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HAZARDS SUMMARY REPORT
for the
GCRE CRITICAL-ASSEMBLY EXPERIMENTS
by

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HAZARDS SUMMARY REPORT FOR THE GCRE CRITICAL-ASSEMBLY EXPERIMENTS

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Critical experiments are described and the hazards attendant to these experiments are evaluated for a gas-cooled water-moderated reactor design.

The proposed reactor has a core 28 in. high and a nominal 2 ft in diameter. The fuel elements will be concentric metal tubes containing UO_2 dispersed in metal. These elements will be located in axial voids through the core. The water moderator and reflector are unpressurized.

The fuel elements for the critical assembly will be uranium foil wrapped on metal tubes located in sealed aluminum cans. These cans are positioned by locating grid plates at the bottom and top of the core.

The reactor design being studied will be controlled by horizontal-acting control blades. A mock-up will be made in the critical assembly of blades of the desired material supported by brackets mounted to the core support structure.

Shutdown control of the critical assembly is achieved by cadmium control blades and by dumping the water into a storage tank. The maximum normal rate of reactivity addition is limited to an estimated 0.04 per cent per sec for all remotely controlled operations. The system is interlocked to insure that safe operational procedures are followed.

Hazard calculations are made to determine the dosage from direct radiation, 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

The Aerojet-General Corporation is designing and constructing a gas-cooled power reactor for the Army Reactors Branch of the AEC. This reactor will be water-moderated and will contain a number of axial voids through which the gas coolant will flow. The fuel elements are contained in these gas-coolant passages.

There are very few experimental data on reactors of this type. This lack of information, combined with the inhomogeneity of the system, makes analysis of the reactor difficult. Therefore, critical-assembly studies are required to confirm the reactor design and to furnish information which will be useful in optimizing the design and in operating the reactor.

Battelle has the facilities and technical manpower to undertake these investigations and has been requested by Aerojet-General Corporation to perform the necessary experiments. A Hazards Summary Report was submitted and a presentation made to the Advisory Committee on Reactor Safeguards in October, 1954, which evaluated the Battelle site and the facility. Reports summarizing the hazards associated with two critical-assembly experiments have been submitted and approved since that time.

These experiments include a beryllium-moderated and -reflected reactor described in BMI-ACRS-611 and a plastic-moderated critical assembly described in BMI-ACRS-615.

This report presents the hazards considerations pertinent to the critical experiments for the Aerojet-General Gas-Cooled Reactor Experiments.

The detailed description of the critical assembly is outlined in later sections of this report. Briefly, it is composed of about 50 to 70 fuel elements surrounded by light water. The individual fuel elements are concentric metal tubes around which uranium-235 foil is wrapped. The nominal core diameter is 2 ft. The water tank containing this core will be about 4.5 ft in diameter. Control of the critical assembly will be provided by coarse and fine, vertically acting poison rods. Shutdown safety is provided by three to six safety rods and emptying the water moderator from the reactor tank.

Objective of the Experiment

The basic objective of the program will be to obtain information which will aid in preparing a reactor of sound design to begin operation in about 2 years. In order to implement this objective the following experiments are planned.

- (1) No adequate calculational techniques are now available to predict the critical mass for a reactor system of the type being considered. Therefore, a determination of the critical mass under various conditions of fuel-element designs and moderator temperatures is planned. This information will minimize the uncertainty in the quantity for the prototype design.
- (2) Reactivity coefficients for various combinations of fuel-element materials and the worth of various combinations of prototype control blades will be determined. These measurements will yield valuable information regarding the prototype reactor-control problems and will aid in improving the design and control effectiveness of the reactor.
- (3) Flux- and power-distribution measurements will be made both macroscopically over the core and microscopically within fuel elements. Such measurements will be important to the final reactor physics and engineering design studies.
- (4) Some moderator-temperature coefficients will be measured in order to ascertain the importance of the moderator temperature to the design and performance of the prototype.

THE REACTOR SITE

Site Location and Description

The site of the Critical-Assembly Laboratory is in Madison County, Ohio, 15 miles west from downtown Columbus. The property faces on the Georgesville-Plain City Road, an improved county highway which is not heavily traveled. 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 520 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.

