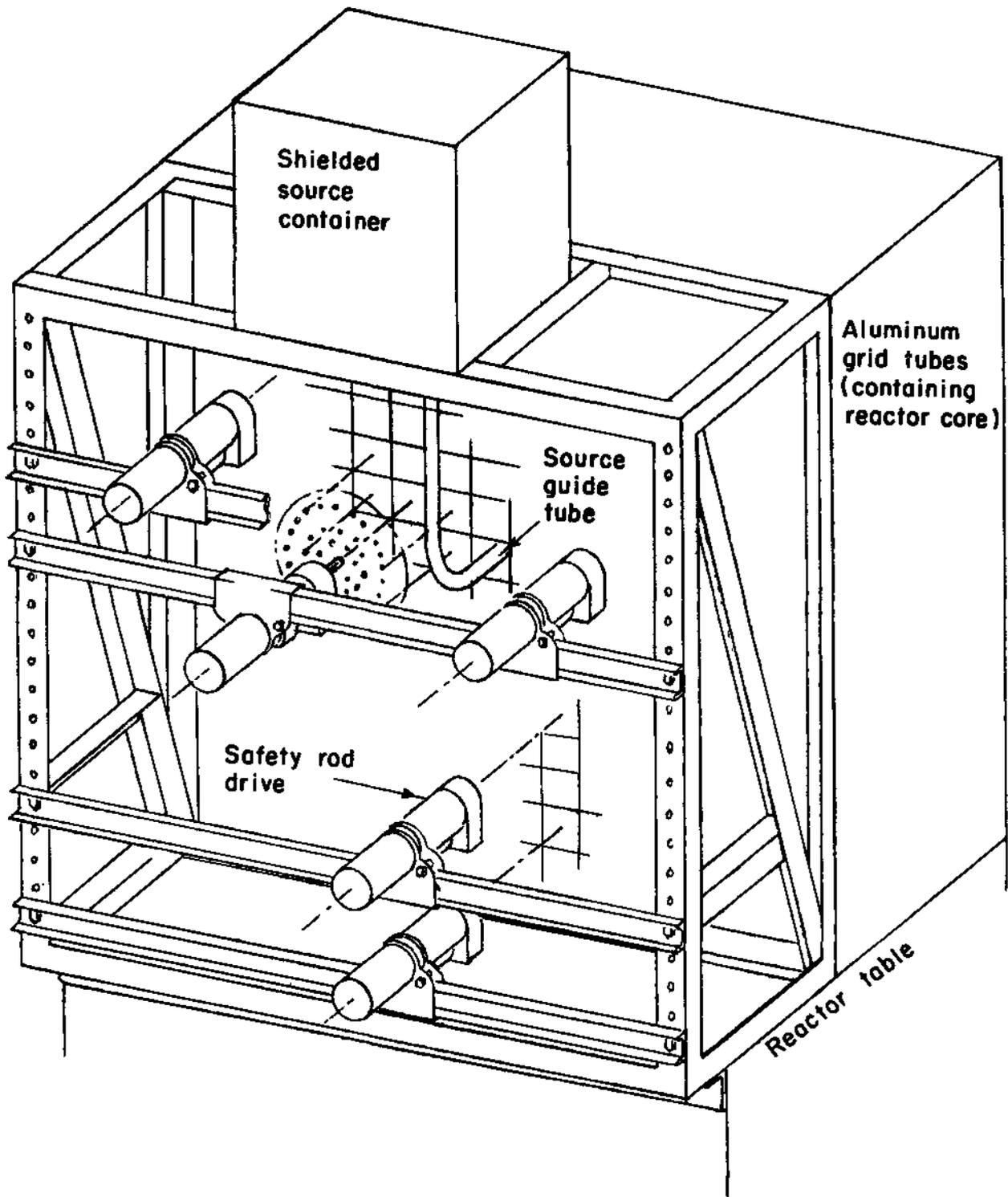


FIGURE 10. FRAME TO HOLD REACTOR CONTROL UNITS

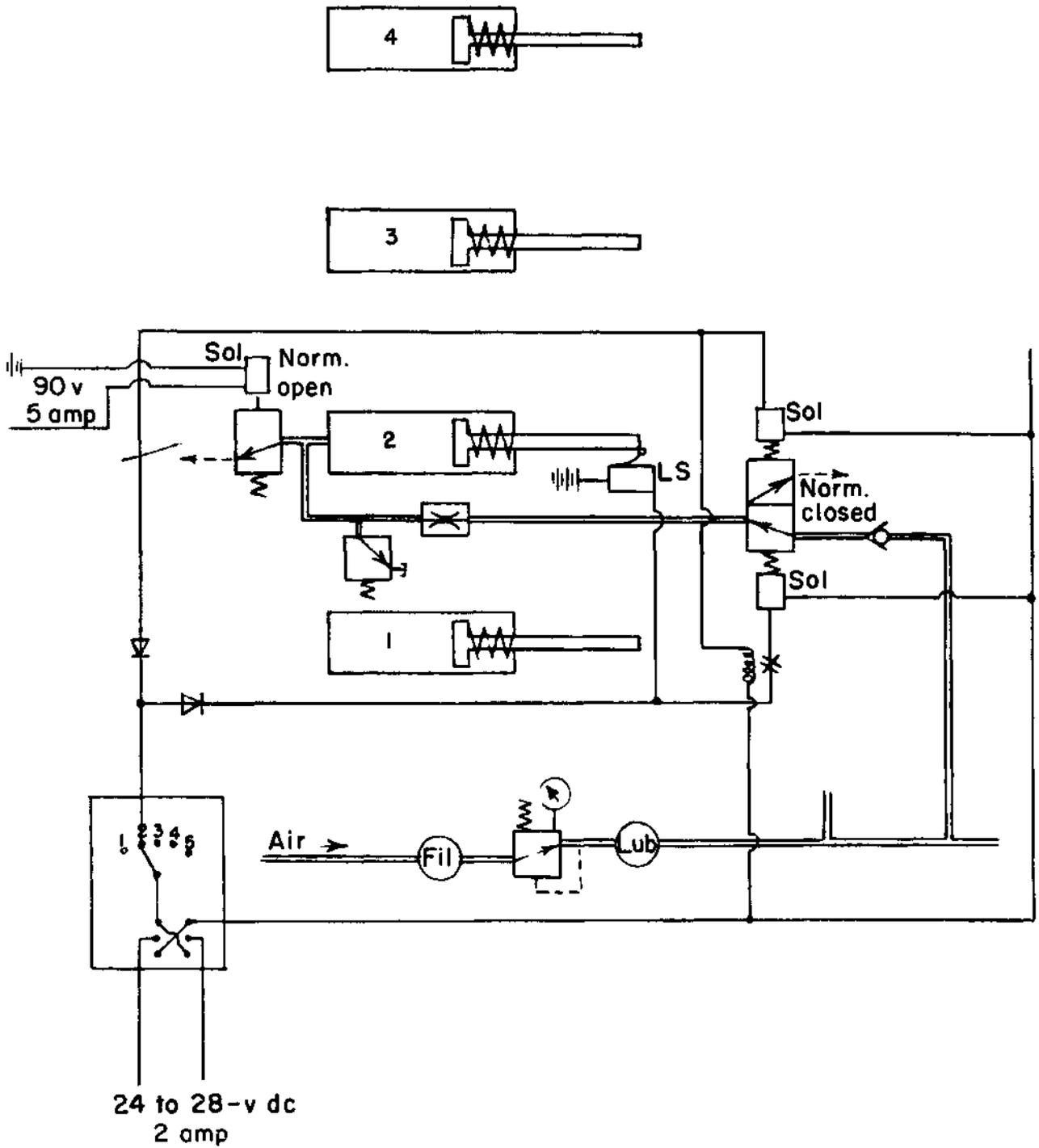


FIGURE 11. SAFETY ROD CONTROL SYSTEM

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Scramming is provided by a normally open 90 volt solenoid having a large port area. This solenoid is energized and held closed for normal operation of the rod. Interruption of the current to the solenoid permits the valve to open with a subsequent quick travel of the rod.

The elemental circuit for the control rods is similar to the safety rod circuits, with two exceptions. There is no scram circuitry on the control rods and the two controlling valve coils are replaced by two sections of the field in the DC controlling motor. The two types of drive systems are therefore identical from the viewpoint of controlling from the console.

To hold the safety rods in the "in" position, a limit switch and separate power supply are provided. These units apply power to the "air in" valve when the rod is at the full in position. When the control switch on the console is operated to withdraw the rod at normal speed, a relay in the circuit withdraws this holding power from the "air in" solenoid valve to permit the opening of the normal exhaust valve to reduce the air pressure and allow the rod to move.

The table drive motors all operate on 220 v 3 phase power and will be controlled by relays. (See Figure 12.) The safety motor, which drives the tables apart, is connected to its power supply through a normally closed relay and a normally closed limit switch. The relay is held open by power from the scram circuits; thus a scram from any source will remove the power from the relay coil allowing the contacts to close driving the tables apart. When the table reaches the full out position, it will open the normally closed limit switch to remove power from the motor. To insure that this limit switch closes when the table is driven in, the power for the final two "in" driving motors also is taken through this switch. Thus, if power is not available to separate the tables, it is not available for bringing the tables together.

The three "in" motors are controlled by normally open relays which close when the power is applied by the switch on the console and the proper cam-operated-unit switch is closed. This same operation also energizes normally open relays which apply power to the clutches. The line from the console switch to the relays for each motor passes through two normally open switches in series. These switches are closed by cams on the table; the size and location of the cams determine which motor will drive the table at any given table location.

No provisions will be made for reversing any of the motors. The table will always be separated at the maximum rate by the safety motor. Commercial control units will be installed which guarantee against failure of one phase permitting this motor to reverse.

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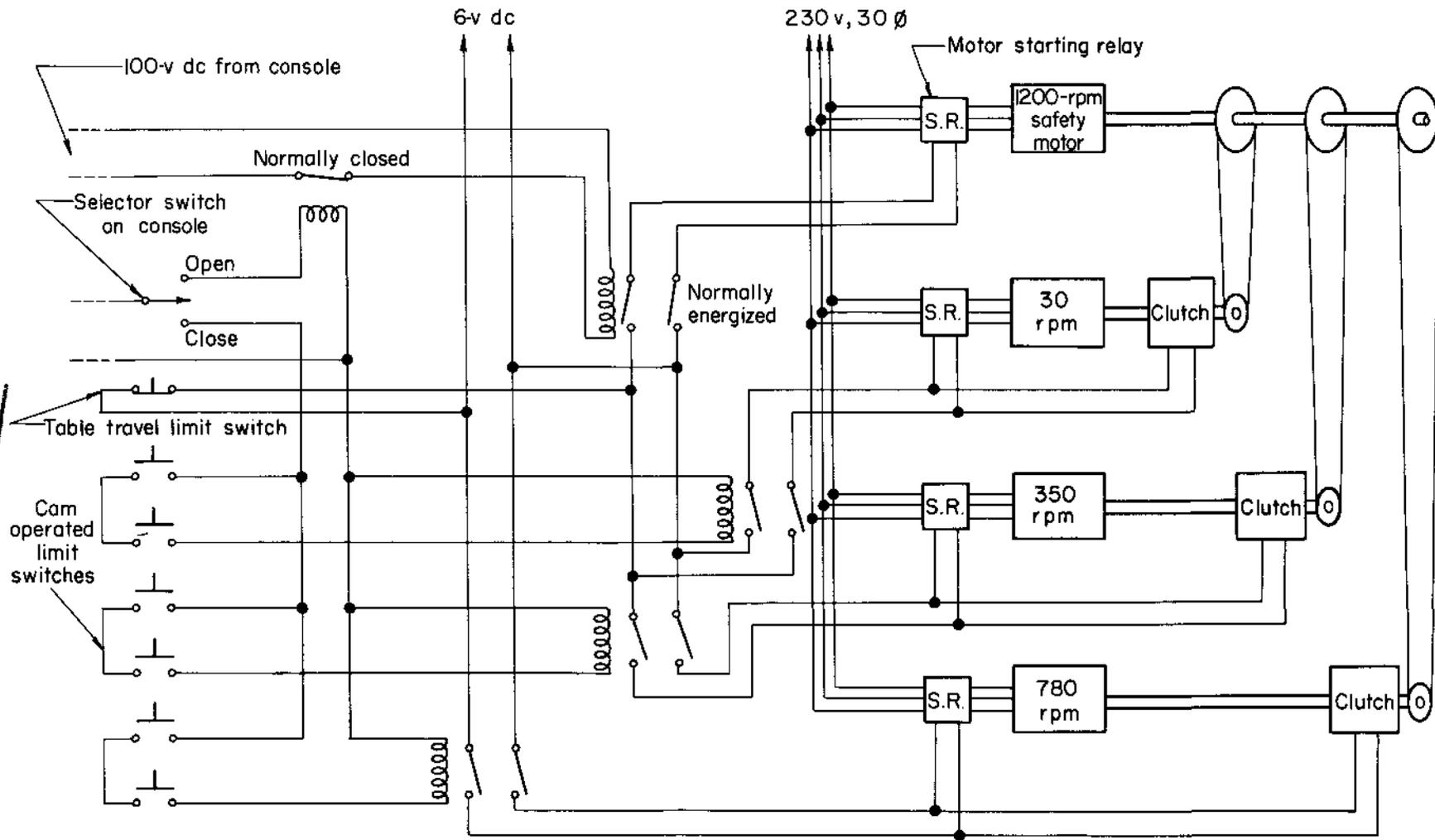


FIGURE 12. TABLE DRIVE CONTROL

All relays normally open except as noted.

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The source is driven into or out of the reactor core through the operation of a switch on the console. The source drive is independent of the remaining part of the control system except for the interlock switch which is independent of the source drive itself.