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 a portion of the Battelle land. The Critical-Assembly Laboratory is labelled ZPR. The nearest boundary of the property is over 1200 ft from the building.

The surrounding area is a farming community and is sparsely populated. The closest town is West Jefferson, population 2500, which is about 2 miles southwest of the laboratory location. The closest building is a barn, 2000 ft northwest of the site, and the closest dwelling is 3100 ft to the southwest. During the summer months, a Girl Scout camp located across the river is inhabited. The site is about 2000 ft from the camp. The estimated total number of residents within 1 mile of the site is 60.

The site is located on level ground having an average elevation of 900 ft above sea level. The average elevation of downtown Columbus is approximately 750 ft. 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 ft 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 ft below the elevation of the site.

Critical-Assembly Building

The Critical-Assembly Laboratory is a building having 10,500 ft² 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 2 ft thick up to a height of 26 ft. 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-in. water-supply main. Battery-powered emergency lights are provided in the assembly and control rooms.

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 ft square and 50 ft high. The three walls facing into the building are of reinforced poured concrete 2 ft thick. The other wall is 1-ft-thick poured concrete. This concrete extends to a height of 26 ft, 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 10-ton crane. The roof is built up of asphalt and gravel over heavy building paper and 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 2-ft 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-deg angle above head height to prevent radiation streaming from the openings from reaching personnel. The openings which are not in use are fitted with shielding plugs; those in use are filled with a material to make them gastight.

Control Room

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

FIGURE 1. TOPOGRAPHICAL MAP

FIGURE 2. BATTELLE ATOMIC ENERGY CENTER

FIGURE 3. AERIAL PHOTOGRAPH OF BATTELLE ATOMIC ENERGY CENTER

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 ft. The walls, floor, and ceiling are of reinforced-concrete construction. Two walls, the floor, and ceiling are 1-ft-thick reinforced concrete. The remaining two walls are 2-ft-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 ft 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 ft² 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.

Safety Equipment

Fire-control and radiation-monitoring equipment are located in the building as shown in Figure 4. The fire-control equipment is concentrated in the areas having the highest probability of accidental fire.

Normally, corridor and room radiation detectors (film badges) are replaced every 3 months.

Second Floor Plan

- ▨ Critical assembly area
- Film badges
- x CO₂ fire extinguishers

First Floor Plan

- G-I powder
- * Tracer lab monitor
- △ Fire hose

FIGURE 4. FLOOR PLAN OF CRITICAL ASSEMBLY BUILDING

DESCRIPTION OF REACTOR ASSEMBLIES

A brief description of the prototype power reactor for which the critical-assembly studies are being made is given below. Following this, a detailed description of the critical assembly, along with the associated controls and instrumentation, is presented.

Power Reactor

The reactor design for which these studies are being made is a package-power type. The reactor will be gas cooled and unpressurized, water moderated. The concentric-tube fuel elements are joined into a common gas plenum at the upper and lower ends. This assembly of fuel elements operates in a tank of water, which serves as moderator and reflector. The moderator water is thermally insulated from the fuel assemblies. This system is designed to provide the desirable thermal efficiency of the gas-cooled system and also the fuel economy of the water-moderated system.

The reactor core will contain a number of fuel elements (estimated to be between 50 and 75) arranged to approximate a 2-ft-diameter cylinder. The fuel subassemblies will be made up from a number of concentric shells containing UO_2 dispersed in a metal. The over-all fueled length is 28 in.

Control of the power reactor will be provided by several rods worth about 0.4 per cent each, driven radially into the core. About 12 radially driven poison blades of about 2 per cent reactivity each will be used for shutdown and safety.

Critical Assembly

The complete critical-assembly experiment consists of the core structure, the reactor tank and associated plumbing, the assembly stand, and the control and research instrumentation. These components and their place in the system are described in detail in this section.

Figure 5 shows the support structure, the mock-up gas plenum, the fuel elements, and the water moderator-reflector. These components are contained in a cylindrical reactor tank 52 in. in diameter and 71 in. high. The tank is mounted on the assembly stand with the bottom of the tank about 7 ft above the floor level. In this way, it may be drained by gravity through two quick-acting 4-in. dump valves into the storage tanks at floor level.