The interlock and "scram" circuits are entirely switch and relay circuits (no electronics). The circuits are designed to actuate on the opening of switches and relays. Relays are connected so that they actuate on removal of power from their coils. In circuits containing large inductive loads, precautions against arcing at relay contacts have been taken. These precautions include the use of double break relay contacts and the use of selenium rectifiers to short circuit the inductive e. m. f.

The functions of the interlock and "scram" circuit are outlined in Figure 13.

Control and Research Instrumentation

The bulk of the reactor instrumentation is contained in a console type control panel in the control room. This console is designed to mount instruments with standard 19 in. relay-rack panels. The relay system is also built up on standard relay-rack panels. Each panel is wired into the system through barrier terminal strips on the unit and fanning strips on the connecting cables. This facilitates removal of any instrument or unit for modification or repairs and also allows access to the terminals for testing and trouble shooting. Where low level signals require it, separate cables of shielded audio leads or coaxial cable are used. Signal cables are grounded at the instrument end only.

Meters and controls necessary for the operation of the critical assembly are located in a position in front of the operator. Above this, on an inward sloping panel, are located the rod position indicators. Meter relay monitors, recorders, and the closed circuit television screen are located around the periphery of the console.

The nuclear instrumentation is shown by the block diagram in Figure 14. Three compensated ion chambers are used as neutron detectors and monitors. One chamber is connected to a linear vibrating reed type electrometer on the console. Negative feedback to the input lowers the input impedance and permits the use of a long cable. The output of this instrument is used for the visual indication of neutron flux. This output is also recorded on a circular chart recorder and is monitored by a meter relay for the safety circuits. The other ion chambers are connected to logarithmic electrometers in the assembly room. The output of these electrometers is connected directly to the safety circuits and also to differentiating circuits on the console to provide visual period indications and

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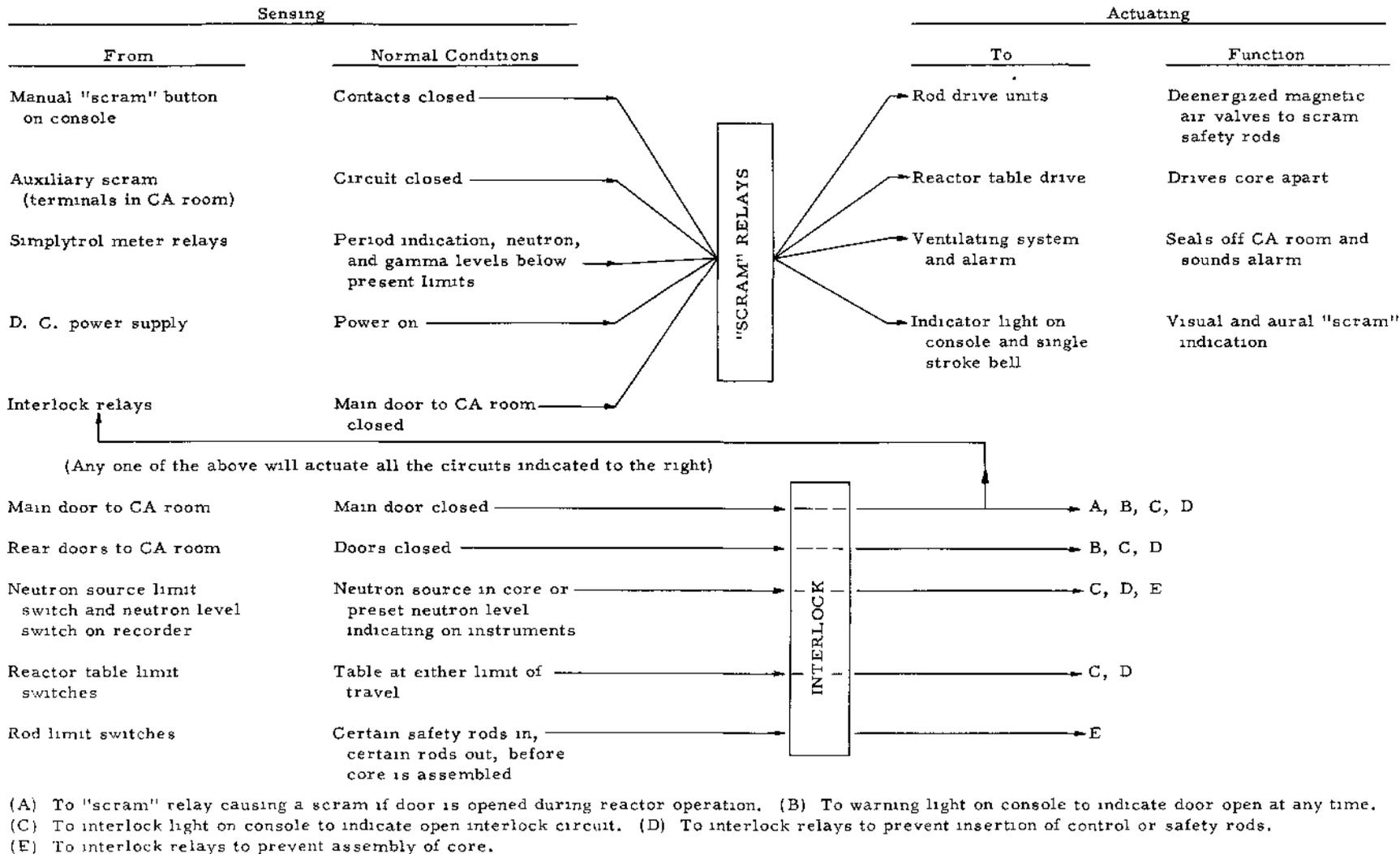


FIGURE 13. INTERLOCK AND SAFETY SYSTEM

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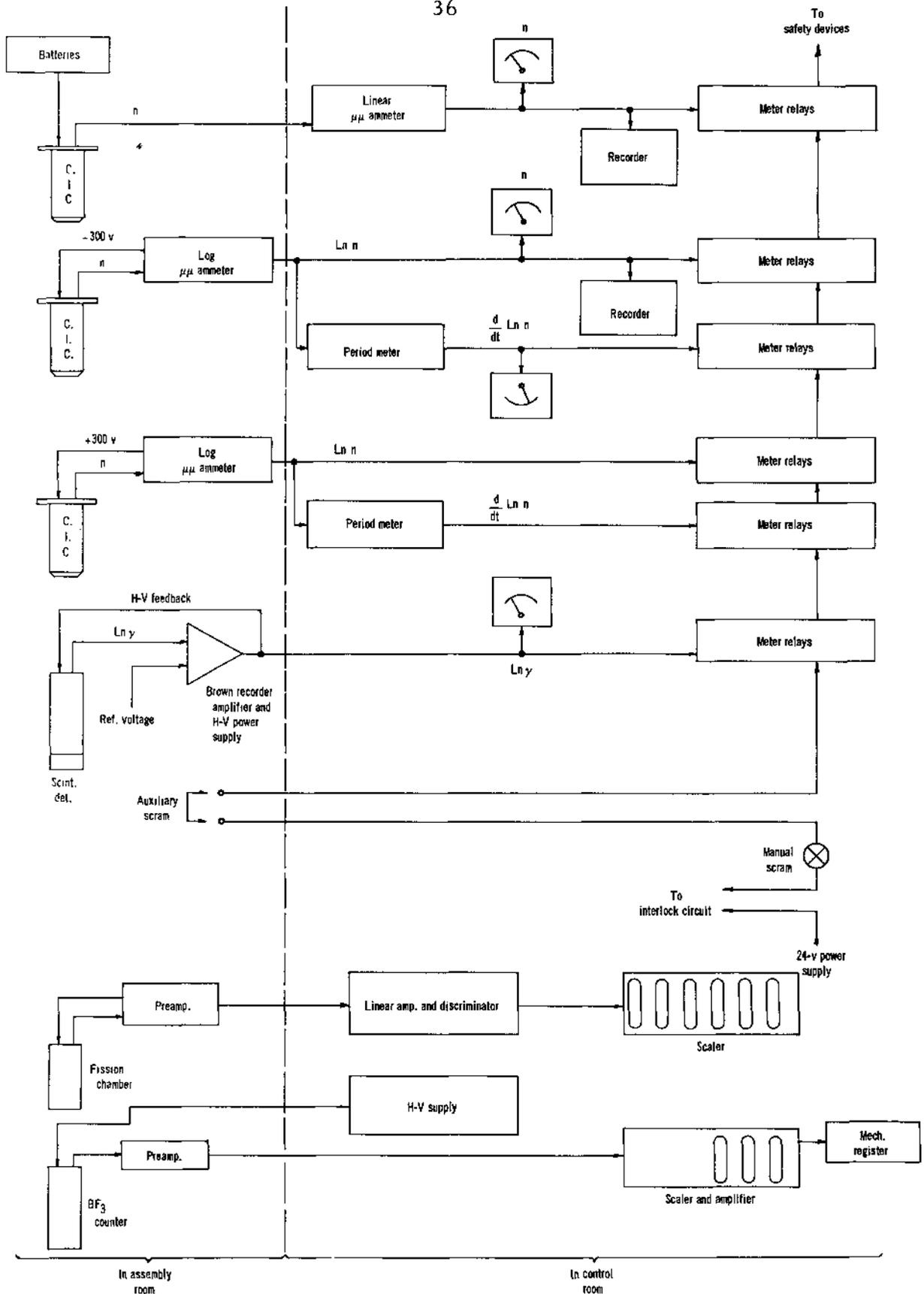


FIGURE 14. BLOCK DIAGRAM OF REACTOR INSTRUMENTATION

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to actuate the safety circuits. On one of these channels the ion chamber will not be compensated since it will be used entirely as a safety circuit. A strip chart recorder records the log level. The slope of the curve on this chart will be a record of the period.

A six decade, wide range, gamma channel consisting of a scintillation detector connected to a servo amplifier gives a log indication of gamma flux and is also connected into the safety circuits.

In addition to the safety or control circuits two research circuits are provided. These channels provide accurate low level information for start-up and accurate neutron level indications at operating levels when required. They can also be used to provide an aural indication of power level and to provide additional visual indication of power level. Both of these channels use pulse counting instruments and are independent of the safety instrumentation. One of these is a fission chamber channel consisting of a linear amplifier and a scaler with provisions for a digital recorder. The fission chamber and preamplifier are located in the assembly room. The other channel uses a boron trifluoride counter as detector. It consists of a scaler with a built-in amplifier and high voltage supply. The counter and its preamplifier are located in the assembly room.

Rod and source positions are indicated by Selsyn indicators. Two sets of Selsyns are utilized for each control rod drive to obtain the required accuracy. Only one set of Selsyns is used with each safety rod since these rods will normally be full in or full out.

The table position will be shown by selsyns also. Two sets of Selsyns will be geared to the lead screw to indicate table position to 0.01 in. over its entire travel. Up to three units of two sets of Selsyns each will be used to indicate final closing at various points on the assembly to an accuracy of 0.001 in. These Selsyns will operate by the table pushing against a spring loaded plunger containing a rack which is geared to the Selsyns. In addition, indicator lights will show which drive motor is operating. If the indicator light and table position do not agree with the schedule built into the table drive, operations will be discontinued until the trouble is found and corrected.

A closed circuit television with a 17-inch screen is located on the console to provide remote viewing of the critical assembly experiment.

Actuating controls on the console are lever switches for operation of rod drives, table drives, and neutron source drives. A selector switch is used to select the rods to be manipulated. A manual "scram" button on the control panel is provided for emergency shutdown. Two terminals in the assembly room are wired into the circuit of the scram button so that experimental setups may be wired into the "scram" circuit. This is indicated as an auxiliary scram in Figure 13.

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Types of Experiments

The discussion below lists the types of studies to be performed with this critical assembly. The experiments mentioned are not listed in any specific order. The order in which the experiments are performed will depend first, on the need for particular information and, second, on scheduling which will attempt to minimize the time required for the over-all program.

Critical Mass Studies

The uranium-235 loading will be determined for the proposed reactor design and for possible modifications in this design. Modifications which may effect the critical mass are the fuel arrangement, the core size, the number, size, and composition of control elements, the materials ratios in the core, and the reflector thickness.

Control Evaluation

Attempts will be made to measure

- (1) The reactivity worth of each of the control cylinders,
- (2) The subcriticality of the reactor at the beginning of life with all the cylinders turned in,
- (3) The worth of one cylinder as a function of position for various positions of the other cylinders,
- (4) The effect of changes in size, number, and composition of the control cylinders.

These quantities will be measured in terms of reactivity change for various per cent burnups in the core. For the burnup studies, uranium will be removed and poisons yet to be selected added to simulate fission product poisons. Some of the measurements will be repeated for end of life conditions and also for several intermediate life conditions of the core.

Power Distribution Measurements

The power production as a function of position will be measured throughout the core volume and in detail through specific fuel elements. Power distribution measurements will also be made near control cylinders

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in their normal operating position and near cylinders which are out of position with respect to the remaining cylinders. These measurements may also be made for core configurations corresponding to various core lifetimes.

The techniques for making power distribution measurements will be primarily that of measuring fission product activity. This may be done by counting uranium foils which have been activated in the reactor or by counting the fission products on aluminum catcher foils placed next to uranium elements. In making these measurements, other materials may be activated, such as manganese, indium, or gold. These foils as well as the uranium foils may be irradiated bare and cadmium-covered.

Reactivity Measurements

The reactivity worth as a function of position in the reactor will be measured for uranium, stainless steel, aluminum, and possibly materials simulating fission products poisoning. This information may be useful in the physics analysis of the core and as a guide in engineering design changes.

Optimization of Design

Experiments may be made to optimize the core design. Such changes as variations in the fuel element arrangement and in the control cylinders geometry may be investigated. The objectives will be to maintain as uniform a power distribution as possible during the entire operating lifetime and under all operating conditions. This must be accomplished while maintaining sufficient reactivity control. This type of experiment will repeat measurements of the type described above with modified core parameters.

Operational Checks

Certain experiments will be performed to check the operation and safety of the critical assembly itself. Also, tests will be made to determine the reliability of the results obtained with the assembly. Such experiments as measuring the reactivity as a function of table separation, the worth of safety and control rods, the uniformity of fuel elements, and the temperature coefficient of the critical assembly will be made. The uniformity of the fuel elements might possibly be checked by substituting a fuel element designated as a standard for other fuel elements in various locations in the reactor. The temperature coefficient will be estimated by allowing the room temperature to change, thus changing the reactor temperature over a small range. The change in reactivity with all other factors remaining constant will be measured.

Special Characteristics of the Critical Assembly

The characteristics given below are those which have been calculated for the initial critical assembly experiment. During the course of investigations, especially if experiments are done to optimize the design, these quantities may change slightly, but not significantly. Composition of the first core is given in Table 2.

TABLE 2. CRITICAL ASSEMBLY COMPOSITION

Material	Volume Per Cent
<u>Core Region</u>	
Be	47
Al	34
Stainless Steel	15
U	3.5
<u>Reflector Region</u>	
Be	76
Al	7
B ₄ C	1-1/2
Stainless Steel	15

Other calculated parameters for the first core which are important in assessing the hazards associated with the reactor are:

Neutron generation time

$$\lambda = 6.1 \times 10^{-6} \text{ sec clean}$$

$$\lambda = 6.5 \times 10^{-6} \text{ sec end of life}$$

Temperature coefficients

$$\Delta k/T = -2 \times 10^{-8}/\text{deg F due to change in thermal base}$$

$$\Delta k/T = -9 \times 10^{-6}/\text{deg F due to expansion, based on expansion coefficient of stainless steel.}$$

Mean neutron energy for producing fission = 240 ev.

The temperature coefficient due to core expansion was calculated for the power reactor for which these studies are being made. Therefore, the stainless steel expansion coefficient was used. The coefficient for the critical assembly is more likely to be dependent on the aluminum grid tubes expansion and is therefore likely to be larger.

However, these coefficients are not important in normal operation since the reactor is not producing significant amounts of heat. In an accident the temperature rise is likely to be so rapid that there is little heat transfer from the uranium to other materials. In this case only the Doppler coefficient would be important. The Doppler coefficient for U-235 is not known with sufficient accuracy at the present time to be used in calculations with any degree of certainty.

Plan of Normal Operation

The procedures discussed below cover the reactor operations which will be required to perform the experiments described in the previous section.

General Limitations Affecting Operations

A basic limitation on the facility will be a rate of change of reactivity limitation of 6 cents per second. This limitation will apply to all controls which may be operated remotely. Therefore, a safety rod may have the same rate of reactivity change as a control rod, the important difference between the two being how they are instrumented and the accuracy with which they may be positioned. This reactivity limitation will also apply to the table drive which will assemble the reactor halves.

The normal procedure for attaining final criticality which will always be followed is to reach critical by inserting fuel rods and not by moving the tables together. No assembly will be constructed which will go critical if the tables were assembled with the rods in the shutdown position.

The sequence of operations is chosen so that any positive reactivity change can be overcome rapidly by a negative change of greater magnitude.

Assembling Operations

The assembly operations are considered as part of the normal operations since it is anticipated that several different loadings will be required during the course of the planned experiments. Therefore, it will be necessary to adopt procedures and controls which will insure that there are no criticality hazards during the course of construction work.

The various procedures which will be used during the construction phases of the program are described in detail in Supplement Number 1 to the Procedures Manual BMI-PM-607. These procedures will not be

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repeated here in detail. The general features which are being adopted to insure safety are: (1) The amount of uranium-235 which will be handled at any one time will be limited to 350 grams in one batch or 700 grams at one operational location with the exception of the reactor assembly itself. (2) The location of the uranium and the amount assembled at any one point will be carefully controlled by having flow and balance records at each point at which the material is handled. (3) A supervisor will be appointed who will have the responsibility for each procedure being carried out.

In addition to these precautions, the multiplication of the reactor will be measured during the course of its construction. This will be done by measuring the neutron level continuously if the neutron intensity in the vicinity of the core does not present a biological hazard to the personnel. If the neutron intensity is too high, the multiplication will be measured after the addition of each 6 fuel elements and at more frequent intervals if the supervisor or the man making the counts requests it. This information will be continuously plotted and evaluated.

Precritical Operation Check-Out

Prior to each day's reactor operation and more often if the supervisor should request it, a complete check will be made of the instrumentation and safety mechanisms of the reactor. Each radiation detector will be checked for calibration. Each scram circuit will be checked and each scram mechanism will be operated. The interlocks will be checked by attempting operation contrary to their design purpose. Following this a visual check will be made of the facility and the room, the door to the assembly room will be closed and the check-out form will be signed by the person making the check.

Approach to Criticality

Each approach to criticality will follow a set procedure which is described below and which to a large extent is enforced by the interlock system. The initial approach which is described first will be used the first time the reactor is made critical and on any other occasions when significant changes have been made in the reactor or in its instrumentation. Once the critical state of a given reactor configuration has been determined the following approaches to critical will be made with less data taking and plotting.

Initial Approach to Criticality. Each approach to critical will be preceded by the operational check-out described above. Following the check-out, everyone will leave the assembly room and the assembly room door will be closed. The sources will be inserted and approximately half the safety rods

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inserted. The interlock system requires that one set of safety rods in each half of the core be inserted and that one set and the control rods be withdrawn before the table can be moved. The position of the remaining safety rods is left to the option of the operator.

The table is then moved in steps with counts taken on the research channels and readings taken on the indicating safety channels at each step. A typical table closing may require stops and counts being taken at 5-, 4-, 3-, 2-, and 1-foot and at 6-, 3-, 2-, 1-, and 1/2-inch table separation. More points may be taken if the operator thinks it desirable. Counts are taken again with the table in the closed position. Then the safety rods are inserted in steps which are estimated to give 20 cents reactivity change per step with counts and detector levels being recorded at each step. A similar process is followed with the control rods with the steps becoming smaller as criticality is approached.

When the reactor power rises into the operating range, the sources, one or both at the option of the operator, will be withdrawn in steps and the control rod adjusted to keep the power at a nearly constant level. When both sources are fully out and the power remains level the reactor is critical.

During the course of this approach to critical the counting rates and the detector readings will be plotted as the inverse count rate against the position of the table and then against the position of the safety rods. This plot will be extrapolated to zero to predict the critical condition. This extrapolation will be revised after each new count.

Subsequent Approach to Criticality. Once the critical state of the reactor has been determined by the method described above, a simpler means of making the reactor critical can be used. This will change from the initial approach only in that the table and the safety rods will be moved continuously rather than step-wise and counts on the pulse counting channel need not be taken or plotted. However, in these cases the pulse counting channels will be in operation so that the operator will have an additional visual and aural signal of the power level. The progress in this approach to critical will be visible on the two recorders and meters on the console which will be in operation during all the reactor experiments.

Processing and Disposal of Radioactive Materials and Solutions

The only processing of material planned for the Critical Assembly Building is the preparation of small uranium foils. Any other processing

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will be done in the uranium area at the main Battelle Laboratories in Columbus, which are equipped and experienced in all phases of material handling.

Only solid materials are planned for use in this critical assembly program. Thus, there will be no solution disposal problem.

Since all experiments will be done at low power (0.01 to 10 watts) there will be no high level induced activities. The possibility of fission product contamination will be examined by taking periodic air samples and smear tests.

Safety Mechanisms

The proposed critical assembly is not inherently safe against power surges; therefore considerable emphasis has been placed on designing the facility to minimize the possibility of accident either from malfunctioning of equipment or from operator error. The principle features incorporated to make the assembly safe are described below.

Safety Rods

The shutdown available in the safety rods will normally be \$5, but will never be allowed to drop below \$3 in reactivity. These rods will travel 95 per cent of their total distance out of the core in less than 0.2 sec.

The safety rods will normally be attached to 10 independent drives, 5 in each half, with a minimum of 8 drives permitted for operation. These will be tested individually for scram time. They will also be tested for proper operation before each approach to critical which will insure that 80 to 90 per cent of the total rod control will be available for shutdown. With continual checks and maintenance it is unlikely that two failures will occur simultaneously.

The rate of reactivity change for safety rod travel has been estimated at 6 cents/sec. If rods were moved continuously through the critical point it would take over 16 seconds to reach prompt critical which would give the scram circuits more than adequate time to operate if the operator did not take corrective action.

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Control Rods

There will normally be 50 to 70 cents reactivity available in two control rods resulting in a maximum of 1 cent per second reactivity change. If the experiments require it the worths may be increased, but will never exceed the values permitted for safety rods.

The control rods will be interlocked so that they must be out (least critical position) for reactor startup.

The control rod selection will be on the same selector switch used for safety rods; thus only one rod may be driven at a time.

Table Separation

The final closing of the tables will be 1/2 in. per minute, resulting in an estimated reactivity change of 4 cents/sec. The tables will always separate at 40 inches per minute resulting in 80 times as rapid reactivity change for shutting down. The operator will have a choice of scram button or table drive switch for separating the tables, but will have no choice of separating speeds.

Emergency means of separating the tables in case of electric power failure will be provided so that personnel will not have to enter the assembly room with the reactor halves assembled. This may be a battery powered DC motor, an air powered motor, or a weight attached to a rope over a pulley which would unwind from a drum on the table drive shaft.

Electrical Interlocks

Electrical interlocks have been included to prevent the following:

- (1) Entering the assembly room when the reactor is operating or in an operating condition.
- (2) Startup without a neutron source in each reactor half.
- (3) Reaching critical by moving tables together (must go critical with rods, which is a more quickly reversible process).
- (4) Moving tables together without safety rods cocked.
- (5) Moving rods and table at the same time.

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Instrumentation

The safety instrumentation includes four detectors with one level scram originating with each of the four and a period scram from two of the four. All instruments are DC devices, minimizing the possibility of scram failure from overloading.

With the exception of the contact meters originating the scrams, all scram relays are held in by electric power, causing a scram on the receipt of a scram signal or power failure. Each scram circuit has two contact meters in parallel so that a single failure would not prevent a scram from taking place in that circuit. A double failure in a single scram circuit would still leave five other scram circuits operative.

The instrumentation also includes two pulse counting channels to give more accurate data when needed, particularly for reactor startup. One of these channels includes a mechanical register which will give the operator an aural indication of power changes.

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MAKE-UP OF SURROUNDING AREA

The location of the site and a brief description of the geography of the region was given in a previous section. The sections below discuss other features of the area with emphasis on factors which may be related to the operation of the critical facility.

Population Distribution

The proposed site is located about 15 miles west of the center of Columbus, Ohio, in a sparsely populated area. A 10-mile-radius circle enclosing the site includes a small portion of Columbus having a population of about 20,000 people. The only other significant population center near the site is West Jefferson, Ohio, located about 2 miles from the site, with a 1953 population of 1647.

The population distribution at various radii from the site was obtained, for the most part, from the Chamber of Commerce 1954 statistics and the State-Wide Highway Planning Survey, 1953. The distribution of population at various distances from the site is shown in Table 3.

TABLE 3. POPULATION DISTRIBUTION WITHIN VARIOUS DISTANCES FROM SITE

Distance Radius	Population
1500 feet	0
2500 feet	0
1 mile	60
2 miles	1,150
5 miles	5,100
10 miles	43,000

Industry Adjacent to the Site

Within the 10-mile circle are located 12 industries. Ten of these employ less than 100 people while the Westinghouse Electric Corporation and the General Motors Corporation, both 8 miles from the site, employ 2068 and 3931, respectively. Closest to the site are two small industries in West Jefferson which employ less than 100. The industries and information pertinent thereto are tabulated in Table 4.

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TABLE 4. INDUSTRIES WITHIN A 10-MILE RADIUS OF PROPOSED REACTOR FACILITY SITE

Industry	Products	Number of Employees ^(a)	Distance From Site
Terndstedt, Columbus Division, GMC	Auto parts	3931	8 miles
Westinghouse Electric Corporation	Refrigerators, appliances	2068	8 miles
Stokely's Canning Company ^(b)	Canned sweet corn	75-85	2 miles
Hartley Printing and Publishing Company	Printing and publishing	27	10 miles
Columbus General Machine, Inc.	Dies, tools, fixtures	12	10 miles
H. J. Upperman and Sons	Lumber	12	9 miles
Columbus Stationery Company	Stationery	Less than 10	9 miles
Five Manufacturing Company	Farm wagon unloaders	Less than 10	9 miles
Georgiton Candy and Ice Cream Company	Ice cream, candy	Less than 10	10 miles
Merriman Cement Products, Inc.	Cement blocks	Less than 10	2 miles
Stiles Gauge Pin Company	Gauge pins	Less than 10	9 miles
West Jefferson Sand and Gravel Company	Sand and gravel	Less than 10	3 miles

(a) May, 1956.

(b) Seasonal. In full operating in August and September, only. Closed in 1954.

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At a distance of 8 miles from the site is located a new housing project, Lincoln Village. Opened for settlement in July, 1954, the village has a present population of approximately 850. Future expansion is to include several industries; the information regarding these industries has not yet been released.

Seismology

In determining the earthquake probability of a given area, all that can be done is to examine the earthquake history of that vicinity and then conjecture that future earthquakes are more likely to occur in places where there have been previous ones.

Several areas in western Ohio have suffered minor earthquake damage. There is no record of earthquakes having occurred in West Jefferson, Ohio, and immediate vicinity. The nearest seismic activity in recent years was recorded in 1937 in the Anna, Ohio, proximity, which is over 50 miles from the proposed reactor site.

The information on western Ohio earthquakes was obtained from Reverend V. C. Stechschulte, Xavier University, Cincinnati, Ohio, and from the U. S. Coast and Geodetic Survey. Letters from these two sources are included in Appendix C.

Climatology

The climate in the Columbus area is definitely temperate and continental in character. The normal mean daily temperature for June, July, and August is 73.3 F, although 90 degrees or higher is expected about 20 times per year. During the months of December, January, and February, the normal seasonal temperature is 31.2 F with 3 subzero nights per winter average.