The approximate locations of the critical-assembly control blades (vertical acting) and the mock-up prototype control blades (horizontally mounted) are visible in Figure 5. Space has been provided for six safety blades 5-3/4 in. wide and one shim control blade 1-1/4 in. wide for the critical-assembly operation. These blades are a sandwich of metal and cadmium about 0.2 in. thick. The fuel-element spacing plate is provided with safety- and control-blade guides. The mock-up prototype control blades will be mounted on a bar which is rigidly supported between the fuel-element support plates. These blades will be plates of tungsten up to 5-1/2 in. wide.

The fuel elements will consist of concentric metal tubes, each wrapped with highly enriched uranium foil. The foil sheets are 0.001 in. by 28 in. by various widths and will be held on the tubes by tapes. The concentric tubes will be spaced and supported by brackets attached to a central support tube or mandrel. The fuel element will be located in a sealed aluminum tube with insulation between the aluminum and the outermost metal tube. The assembly is shown in Figure 6. The core will contain between 50 and 75 of these fuel elements mounted between two aluminum supporting plates. The supporting plates are in turn rigidly fastened to the water tank. Locating grid plates will be provided at both top and bottom as shown in Figure 5.

A monitoring technique will be set up to detect any seepage of water into the fuel-element cans. This may be done by electrically insulating the central support tubes from the sealed aluminum jacket tube and continuously reading the electrical resistivity of the insulation. Should a leak occur, the breakdown in resistivity will be apparent.

Figure 7 illustrates the details of the plumbing and heating systems. The dump system consists of the two quick-acting 4-in. valves (air-to-close, spring-to-open type) and the storage tanks. A centrifugal pump of low capacity (30 gpm maximum) will be used for filling the reactor tank. Accurate liquid level can be obtained with the tank filled by means of the overflow pipe. Other intermediate levels will be obtained by slow-fill methods and by draining the water back through the fill line. A steam-to-water heat exchanger and electrical heaters will be used for heating the reflector region for temperature-coefficient studies. The water will be continuously agitated during these tests to obtain near isothermal temperature effects. The temperature will be measured by platinum resistance thermometers and recorded on a strip-chart recorder.

Some of the details of the location of the instrumentation, neutron source, and other control and research devices appear in Figure 8. All instrumentation projecting into the core tank will be completely sealed to prevent leaks. The details of the source-rod drives and liquid-level indicators are described in another section.

The main supporting columns of the reactor test stand will be either 8-in.-wide-flange beams or 8-in.-diameter pipe. A support deck for the reactor tank assembly will be composed of structural steel beams welded or bolted to the main supporting columns. The reactor core tank will be bolted to a 1-in.-thick steel plate which will rest on the support deck. A superstructure of structural steel members will support the control drives. Figure 8 shows the location of the platforms, source mounting bracket, etc.

FIGURE 5. GCRC CRITICAL ASSEMBLY CORE AND TANK

FIGURE 6. FUEL-ELEMENT CONSTRUCTION

FIGURE 7. SCHEMATIC OF PLUMBING SYSTEMS

FIGURE 8. SCHEMATIC OF TEST STAND
AND CONTROL EQUIPMENT

CONTROL OF THE CRITICAL ASSEMBLY

The control and monitoring system of the critical assembly is designed to provide the operator with complete information regarding the condition of the reactor at any time. Interlocks and mechanical devices are provided to prevent unsafe operation of the critical assembly and to provide for automatic shutdown.

Control-Blade Operation

Seven vertical-acting regulating and shim blades are available to the operator. Six are metal-cadmium sandwich construction 5-3/4 in. wide by 0.2 in. thick. The other blade is of the same thickness and construction but only 1-1/4 in. wide. Each of the rods has its own rod-drive unit but only one power supply is provided. A selector switch makes this power available to only one blade at a time. It is planned that the experiments will go critical with all driven rods completely withdrawn with the possible exception of a small amount of shim control. In this way, the excess reactivity will be very small.