The primary prevailing wind direction is from the southerly quadrant (41 per cent of the winds blow from the general direction SE to SW). The secondary prevailing direction is from the NW. The average wind speed is 8.4 mph with 58 per cent of the winds occurring in the 4 to 12-mph interval. No large seasonal or diurnal variation exists in either the direction or speed of the winds. During a 51-year period in Columbus, peak wind speeds have been observed to exceed 51 mph in every month of the year. The maximum recorded speed was 84 mph occurring in July.

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Rainfall, averaging nearly 3-1/2 in. per month, is quite evenly distributed from April to August. The record single day's rain was 3.87 in. in July, 1947. The bulk of the summer rainfall comes in frequent thunderstorms, and tornadoes are not unknown. However, local storm records show that only 4 tornadoes occurred in the Columbus area since 1931.

Since Ohio is located in the path of many winter storms, Columbus receives a generous amount of cold-season precipitation. The bulk of it comes in the form of rain, but the average winter will yield a total of 22 in. of snow. This is quite variable and, in the largest snowfall of recent times, in 1950, 7.5 in. fell in one day.

Climatology data abstracted from a report prepared by the Scientific Division of the U. S. Weather Bureau is given in Appendix B.

Geology and Hydrology

The principal glacial deposits at the surface in the Battelle site area consist of till and outwash which accumulated as the Wisconsin ice sheet of the Pleistocene Age receded. The till, an unstratified matrix of clay containing rock fragments, underlies the Battelle site to depths ranging from approximately 60 to 200 ft. The outwash, composed of stratified layers of sand and gravel, is thin and discontinuous in the site vicinity. Fringing the locality is a narrow strip of Columbus limestone, forming in places a 3-ft surface stratum.

Underlying the glacial deposits of the area are several hundred feet of nearly horizontal beds of limestone, dolomite, and shale through which preglacial streams carved a branched valley system. The distance from the soil surface to the bedrock on the Battelle property ranges from a few feet in areas along Big Darby Creek to over 200 ft in the northwest corner of the property.

There are two aquifers in the Battelle site area. One is shallow and of minor importance underlain by the major aquifer of sand, gravel, and limestone. Yields up to 300 gallons per minute have been obtained from wells drilled into the principal aquifer in the area.

The ground water comes entirely from local precipitation and the shallow aquifer is recharged almost uniformly from the precipitation. The water table is everywhere less than 40 feet from the surface, and the contours are a subdued replica of the surface topography. Calculations indicate that water in the principal aquifer in the vicinity of the Battelle site is moving at a rate somewhat less than 1 ft per day. The water in the till

overlying the principal aquifer is estimated to flow at a considerably lower rate, measurable in hundredths of a foot per day.

Ground-water movement downward through the thick till takes place very slowly. A long period of slow percolation occurs before water reaches a zone in which it may move laterally at appreciable rates. All the ground water is discharged into Darby Creek; hence, water entering the ground on the Battelle property is already near its place of discharge.

Darby Creek accounts for the principal surface-water flow. The mean flow is 420 cubic feet per second, based on a 24-year record. Ground-water seepage from the impermeable deposits in Madison County adds little to stream flow. The water of Darby Creek is of good quality and is not polluted.

The conclusions of a report prepared by the U. S. Geological Survey on the geology and hydrology of the Battelle site are given in Appendix D. It is concluded that in case of liquid spillage most of the liquid would flow overland to the Darby Creek and the remainder, once reaching the water table, would also discharge into the creek. The chances for radioactive contamination of well water in the surrounding area are considered nil.

HAZARDS DURING NORMAL OPERATION

The potential hazards of normal operation are not great. These hazards are the possibility of airborne fission products, since the reactor fuel will be uranium metal strips, and the hazard due to direct radiation through the assembly room walls. The fast neutron intensity outside of the assembly room will be particularly important, since this reactor has an intermediate energy spectrum with a high uranium loading and a high leakage. However, this high neutron leakage will allow operations at extremely low power, since it is the leakage neutrons which are detected in the neutron sensitive detectors.

Fission Product Hazard

It is estimated that less than 0.3 per cent of the fission products will escape from the uranium fuel. This estimate was arrived at in the following manner. The uranium fuel will be coated with a Telfon-Type coating 0.0002 in. thick, which is approximately 1.5 milligrams/cm². If the fission products are assumed to have an average range of 3.2 milligrams/cm², the depth from which fission products may escape from uranium will then be the difference in these two values, or 1.7 milligrams/cm². This density

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corresponds to approximately 0.04 mil of uranium. Therefore, of the fission products traveling perpendicular to the surface, only 1 per cent escape from the 4,0-mil-thick foil. Since fission products will be traveling in all directions, an estimate could be made of the number escaping by an integration process. However, an estimate sufficiently accurate for these purposes can be made by considering that there are 6 coordinate directions in which fission products can go and only one of these is out from the surface. Therefore, only 1/6 of those fission products originating close enough to the surface to escape will actually escape. This factor must then be multiplied by 2 since there are two surfaces to each uranium strip. These three factors multiplied together lead to 0.3 per cent total fission product escape.

The uranium strips will be tightly packed with stainless steel and aluminum strips in stainless steel boxes. There will be no air flow over these strips. Therefore, any fission products escaping from the uranium strips will have a high probability of being captured in the stainless steel or aluminum strips. Only a very insignificant number will be stopped in the air and thereby cause a contamination problem.

When fuel bundles are torn down for storage or rebuilding, smear test will be taken to determine if there is any contamination which can be picked up and transferred.

Direct Radiation Hazard

Because of the nature of this reactor, it is expected that the fast neutrons will be the most hazardous radiation to personnel. A very conservative estimate of the fast neutron intensity outside of the assembly room was made using an attenuation factor through concrete for fast neutrons, and assuming that neutrons are either absorbed in the uranium or lost by leakage from the core. This assumption is very conservative in that neutrons will be absorbed in the other materials in the core, and will be degraded in energy in passing through the beryllium reflector.

This conservative calculation indicates that one watt continuous operation on a 40-hour week basis will lead to only 1/4 the permitted fast neutron tolerance in the hallway outside the assembly room. At all other locations in the building, except for the vault, the neutron level will be lower. In the vault, the neutron level will be slightly higher, but this is to a point that will not be continuously occupied.

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The hazard from gamma radiation from the reactor was also calculated. This calculation indicates that, on the basis of continuous operation on a 40-hour week, the gamma ray dose would be approximately 1/8 of the allowable dosage.

The above figures indicate that one watt continuous operation is a conservative power level. This will be a much higher power level than required for most reactor operations. However, for activating some materials from 1 to 10 watts may be required for short periods of time. Such irradiation times will be a very small fraction of the total available time and such experiments are a small part of the total program.

Radiation surveys about the building, both inside and out, will be made during the early stages of the reactor operation.

People will be excluded by a fence from the area to the rear of the building where the truck entrance door provides practically no shielding against radiation.

HAZARDS AFTER AN ACCIDENT

Of major concern here are the hazards created in the area following a reactor accident. In most azimuthal directions, the exclusion radius is considerably more than 1500 ft. The minimum distance to the fence is 1200 ft and a swamp is on the other side of this fence. Thus, an actual vaporization and release of the fuel elements must occur to endanger public persons. Possible danger to the population might result from (a) direct irradiation by a cloud of fission product gases, (b) direct irradiation by contaminated ground surfaces due to a fall-out or rain-out from the cloud, and (c) inhalation of radioactive particles or gases and other toxic materials.

Radiation From Radioactive Cloud

To evaluate the cloud problem, the nomographs constructed by J. Z. Holland were employed. The meteorological data were prepared by the Scientific Services Division of the U. S. Weather Bureau and are shown in Table 5.

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TABLE 5. PARAMETERS FOR DOSAGE CALCULATIONS(a)

h, meters	n	c ²		
		u = 1 m/sec	u = 5 m/sec	u = 10 m/sec
25	1/5	.064	.046	.040
	1/4	.021	.014	.012
	1/3	.009	.005	.004
	1/2	.006	.002	.002
50	1/4	.015	.010	.008
100	1/4	.008	.005	.004

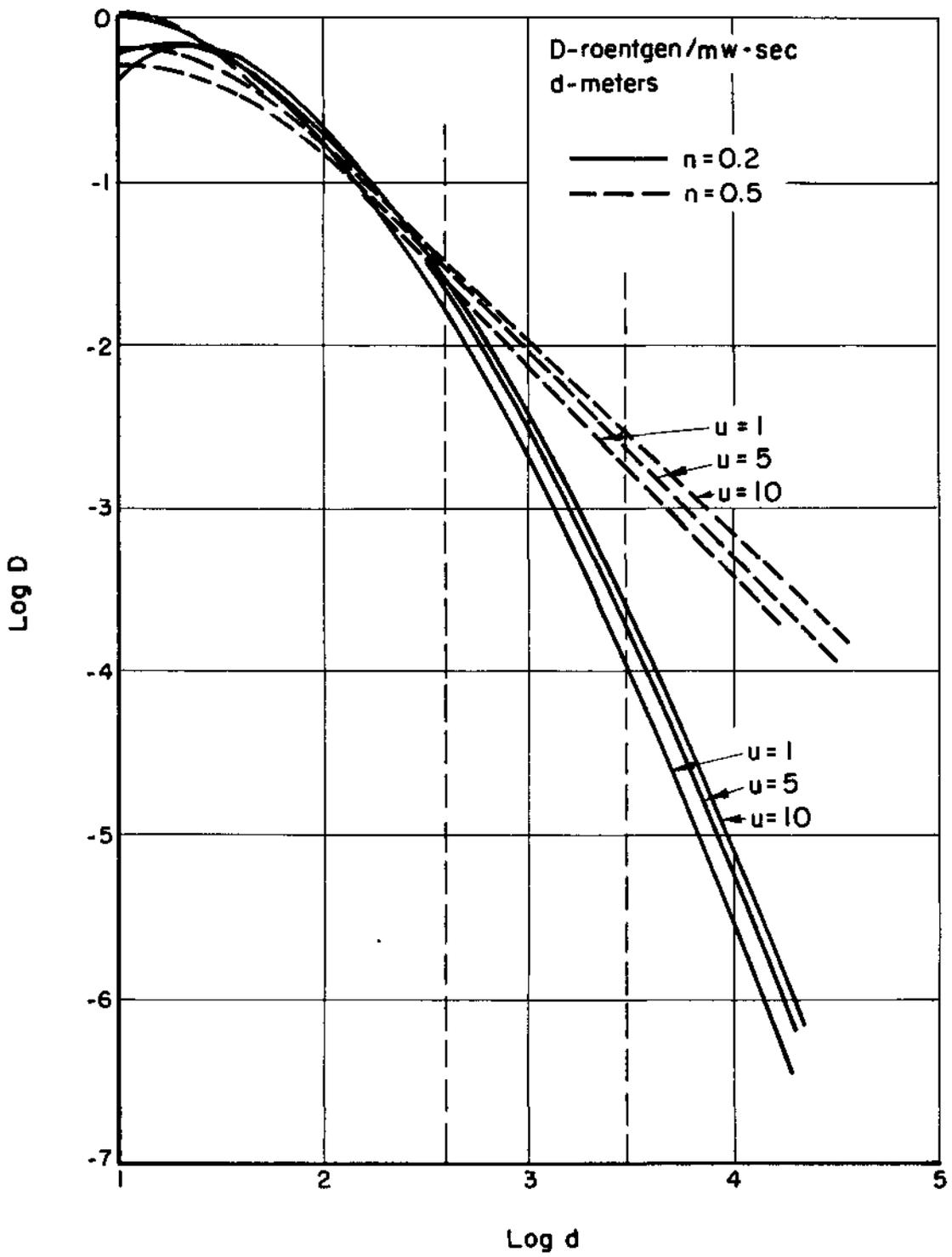
(a) h = height of cloud from ground, m
 u = mean wind speed, m/sec $\frac{n}{u}$
 c = virtual diffusion coefficient, m²
 n = Sutton stability index, dimensionless.

The stability parameter, n, may vary from 0 to 1 such as

Large lapse	1/5
Zero or small lapse	1/4
Modified inversion	1/3
Large inversion	1/2

Using these data and the nomographs, the total radiation dose was determined for various distances downwind from the reactor for a megawatt-second fission energy release. It was estimated that the initial cloud height and cloud radius would be 25 meters and 8 meters, respectively. The results are shown graphically in Figure 15 for several meteorological conditions.

From Figure 15 it may be noted that at the exclusion area limit the dose is approximately $0.03 \frac{r}{(Mw)(sec)}$. To attain a lethal dose at this location, i. e., one expected to kill about one-half of the persons exposed to it (~400 r), would require an accident characterized by a 1.3×10^4 MW-sec fission energy release. It is shown in a later part of the report that even for a very severe accident the fission energy release does not approach this value.



(d = Distance Downwind From Cloud Origin) A-18546

FIGURE 15. DOSES DUE TO PASSING RADIOACTIVE CLOUD

Radiation Due to Fall-Out or Rain-Out
From the Radioactive Cloud

In this case, the fission products contained in the cloud following an accident are assumed to fall to the ground as the cloud travels downwind. The dose one meter above ground level after the fall-out was calculated as a function of distance downwind. The equations used are reported in Appendix D. The doses calculated are shown in Figure 16.

From Figure 16 it may be noted that at the exclusion area limit the integrated dose is approximately 2.5 r/Mw-sec and at the nearest village the dose is approximately 0.1 r/Mw-sec. It should be emphasized that these doses are based upon continuous long term exposure following fall-out. If a shorter more practical length exposure is considered, the dose is much less. For instance, in the first hour of exposure at the nearest village, the dose is only 0.028 r/Mw-sec.

Inhalation of Radioactive Material

The equations used for calculating the inhalation activity are given in Appendix D. The results are shown graphically in Figures 17 and 18.

The maximum permissible amounts of the various isotopes of major concern are given in Table 6.

TABLE 6. MAXIMUM PERMISSIBLE AMOUNTS OF SOME ISOTOPES

Isotope	Maximum Permissible Amounts in Total Body, micro-curies
I ¹³¹	0.3
Sr ⁸⁹	2.0
Sr ⁹⁰ + Y ⁹⁰	1.0
Ba ¹⁴⁰ + La ¹⁴⁰	5.0
Ce ¹⁴⁴ + Pr ¹⁴⁴	5.0
Y ⁹¹	15.0

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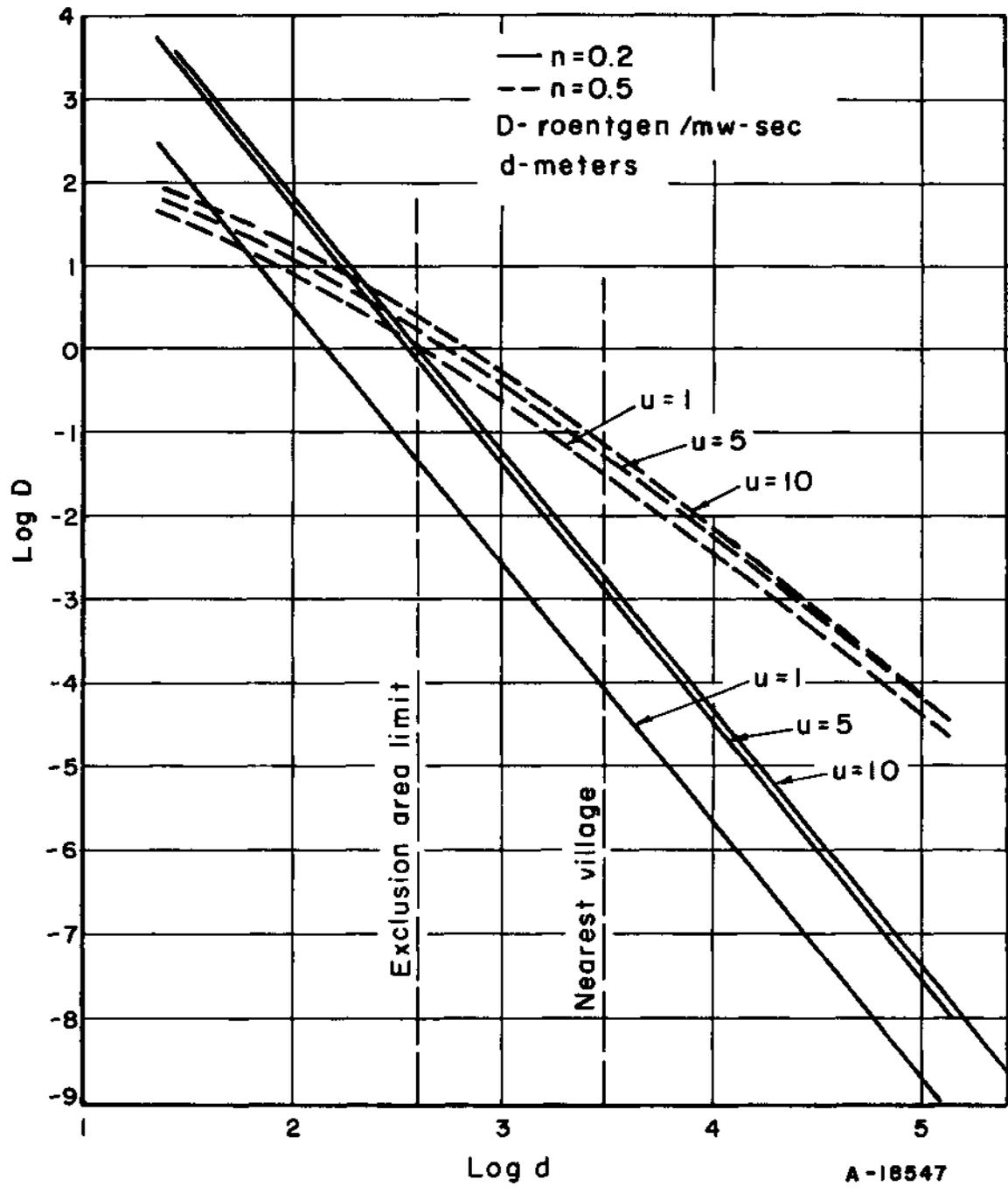


FIGURE 16. DOSES DUE TO FALL-OUT FROM PASSING RADIOACTIVE CLOUD

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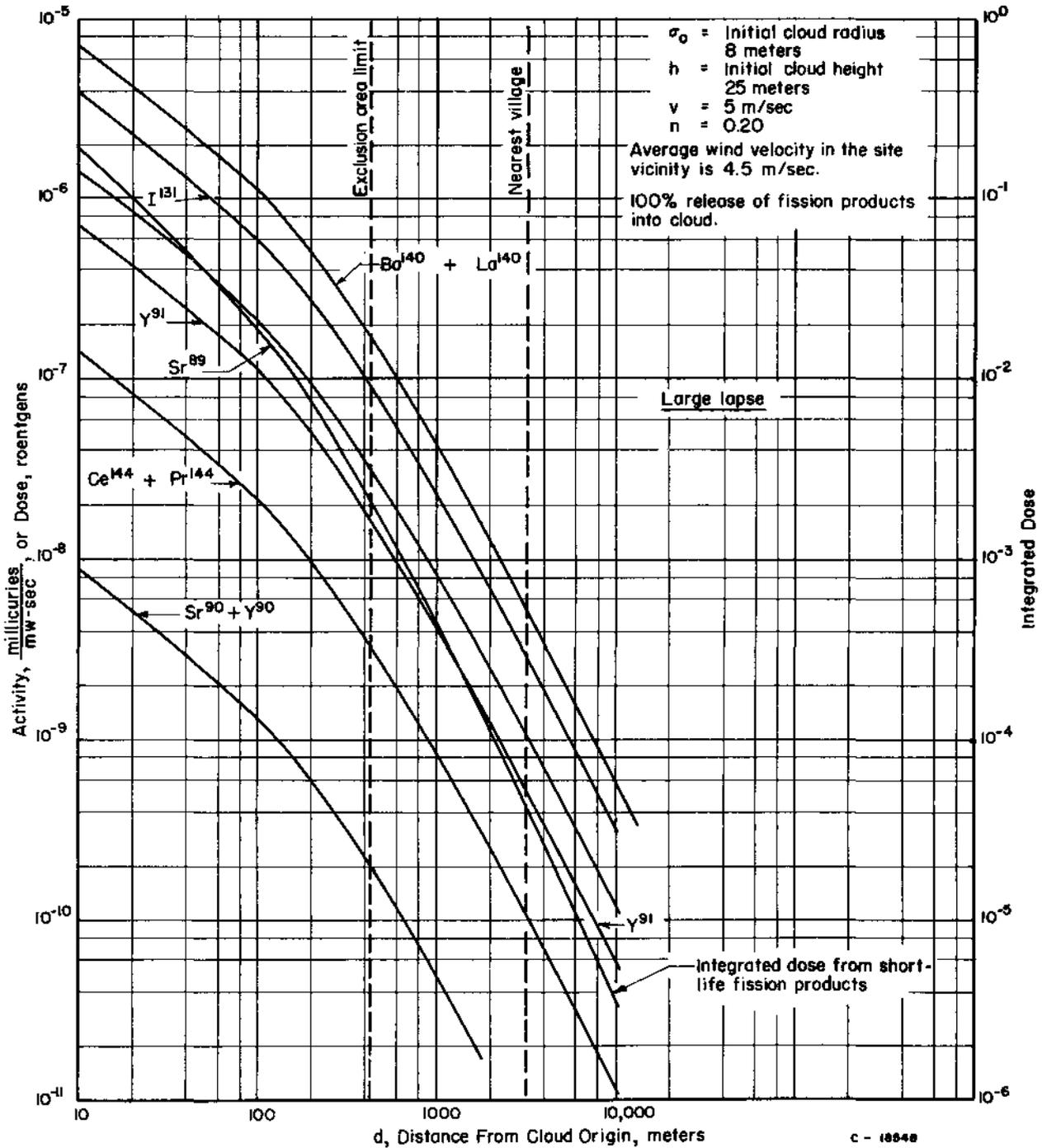


FIGURE 17. INHALATION ACTIVITIES FROM RADIOACTIVE CLOUD

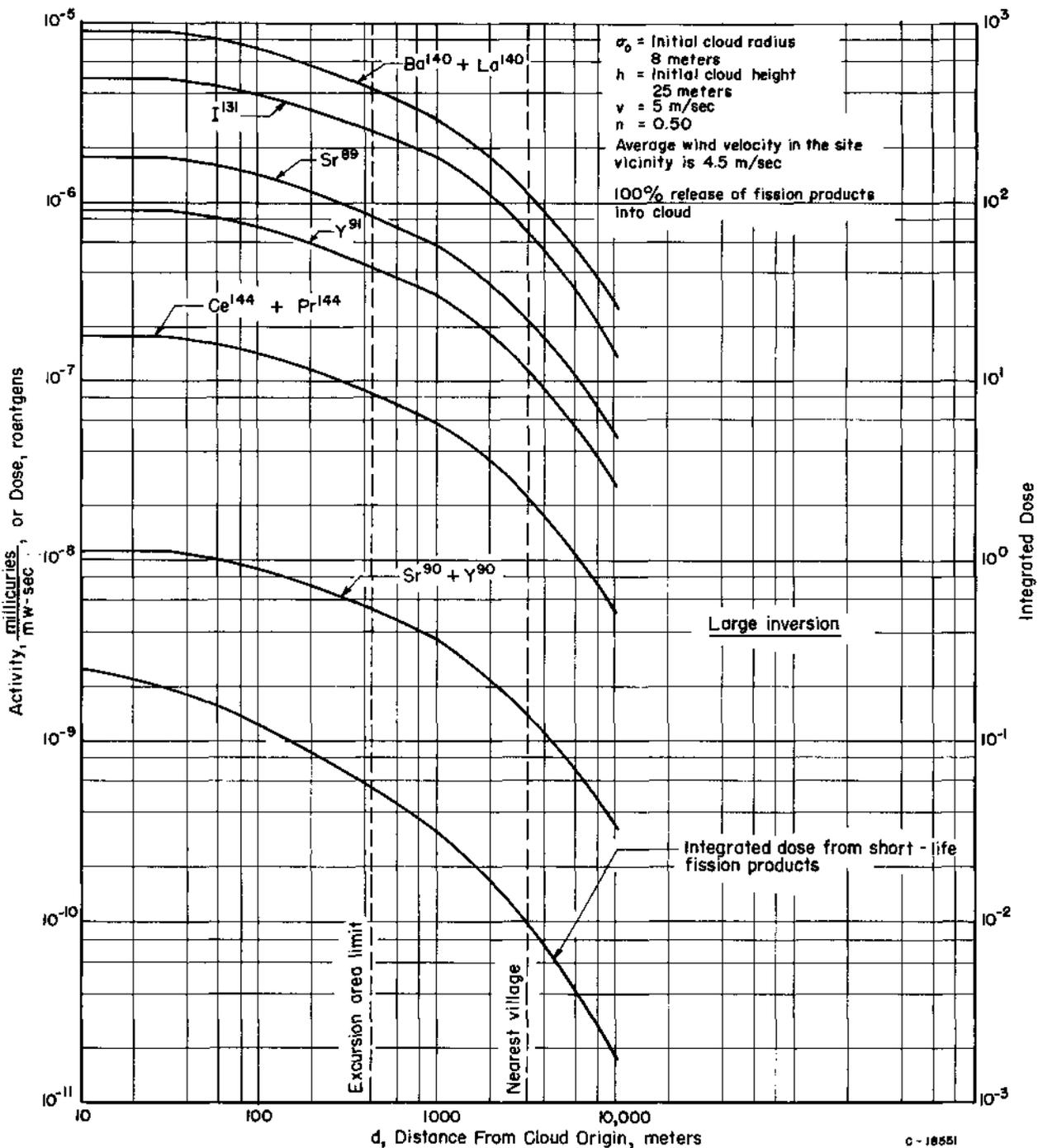


FIGURE 18. INHALATION ACTIVITIES FROM RADIOACTIVE CLOUD

The most critical isotope is I^{131} . From Figure 18 the activity of this isotope for the worst diffusion conditions is 2.5×10^{-3} μ curies/Mw-sec at the exclusion area limit and 7×10^{-5} μ curies/Mw-sec at the nearest village. Thus an accident of greater than 100 Mw-sec is necessary to exceed the permissible tolerance value at the exclusion area limit and an accident greater than 4000 Mw-sec is necessary to exceed the permissible tolerance value at the nearest village.

Beryllium Hazard

The hazards resulting from an exposure to beryllium are difficult to evaluate. This point is shown by some of the material below, which is quoted from a paper titled "Epidemiology of Beryllium Intoxication" by J. H. Sterner and M. Eisenbud, published in the A.M.A. Archives of Industrial Hygiene and Occupational Medicine, Vol. 4, pp 123-151.

"One episode for which air-analyses data are available indicates that a 20-min. inhalation of beryllium fluoride in concentrations which ranged up to 400 to 650 micrograms of beryllium per cubic meter of air produced acute disease in three individuals. . . . The total amount of beryllium inhaled during this brief episode approximated 45 micrograms. "

"The occurrence of nonoccupational beryllium poisoning in the vicinity of a plant producing beryllium compounds from ore has afforded a unique opportunity to conduct environmental studies. . . . in the seven years during which this plant was active. The measured concentrations of beryllium in the neighborhood of this plant were in excellent agreement with the concentrations which were predicted from knowledge of the rate at which beryllium was being emitted and the theory of turbulent diffusion of stack gases. . . . Within 1/4 mile (4000 m.) of the plant, where the concentrations of beryllium were of the order of 1 microgram per cubic meter, the incidence of berylliosis was approximately 1 per cent. Within the plant the employees were exposed to milligram concentrations of beryllium but only six cases of berylliosis have developed out of the total employment of about 1,700 persons. . . . It will be noted that the berylliosis occurred in employees whose length of service was less than four months. One ordinarily expects to find chronic occupational disease among those whose histories indicate the longest exposures. The reverse seems true in this plant. . . . the principal constituent of the effluent from this plant was known to be beryllium oxide, and this compound was therefore the dominant contributor to the neighborhood exposure. " "In one of the plants. . . . there have been no chronic cases in the 10 years during which this plant has operated. . . . The total number of persons in the. . . . plant did not exceed 200 and the absence of chronic cases is therefore not surprising in view of the low attack rate in the second plant. " (The second plant is the one with 1700 employees.)