The control-rod drive consists of a motor-clutch-type drive unit, two Selsyn position indicators, a rack and pinion gear, and a hydraulic shock absorber. The details of the drive unit are shown in Figure 9 and four of the power units and one rack-and-pinion unit is shown mounted on a test stand in Figure 10.

The motor-clutch unit contains a reversible variable-speed, a-c motor with a gear reducer which permits the motor a maximum speed of 3 rpm. The clutch is a stationary field, electromagnetic type which operates on 90 v dc. It may be disengaged automatically or manually from the reactor control console by removing the power to the magnet.

Following the clutch is a unit containing the Selsyn position indicators and a pair of bevel gears. This whole unit can be pivoted about the drive shaft from the clutch. This arrangement permits locating the motor-clutch unit permanently and turning the Selsyn unit so that the drive shaft to the control rod will reach the rack-and-pinion unit regardless of its direction from the motor-clutch unit.

One Selsyn is coupled to the drive shaft through a permanent-magnet coupling and a pair of spur gears. This Selsyn makes one revolution for 1 in. of control-rod travel. The other Selsyn is connected to the drive shaft through a worm gear and the spur gears which connect the first Selsyn. This second Selsyn gives one revolution for 50 in. of rod travel. These Selsyns are connected to the rod side of the clutch so that they always indicate the control-rod position rather than the position of the driving motor. The permanent-magnet clutch permits the rod to scram without the fine-reading Selsyn following the drop. A magnet rezeros the fine Selsyn after the drop.

The shaft from the drive unit goes to the rack-and-pinion unit and drives the pinion gear. The rack is fastened to a round steel shaft which goes through the bearings in the rack-and-pinion housing. The control rod is connected to the lower end of this shaft. A hydraulic shock absorber is attached to the upper end. This consists of a piston fastened to the rod and a tapered cylinder filled with oil. This shock absorber strikes the upper face of the rack-and-pinion housing to arrest the motion of the steel shaft to which the control rod is attached.

N31849

FIGURE 9. ROD-DRIVE UNIT

N31847

FIGURE 10. ROD-DRIVE ASSEMBLY

The motor can drive the rod upward or downward at any speed up to 22 in. per min. When the power is cut to the magnetic clutch, the control rod falls under the action of gravity with an acceleration of approximately 0.7 times gravity to scram the reactor.

The rack-and-pinion unit is supported by a pair of steel bars spanning the reactor assembly. The unit is clamped to these bars by several bolts which pass through a flange in the rack-and-pinion housing and a clamp ring below the bars. A typical support can be seen clearly in Figure 10. The position of the support bars may be varied and the position of the rod between the bars can be varied over a considerable range. This permits complete freedom in the location of the control rods in the reactor core.

Source Drive

A polonium-beryllium neutron source will be used for reactor startup. The source will be in a capsule approximately 0.225 in. 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 source will enter the reactor from above through a tube which will penetrate into the reactor core.

The source will be pushed from its shielded container into the core and pulled out by means of flexible cable. This cable will have a spirally wound wire on it and will be driven by a gear so that the gear and cable have the appearance of a worm and worm gear. However, it will act like a rack-and-pinion unit. The power will be from a d-c geared motor. Limit switches operating off the end of the cable will prevent overdriving of the source in either direction. An interlock switch will be activated only over the last 6 in. of travel to permit the operation of the rod drive and the filling pump motor if the flux is not above a prescribed level.

Moderator-Water Level Controls

In addition to inserting the control blades, the reactor will be shutdown normally by dumping the moderator water into the storage tanks. The plumbing system for this operation and other water-level control is shown in Figure 7. Fine control of the water height is provided by the valve in the pump line which may be operated from the reactor console. With the valve open water will leak back through the pump impeller at a slow rate so that accurate level control is possible.

The moderator level will be indicated by two methods. One method uses a Tank-o-Meter indicator and provides a rough value ($\pm 1/20$ in.) of the liquid level. The other method is an electrical-circuit liquid-contact device which gives values of the liquid level to within ± 0.01 in. The mechanical schematic of the Tank-o-Meter indicator and the circuit diagram of the liquid-contact device are shown in Figure 11.