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From the data on the acute cases given above it is possible to arrive at an approximate relation for the amount of beryllium inhaled in a given circumstance. This is

$$\text{Dose inhaled} = \text{Concentration} \times \text{Time} \times \text{Constant}$$

where the constant has been empirically determined to be 0.0043. If this relation is applied to the nonoccupational cases described above, and it is assumed that the average individual involved spent half of his life at his home, the dose inhaled turns out to be 1100 micrograms of beryllium. This dose resulted in a 1 per cent incidence of berylliosis.

To investigate the hazard in the present case, 10 per cent of the beryllium (amounting to 3.5×10^4 g) is assumed to be released at the critical assembly building in a cloud 8 m in radius at a height of 25 m. Using the nomograms again, the results shown in Figure 19 were obtained. These results indicate that for such a release the beryllium dose at the nearest boundary of the Battelle property is only 700 micrograms. This is less than the 1100 micrograms found to produce sickness in 1 per cent of the people.

POSSIBLE ACCIDENTS

Experimental procedures used in these criticality studies predict the beginning of any hazardous condition and permit the operator to take corrective action. Even if the operator makes a mistake in the procedure or part of the equipment fails to function properly, one or more of the many safety mechanisms will control the situation and prevent a hazardous situation from developing. However, in considering the hazards associated with this experiment several extreme accidents and the resulting reactivities have been investigated. Three of these cases are presented in the following sections. The first of these is a highly improbable if not impossible accident which is postulated to determine the maximum conceivable reactivity addition. The second accident deals with the maximum rate of reactivity addition. This was determined to be an accidental closure of the table halves by the separating motor. This motor has a much higher speed than the motors normally used for closure. Finally, an accident involving simultaneous maloperation of the safety rods is considered. The calculated energy releases for various cases are also included.

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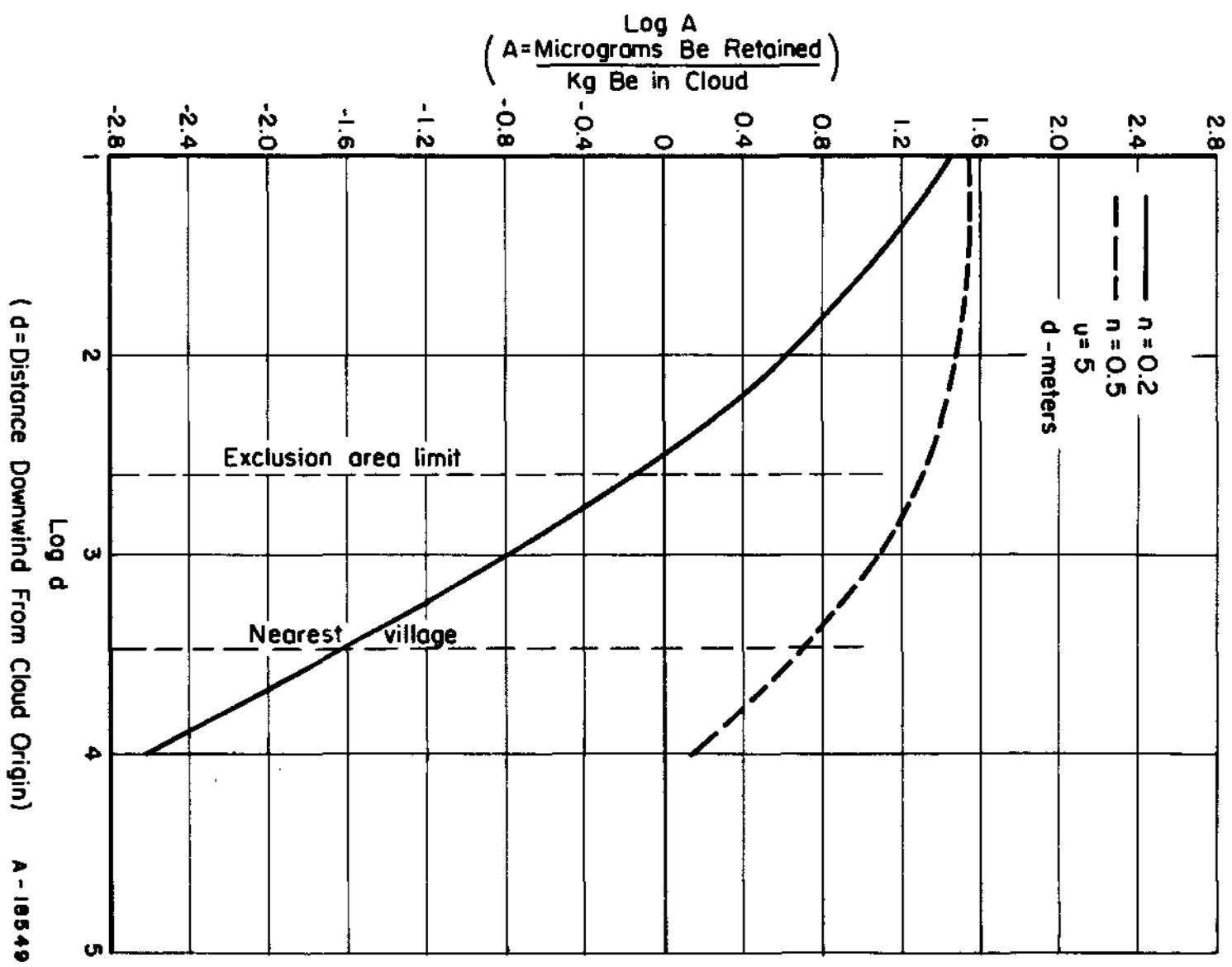


FIGURE 19. BERYLLIUM DOSES DUE TO PASSING CLOUD A-18549

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Maximum Reactivity Possible

It is postulated that in the process of determining the shutdown of the reactor and the worth of the control cylinders, the critical mass is determined with the simulated control cylinders turned full in (maximum poison). Then the loading and the control cylinder positions are modified to give initial conditions for the proposed power reactor. These modifications could continue with the addition of poisons to simulate fission products, until the end of life condition with peak xenon is mocked up. If then additional information were required at the first reactor condition, simulated control cylinders turned full in, it is possible that the crew could remove the poison and add the fuel to arrive at the initial loading, without rotating the control cylinders. Such a sequence of mistakes would leave the reactor with a potential 13 per cent excess reactivity.

Even with 13 per cent excess reactivity, the normal operating procedures would warn of the condition and if the normal procedures were disregarded the safety mechanisms would prevent the reactor from being assembled. However, if all the safety mechanisms were inoperative, an accident would disperse the core before all the 13 per cent reactivity could be introduced. For all 13 per cent to be added the core must be assembled at about 250 ft/sec. which appears quite impossible. This then is taken as the maximum hypothetical accident, i. e., 13 per cent excess reactivity added instantaneously. The energy release is reported in a later section.

Maximum Rate of Reactivity Addition Upon Closure

To arrive at the maximum rate at which reactivity could conceivably be added to the assembly, it was assumed that an accident occurs on the first approach to critical after a change in core configuration has been made. Furthermore, this change has left the core with a greater reactivity than can be controlled by the safety rods. If the operator begins to close the tables and does not follow the power build-up as the core halves are assembled, a scram will occur. However, this negative reactivity has been assumed insufficient to shut down the core. It is now assumed that the motor controller, which prevents the 3 phase motors from operating if one phase of the line fails, becomes inoperative at the moment of the scram. Then the safety motor which is idling in the direction of assembling the reactor will continue to run on single phase power and the reactor halves will close at 40 inches per minute, introducing an estimated \$3.20 per second rate of reactivity change.

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Maximum Rate of Reactivity Addition by Simultaneous Safety Rod Operation

In making subcritical multiplication measurements, often better results are obtained if core symmetry is maintained. Such an experiment may result in a condition where all of the safety rods are at an intermediate position with the reactor subcritical. Then if, through malfunctioning of the controls, all safety rods were to be inserted simultaneously, the reactivity would increase at an estimated rate of \$0.60 per second.

Beryllium Vaporization

In the cases of reactor accidents which have been postulated, there is a probability that some beryllium or its oxide would be released to the atmosphere. Although the course of a possible accident would be so rapid that there would be little heat transfer to the beryllium (all uranium is in stainless steel boxes with no bonding between any of the materials), the temperatures reached in the core would be very high, and some beryllium is likely to oxidize or vaporize. "The Reactor Handbook", Vol. 3, Section 1, p 87, gives some reaction rates of beryllium with air. At 1950 F a beryllium specimen gained weight at the rate of 0.00016 gram per cm² the first half hour and 0.00067 gram per cm² in the first two hours. The total oxidation of the specimen in a half hour, if occurring at the same rate on the 3.5×10^5 grams of beryllium in the critical assembly core, would result in the oxidation of approximately 20 grams of beryllium. Figure 19 shows that a release of this much beryllium would produce an insignificant concentration at the nearest boundary of the exclusion area.

Energy Releases

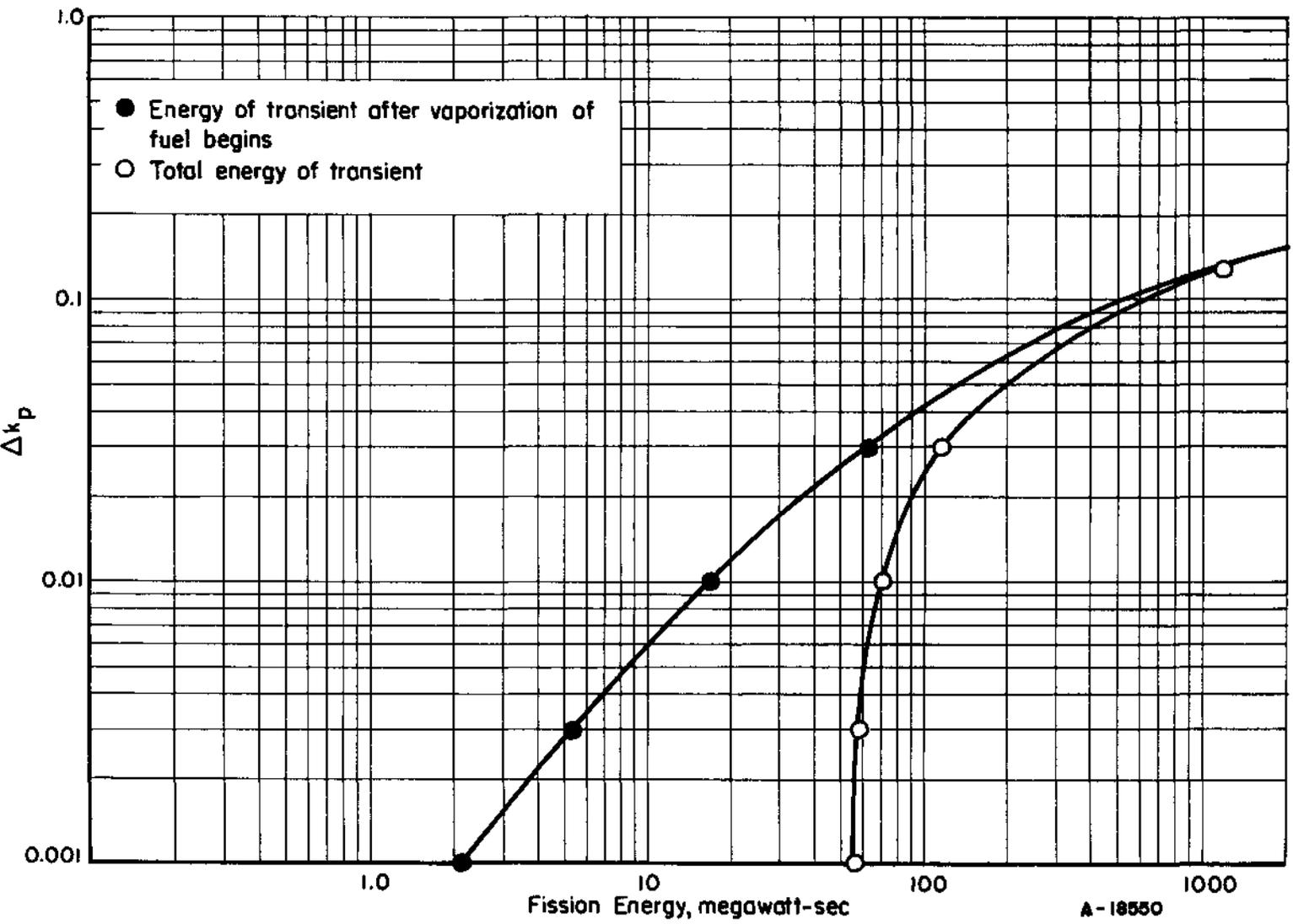
An analysis was made of the fission energy releases associated with various instantaneous additions of reactivity. The shutdown mechanism assumed was core disassembly due to internal pressure from vaporization of fuel. The assumptions and equations employed in this analysis are reported in Appendix E. The results of this analysis are shown in Figure 20.

In connection with the maximum conceivable reactivity addition of 13 per cent ($\Delta k_p \sim 0.14$), it is seen from Figure 20 that the maximum possible release of fission energy is 1.5×10^3 Mw-sec.

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FIGURE 20. FISSION ENERGY ASSOCIATED WITH TRANSIENTS FOLLOWING INSTANTANEOUS REACTIVITY ADDITIONS



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It is shown in Appendix E that the maximum rate of reactivity addition of \$3.20/sec corresponds to a Δk_p of 0.00210 added instantaneously. From Figure 20 it is noted that this corresponds to a fission energy release of 58 Mw-sec.

The relationship between these energies and the various doses can be arrived at by using Figures 15, 16, 17, and 18. These doses are discussed in the next section.

DISCUSSION

The results of the hazards calculation show that the exclusion area is adequate to reduce the dose to a reasonable value even for the maximum hypothetical accident. That is, for the energy release of 1.5×10^3 Mw-sec, the dose from the radioactive cloud is 47 r at the exclusion area limit and is only 4.7 r at the nearest village. The results of the fall-out calculations give doses of 3.7×10^3 r and 1.5×10^2 r for this hypothetical case at the exclusion area limit and the nearest village, respectively. This is the integrated dose for an infinite time.

However, the instantaneous addition of 13 per cent reactivity is certainly in the realm of the impossible. To attain this amount would require closure of the critical assembly halves with a velocity of about 250 ft/sec. Also, the fall-out rates considered were such that maximum doses were obtained at each distance. Thus, the fall-out doses are extreme over-estimates.

The results of the inhalation analysis indicate that inhalation is even less serious than the other hazards considered.

The beryllium doses even in the exaggerated case considered are less than that required to make 1 per cent of people exposed ill.

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APPENDIX A

METEOROLOGY REPORT

Introduction

The purpose of this report is to review the meteorology of the Columbus, Ohio, area for use in the site evaluation and compilation of a hazards analysis.

Source of Data

Although no meteorological data exist for the proposed site itself, very complete meteorological records have been taken for many years at several locations in Columbus. There does not appear to be any significant difference in the general topography of the area between the site and Columbus, so, for most purposes, the meteorological data which have been previously collected should be adequate for this preliminary evaluation. The Local Climatological Data for Columbus, Ohio⁽¹⁾ presents a good general description of the weather of this area. The table headed Normals, Means and Extremes on page 2 of this annual summary presents average data on temperature, degree days, precipitation, snow, humidity, wind, etc.

Climatological Review

In the present brief report, those meteorological parameters will be stressed which influence directly the spread of atmospheric wastes.

Surface Wind Direction

The hourly wind observations for a 6-year period, 1948-1953, for the Weather Bureau Airport Station (WBAS) at Columbus were studied in detail. Table I presents the percentage frequency of the wind direction. The prevailing wind direction is from the southerly quadrant (41 per cent of winds blow from the general direction SE through SW). The secondary prevailing direction is from the NW. There does not appear to be any major change in

(1) U.S. Department of Commerce, Weather Bureau, "Local Climatological Data", for 1953, Columbus, Ohio, U.S. Government Printing Office, price 10 cents.

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wind direction frequency from season to season except that northwesterly winds predominate in the Spring, while the south and southeast winds reach their maximum frequency in the fall. Table I also compares the wind frequencies for two periods of the day - 8 a. m. to 4 p. m. and 5 p. m. to 7 a. m. From a study of these data, it is clear that no large diurnal change in the wind direction should be expected on the average, although night-time conditions favor southeasterly directions and calms, whereas the prevailing daytime wind is south-southwest.

It is necessary to examine the wind structure during period of precipitation in order to consider the effect of washout of possible waste contaminants. Table I also presents the percentage frequency of wind directions at Columbus during those hours when precipitation was falling. (This was approximately 15 per cent of the time.) In this case, also, there does not appear to be any major shift in the prevailing wind direction frequencies, although NW is the primary maximum in this case.

Wind direction is also important when the lower atmosphere is very stable and atmospheric diffusion is at a minimum. From other meteorological studies of this correlation (Cincinnati, Dayton, and Detroit), it seems probable that the most stable weather in the Columbus area would accompany the southerly and southeasterly winds. The northwesterly winds would be associated with unstable or good diffusion atmospheric conditions. This tendency is borne out by the seasonal and diurnal variations, spring and daytime being the periods in which low-level instability is most common.

Surface Wind Speed

Table II presents the percentage frequency of wind speeds in various class intervals. There is a striking persistency to the distribution. Approximately 58 per cent of the winds in the Columbus area will occur in the 4-12 mph speed interval. The average speed is 8.4 mph, although it is slightly weaker in the summer months and stronger in the winter. Winds less than 4 mph occur approximately 21 per cent of the time on the average (10 per cent during the day, 28 per cent during the night, and 10 per cent during those hours when precipitation is occurring). During a 51-year period in Columbus, peak wind speeds have been observed to exceed 51 mph during every month of the year. The highest recorded speed was 84 mph during July.

Two very localized types of storms which are accompanied with high wind speeds deserve special mention - thunderstorms and tornadoes. Thunderstorms occur on the average of 41 days per year primarily in the late spring and summer, although they have occurred during every month of the year. The peak activity is in June and July. These months average 8 thunderstorm days apiece. Thunderstorm activity is extremely variable,

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but a rare severe storm may cause winds in excess of 50 mph, 1 to 3 inches of rain in an hour, and hailstones 1/2 inch or larger in diameter. Situations favorable for the formation of severe thunderstorms are also conducive to tornado formation. However, this more violent storm is rare in the Columbus area. A 35-year study of United States tornadoes shows that 111 tornadoes occurred in Ohio during this period with the largest percentage of these storms in the northern and western portions of the state. Local storm records from 1931 through 1954 show only 4 tornadoes in the immediate Columbus area.

Precipitation

The Columbus area receives approximately 38 inches of precipitation annually which will be spread over approximately 140 days. Precipitation is distributed rather evenly throughout the year with the maximum occurring in the late spring and early summer. The maximum amount of precipitation ever observed in 24 hours was 3.91 inches. Columbus has an average snowfall of 22 inches which falls on approximately 6 days per year. The greatest amount ever recorded for a 24-hour period was 11.9 inches and for one month was 29.2 inches.

Atmospheric Stability

Measurements of the vertical temperature distribution are not made in the Columbus area. However, measurements made at other locations have shown a high degree of correlation between low winds periods, restricted visibility and the occurrence of inversions. Conversely high wind speeds and good visibility are indicative of lapse conditions and good diffusion weather. The Columbus area experiences approximately 15 days on which heavy fog occurs for a few hours. Visibility is reduced to below 6 miles approximately 43 per cent of the hours annually. For just fog, it is reduced to below 6 miles approximately 8 per cent of the time.

Inversions form nearly every night, but there is nothing in the records which could be interpreted to signify that the Columbus area experiences an unusual amount of poor atmospheric stability conditions.

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TABLE I

Percentage Frequency of Wind Direction, Weather Bureau Airport Station, Columbus, Ohio
(Based on hourly observations January 1943 through December 1953)

	Winter	Spring	Summer	Fall	Annual (All obs.)	Annual (8 a.m.-4 p.m.)	Annual (5 p.m.-7 a.m.)	Annual (only when precipi- tation was occurring)
N	3.9	5.7	6.9	4.5	5.3	4.5	5.8	4.3
NNE	2.4	3.8	4.0	3.0	3.3	3.0	3.5	2.7
NE	3.2	3.7	4.7	2.9	3.6	3.1	4.0	2.9
ENE	2.9	3.4	3.0	2.6	3.0	2.6	3.1	2.8
E	3.8	4.9	4.9	4.2	4.5	3.4	5.1	3.3
ESE	4.2	4.0	3.4	3.6	3.8	2.5	4.6	4.2
SE	9.0	8.3	9.9	11.0	9.5	7.0	11.1	8.9
SSE	8.4	5.2	7.5	8.7	7.4	6.5	8.0	6.9
S	10.8	7.6	9.2	10.5	9.5	10.8	8.7	8.4
SSW	9.6	7.4	7.0	8.2	8.0	11.4	6.0	9.6
SW	6.8	7.4	6.6	7.0	7.0	10.6	4.7	7.6
WSW	4.7	5.0	2.8	3.5	4.0	5.1	3.4	5.4
W	4.5	4.6	1.8	2.8	3.4	4.1	3.0	4.8

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TABLE I (Continued)

	Winter	Spring	Summer	Fall	Annual (All obs.)	Annual (8 a.m.-4 p.m.)	Annual (5 p.m.-7 a.m.)	Annual (only when precipi- tation was occurring)
WNW	7.5	7.1	3.8	5.2	5.9	6.7	5.5	8.6
NW	9.8	10.9	8.3	9.1	9.6	9.8	9.4	11.2
NNW	6.1	7.6	8.0	7.4	7.3	7.4	7.3	6.8
Calm	2.4	3.4	8.1	5.7	4.9	1.6	6.9	1.3

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TABLE II

Percentage Frequency of Wind Speed Groups (mph), Weather Bureau Airport Station, Columbus, Ohio
(Based on hourly observations January 1948 through December 1953)

	Winter	Spring	Summer	Fall	Annual (All obs.)	Annual (8 a. m. -4 p. m.)	Annual (5 p. m. -7 a. m.)	Annual (only when precipi- tation was occurring)
Calm	2.4	3.4	8.1	5.7	4.9	1.6	6.9	1.3
1-3	12.0	11.9	23.4	17.7	16.2	8.9	20.6	8.5
4-12	57.9	56.2	59.0	57.6	57.6	57.2	57.9	55.0
13-24	27.0	27.8	9.8	18.8	20.8	31.5	14.3	34.2
25-31	.7	1.0	*	*	0.5	0.7	0.4	0.9
32-46	*	*	*	*	*	*	*	*
46	-	-	-	*	*	-	*	*
Mean Wind Speed (mph)	9.7	9.7	6.4	7.9	8.4	10.5	7.1	10.8

* Only a few observations.

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APPENDIX B

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DEPARTMENT OF COMMERCE
U. S. COAST AND GEODETIC SURVEY
WASHINGTON 25

August 24, 1954

Mr. James N. Anno, Jr.
Battelle Memorial Institute
505 King Avenue
Columbus 1, Ohio

Dear Sir:

Replying to your request of August 19, 1954, we find no record of earthquakes having occurred in West Jefferson, Ohio, and immediate vicinity. However, as Rev. V. C. Stechschulte stated, there have been several minor earthquakes in western Ohio. Among the most noteworthy are the following which are briefly described in SP 609, Part 1, Earthquake History of the United States.

- 1776 Muskingum River, Ohio
- 1875 Urbana & Sidney
- 1884 Columbus
- 1901 Wellston & Portsmouth
- 1909 Ohio Valley (38.7 N. & 86.5 W.)
- 1929 Bellefontaine
- 1931 Anna
- 1937 Anna & Sidney (Mar. 2 and 7)

If we may be of further service please do not hesitate to write again.

Very truly yours,

/s/ Robert W. Knox

Acting Director

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XAVIER UNIVERSITY
CINCINNATI 7, OHIO

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August 18, 1954

Mr. Jim Anno
Battelle Memorial Institute
Columbus, Ohio

Dear Mr. Anno:

This is in reply to your telephoned request of yesterday afternoon.

The problem of determining seismic risk in a given area is largely a guessing game. All that can be done is to list the earthquakes, with the damage done, that have occurred there within the comparatively few years of our historical record, and then to say that where earthquakes have occurred in the past, they may more likely occur again in the future rather than in places where there has been no seismic record.

The catalogs that would be pertinent to your purpose would be:

Serial 609, Earthquake History of the U. S. , Part I (pp 39-46).
Serial No. 511, United States Earthquakes, 1929 (p. 8)
Serial No. 553, " " " , 1931 (p. 7)
Serial No. 619, " " " , 1937 (p. 8, 9).

These will give you more detail than is indicated by the maps, listing places where the earthquakes were felt and where damage may have been reported. What it will all add up to is that there has been minor damage approximately within the 50-mile circle with more severe damage in a small area around Anna, Ohio. The small earthquake in the vicinity of Zanesville two or three years ago would make no significant change in the picture.

Sincerely yours,

V. C. Stechschulte, S. J.

(Rev.) V. C. Stechschulte, S. J.
Director of the Seismological
Observatory

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APPENDIX C

CONCLUSIONS OF GEOLOGY AND HYDROLOGY REPORT

The conclusions of a report* on the geology and hydrology of the Battelle site are reprinted here. This report appears in full in a previous Hazards Report on the critical assembly. **

"The Battelle site seems to be almost entirely safe for the operation of a nuclear power reactor, with respect to the effects on the ground water resources resulting from accidental spillage of radioactive fluids in the site area. In the event of a spill, most of the liquid would flow overland to Darby Creek. Only a small portion would infiltrate the soil and seep downward to the water table or reach the principal artesian aquifer.

"Infiltration would be greatest when the soil is dry, especially during periods of large soil moisture deficiency, such as would occur during the growing season. Conditions least favorable for infiltration, and those which would promote most rapid runoff, would result from frozen or saturated ground, or would occur during a heavy rainstorm when a large volume of water is flowing overland towards Darby Creek.

"The course of a spilled liquid, once it reached the water table, also would be towards Darby Creek where it would discharge into the stream through springs and seeps. The shallow aquifer is unimportant as a source of water in the site area and the chances are almost nil under present conditions that a contaminant introduced into the shallow aquifer on the Battelle property would be diverted to wells or otherwise intercepted by man in the course of its slow underground journey to the discharge area.

"A greater chance for contamination of ground water supplies would result from radioactive fluid entering the principal aquifer in the site area. This danger is slight, however, for the principal aquifer receives most of its recharge in upland areas some distance west of the Battelle site. Only a relatively small amount of water percolates through the till to reach the principal aquifer in the immediate area of the site. Moreover, a contaminant reaching the principal aquifer in the immediate area of the Battelle property would already be down-gradient from almost all the wells in the area.

* Norris, Stanley, "Hydrology of a Proposed Reactor Site Near Columbus, Ohio", U. S. Geological Survey, Water Resources Division, Columbus, Ohio (August, 1954).

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"Practically the only potential danger of contamination to a specific ground water supply is to the supply now being developed by the Battelle Memorial Institute near the southeast corner of their property. Pumping from the well now being drilled, and from possible future wells, would lower ground water levels in the principal aquifer in the immediate area of the wells. This would induce more recharge locally from the overlying till by creating a cone of influence which might encompass areas where the danger of contamination would be greatest. In view of this possibility a careful record should be maintained of the natural radioactivity of the water from the Battelle plant wells, to be used as a basis for comparison to detect any contamination should spillage occur.

"The greatest danger to human life and property resulting from spillage of radioactive fluid in the Battelle site area would be for the fluid to reach Darby Creek in volume and to move downstream in toxic concentration. Darby Creek is not now used either as a source of public or of large-scale industrial water supply, though it undoubtedly will be used eventually for both purposes. It is however, an important source of water for stock all along its course and it flows into the Scioto River above several important water supply developments, including the supply for the Atomic Energy Commission plant in Pike County.

"The velocity of flow in Darby Creek ranges from very low, when the stream is in pool stage and the only perceptible flow is over the riffles, to very high when the stream is in flood. The velocity of flow is important to any further evaluation of the fate of a spilled liquid once it reaches the stream and it should be studied under various conditions of discharge. Surface water samples should be collected and analyzed to determine the natural radioactivity of the stream, and the records maintained as a basis for future comparison."

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APPENDIX D

HAZARDS CALCULATIONS

Dose Rates From Fall-Out From an Explosion Cloud

The term wash-out is here used to describe that condition in which continuous rain-out and fall-out from a radioactive cloud occurs as it drifts downwind from the origin. The fundamental assumption is that a constant fraction of the radioactivity in the cloud falls out per unit distance as the cloud travels downwind and that for each distance this fall-out rate is that which renders the dose a maximum at that point. Assuming uniform concentration of activity and correcting for attenuation, the dose rate may be expressed as*

$$D_r = \frac{2\sqrt{\pi} \alpha C_x}{\sigma} E_1\left(\frac{h}{\lambda}\right), \quad (D-1)$$

where

D_r = dose rate, r/hr/MW-sec

α = unit dose rate per unit concentration on the ground, 0.57 E(mev), $\frac{\text{r meter}^2}{\text{curie-hr}}$

E = energy per disintegration, 0.5 mev

$$C_x = \frac{C_o}{eX_o}$$

X_o = distance from cloud origin to the point where the dose rate is being considered, m

C_o = activity of the cloud without fall-out but including decay at distance X_o , curie/MW-sec

σ = cloud radius at distance X_o , m

$$E_1\left(\frac{h}{\lambda}\right) = \int_{\left(\frac{h}{\lambda}\right)}^{\infty} \frac{e^{-t}}{t} dt$$

*This equation is derived in: Method of Evaluating Radiation Hazards From a Nuclear Incident, KAPL-1045, Fitzgerald, J. J., Hurwitz, H., Jr., and Tonks, L. (1954).

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h = height above the ground that the dose is measured
(1 meter)

λ = attenuation length, 200 m.

The above equation overestimates the dose rate at all distances.

The value of the activity of the cloud at any time, assuming no fall-out, may be derived as follows: According to the empirical relationship of Way and Wigner* the rate of release of both beta and gamma energy after fission is $\cong 2.66t^{-1.2} \frac{\text{mev}}{\text{sec-fission}}$ for time between 10 seconds and 100 days after shutdown. Hence, the rate of energy release in the cloud per megawatt-second is

$$\frac{2.66 \times t^{-1.2} \times 3 \times 10^{16} \text{ mev/sec}}{\text{MW-sec}}$$

or

$$7.98 \times 10^{16} t^{-1.2} \frac{\text{mev/sec}}{\text{MW-sec}}$$

where

$$1 \text{ MW-sec} = 3 \times 10^{16} \text{ fissions.}$$

Hence,

$$\begin{aligned} C_0 &= \frac{7.98 \times 10^{16} t^{-1.2}}{0.5 \times 3.7 \times 10^{10}} \frac{\text{curies}}{\text{MW-sec}} \\ &= 4.31 \times 10^6 \times t^{-1.2} \frac{\text{curies}}{\text{MW-sec}} \end{aligned} \tag{D-2}$$

where $0.5 \frac{\text{mev}}{\text{disintegration}}$ and $3.7 \times 10^{10} \frac{\text{disintegrations/sec}}{\text{curie}}$ are assumed.

If it requires time t_0 for the cloud to reach X_0 , the dose rate at X_0 at time $t > t_0$ is given by

$$D_r(t) = D_r(t_0) \left(\frac{t}{t_0} \right)^{-1.2} \tag{D-3}$$

where $D_r(t_0)$ is the dose rate at X_0 immediately after fall-out. $D(t, X_0)$, the accumulated dose at X_0 over some time interval, $t_0 < t < t_1$, is given by

$$D(t_1, X_0) = 5D_r(t_0)t_0 \left[1 - \left(\frac{t_0}{t_1} \right)^{0.2} \right] \tag{D-4}$$

*Way, K., and Wigner, E. P., Phys. Rev., 70, 115 (1946).

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The value of the cloud radius, σ , at various distances downwind is obtained from the nomographs constructed by J. Z. Holland.

The results, giving the doses for various distances, stability parameters, and wind velocities, are shown in Figure 16.

Inhalation Dose From Radioactive Cloud

An observer receiving an internal dose from inhalation of a radioactive cloud from an explosion would be in danger of having the thyroid gland, bones, and lungs affected by the more hazardous fission products I^{131} , Sr^{89} , $Sr^{90} + Y^{90}$, $Ba^{140} + La^{140}$, $Ce^{144} + Pr^{144}$, and Y^{91} . The short-lived fission products will also damage the lungs.

Experimental work* has shown that the lungs retain about 50 per cent of the total small particles in the inhaled air. This 50 per cent factor will be assumed to apply to all fission products inhaled. The total activity resulting from the retention of radioactive isotopes in the lungs must be further reduced by an "effective" retention factor, f_o , which varies with the isotope and corresponding organ which is affected. The activity retained in an organ may be obtained from the equation

$$\frac{A(\text{millicuries})}{\text{MW-sec}} = (0.50)(16.8 \times 10^{-3})f_o \frac{J}{v} \frac{Q}{v^{2/3}} \quad (\text{E-5})$$

where

J = inhalation rate, 17 liters/min

v = cloud velocity, m/sec

Q = total curies of given isotope in the cloud at time of inhalation per MW-sec

$V = (\sqrt{\pi} \sigma)^3, m^3$

σ = radius of cloud at time of inhalation, m

f_o = effective retention factor.

The integrated lung dosage for the short-lived fission products is given by

$$D = \frac{88.6}{v_0^2} \left(\frac{v}{x}\right)^{0.2} \frac{r}{\text{MW-sec}}, \quad (\text{E-6})$$

*On the Retention of Air Borne Particles in the Human Lung, IL., H. D. Landahl and T. N. Tracewell, TID-365.

where it is assumed that each disintegration in the lung leads to the absorption of 0.5 mev of energy in the lung tissue.

$$Q = f e^{-\lambda \left(\frac{X}{v} \right)} \lambda \frac{3 \times 10^{16}}{3.7 \times 10^{10}} \frac{\text{curies}}{\text{MW-sec}}$$

$$= 8.11 \times 10^5 f e^{-\lambda \left(\frac{X}{v} \right)} \frac{\text{curies}}{\text{MW-sec}} \quad (D-7)$$

Table E-1 gives the important data for the hazardous isotopes of interest. The cloud radius at the time of inhalation was obtained from the nomographs constructed by J. Z. Holland.

The short-lived fission products were assumed to decay according to the Way-Wigner formula. The amount of activity from this source is given by Equation A-6 where

$$Q = 2.16 \times 10^6 \left(\frac{v}{X} \right)^{1.2} \frac{\text{curies}}{\text{MW-sec}} \quad (D-8)$$

Energy Releases

Instantaneous Reactivity Additions

The analysis presented here is an attempt to estimate the fission energy release after an instantaneous addition of reactivity. It is postulated that the reactivity is decreased only by core expansion caused by internal pressure buildup due to vaporization of fuel. The vaporization of fuel is assumed to begin after sufficient energy has been supplied to heat all of the fuel to the boiling temperature. If the core contains 80 kg of fuel, the energy required to reach the boiling temperature is 54 MW-sec. An additional 133 MW-sec would vaporize the fuel. The model chosen is a spherical reactor with reflector.

The equations which were used to define the physical model are as follows:

$$\Delta R(t) = \frac{2\lambda}{\rho R} \int_0^t \int_0^{t'} P_0(t'') dt'' dt' \quad (D-9)$$

$$\frac{dP(t)}{dt} = \frac{\Delta k_p(t)}{\ell} P(t) \quad (D-10)$$

$$\Delta k_p(t) = \Delta k_o - \alpha_1 \Delta R(t) \quad (D-11)$$

$$\frac{dN(t)}{dt} = \left[\frac{N_o - N(t)}{N_o} \right] \frac{P(t)}{\Delta H_v} \quad (D-12)$$

$$p_o(t) = \left[p^*(t) - p^*(o) \right] \alpha_2 \quad (D-13)$$

$$p^*(t) = \frac{\bar{R}N(t)T(t)}{V(t)} \quad (D-14)$$

$$\frac{dT(t)}{dt} = \frac{P(t)}{C_v N_o} - \frac{\bar{R} T(t)}{C_v} \left[\frac{1}{V(t)} \frac{dV(t)}{dt} - \frac{1}{N(t)} \frac{dN(t)}{dt} \right] \quad (D-15)$$

$$\frac{dV(t)}{dt} = 4\pi R^2 \frac{d\Delta R(t)}{dt} \quad (D-16)$$

where

t = time after vaporization begins, sec.

$\Delta R(t)$ = radial distance core-reflector interface changes as a function of time, cm.

R = initial effective radius of core-reflector interface, 44 cm.

$p_o(t)$ = pressure above atmospheric pressure at center of core at time t , atm.

ρ = mean density of core.

$P(t)$ = reactor power at time t , MW.

$\Delta k_p(t)$ = prompt excess multiplication at time $t = k_{eff}(1-\beta) - 1$.

k_{eff} = effective multiplication factor.

ℓ = mean neutron generation time, sec.

Δk_o = initial value for $\Delta k_p(t)$.

α_1 = core expansion reactivity coefficient, cm^{-1} .

$N(t)$ = number of g-moles of U^{235} in gas phase at time t .

N_0 = number of g-moles of U^{235} in core.

ΔH_v = heat of vaporization, MW-sec/g-mole.

$p^*(t)$ = average gas pressure in core at time t , atm.

$T(t)$ = temperature of gas phase at time t , °K.

$V(t)$ = volume of gas phase at time t , liter.

\bar{R} = gas constant, $\frac{\text{liter-atm}}{\text{g mole-}^\circ\text{K}}$

C_v = heat capacity of gas phase at constant volume,
 $\frac{\text{liter-atm}}{\text{g-mole-}^\circ\text{K}}$ or $\frac{\text{MW-sec}}{\text{g-mole-}^\circ\text{K}}$

Equation D-9 is an approximate solution of the equation of motion for mass particles in the core. This equation is obtained in a manner similar to that employed by Mills*, except the pressure variation in time is not assumed to be known. The parameter λ , following Mills, is taken to be 1/3.

Equation D-10 relates the rate of power rise with the multiplication and power at time t . It is approximate in that delayed neutrons are neglected, but since prompt critical conditions are of interest this is not serious. The mean generation time, ℓ , was calculated to be approximately 6 μ sec.

Equation D-11 relates the core expansion to the reactivity. The coefficient α_1 was calculated to be 0.0332.

Equation D-12 relates the rate of vaporization of fuel to the power level and the number of moles already vaporized. The values of N_0 and ΔH_v are 340 moles and $0.389 \frac{\text{MW-sec}}{\text{mole}}$, respectively. It is assumed that only fissions which occur in the solid phase are effective in vaporizing fuel. The fissions occurring in the dense gas phase are assumed to increase the internal energy of the gas.

Equation D-13 relates the pressure at the core center to the average pressure in the gas core. The constant α_2 was calculated to be 1.25.

Equation D-14 is the perfect gas law and Equation D-15 is an energy balance for the gas phase. Equation D-16 is an approximate equation relating the rate of gas volume change to the core expansion.

*Mills, M. M., "On the Hazards Due to Nuclear Reactors", Reactor Science and Technology, p 55.

Equations D-9 and D-16 were solved by means of an electronic operational analog. The initial conditions were chosen so that for a given value of Δk_0 , the power level was such that 54 MW-sec had already been supplied, i. e.,

$$P(0) = 54 \text{ MW-sec} \left(\frac{\Delta k_0}{\ell} \right).$$

The transient was terminated when $k_p(t) = 0$ and the fission energy of the transient was obtained. The results of several determinations of the energy release for different values of Δk_0 are noted in Figure 20.

Gradual Reactivity Addition

The energy release for the case of maximum rate of reactivity addition upon closure (\$3.20/sec) was calculated as follows: An approximation for the inverse period, p , when an energy E has been generated is

$$p^2 = \frac{2}{\ell} \frac{dk}{dt} \ln \frac{Ep}{P_0 10^2} \quad (\text{D-17})$$

where

$$\frac{dk}{dt} = \text{rate of reactivity addition, sec}^{-1}$$

$$E = 54 \text{ MW-sec}$$

$$P_0 = \text{startup power MW.}$$

Taking $\frac{dk}{dt}$ corresponding to \$3.20/sec and $P_0 = 5 \times 10^{-10}$ MW, it is found that $p = 350 \text{ sec}^{-1}$ and this corresponds to an instantaneous addition of $\Delta k_p = 0.00210$. Approximately 1/4 millisecond is required to disassemble the core after $E = 54 \text{ MW-sec}$ for this case and hence the reactivity added during this time is negligible. Consequently, D-17 may be used in conjunction with Figure 20 in order to estimate the fission energy release for the addition of reactivity upon closure.

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