The Tank-o-Meter operates by measuring the pressures required to just bubble gas through the line located in the core tank. This pressure is measured by the manometer fluid in the Tank-o-Meter. By selecting a manometer fluid having lower density than water, the liquid level in the reactor tank can be read on an expanded scale.

Schematic of Tank-o-meter Indicator

Schematic of Accurate Liquid Level Detector

A-26210

FIGURE 11. SCHEMATIC OF WATER HEIGHT MEASURING INSTRUMENTS

The liquid-level meter is a hunt and seek, electronically controlled device. When a plumb bob contacts the liquid surface current is conducted through the liquid which causes a reduction of the voltage at the control grid of a 6AG7 tube. The resulting plate-current energizes a relay which closes a drive-motor circuit causing the plumb bob to be lifted. When contact at the surface is broken the relay opens and the bob is driven downward until contact is restored.

Upper- and lower-limit switches are incorporated in the drive-motor supply circuit. A Selsyn transmitter geared to the drive motor permits remote indication of water level to an estimated 0.01 in. If it is necessary to improve the accuracy, the liquid-contact pointer may be driven manually rather than automatically in a "hunting" circuit.

Scram Relays and Interlocks

The interlock and "scram" circuits are entirely switch and relay circuits and contain no electronics. The circuits are designed to activate when switches and relays are opened with the exception of the contact-meter relays in the scram circuits. Thus, with the exception of the contact meters, relays are fail-safe and are connected so they activate on removal of power from coils. In circuits containing large inductive loads, precautions have been taken to prevent arcing at relay contacts. These precautions include the use of double-break relay contacts and selenium rectifiers to short circuit the inductive emf.

The functions of the interlock and scram circuits are outlined in Figure 12.

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. 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 for the operation of the critical assembly are located in a position in front of the operator. The rod-position indicators are on an inward-sloping panel above this. 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 of Figure 13. Three 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 monitored by a meter relay for the safety circuit. The other ion chambers are connected to logarithmic electrometers in the assembly room. The output from these electrometers is connected

(A) To scram relay causing a scram if door is opened during reactor operation. (B) To warning light on console to indicate door open at any time. (C) To interlock light on console to indicate open interlock circuit. (D) To interlock relays to prevent withdrawal of control rods. (E) To interlock relays to prevent filling the core tank with water. (F) To interlock relays which prevent operation of core-material reactivity changing device.

FIGURE 12. FUNCTIONS OF INTERLOCK AND SAFETY SYSTEM

FIGURE 13. BLOCK DIAGRAM OF REACTOR INSTRUMENTATION

directly to the safety circuits and also to differentiating circuits on the console to provide visual period indications and to activate the safety circuits. A strip-chart recorder provides a record of the log level. The slope of the curve on this chart will be a record of the reactor period.

A gamma channel consisting of a high-voltage supply and a scintillation detector gives a visual indication of the power level. This level 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 startup and accurate neutron 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. One circuit 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 linear amplifier, scaler and high-voltage supply. The counter and its preamplifier are located in the assembly room.

Both of these channels are pulse-counting instruments and are independent of the safety instrumentation.

TYPES OF EXPERIMENTS

Material in this section discusses the type 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 on the need for the particular information and 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 cores having approximately 51 to 75 fuel elements. This will be done for both a clean core configuration and for core configurations containing various combinations of prototype control rods. The uranium loading per fuel element is variable and the number of elements to be used is variable between the limits stated. The core will always have a bottom axial and a radial reflector and most of the experiments will be done with a top axial reflector.

Control Evaluation

The critical-assembly operational control rods will be calibrated to insure the safety of the operations and to provide means of measuring other changes in reactivity. For example, temperature coefficient of reactivity can be measured in terms of the equivalent change in a calibrated rod position.

The reactivity worth of each type of prototype control rod will be measured and the worth of various combinations of these rods will be determined. In general, these studies will determine the integral worth of the rods. A few determinations of differential rod worth will be made on a selected number of the prototype rods.

The critical-assembly operational rods will be calibrated by positive period, rod oscillator, subcritical multiplication, and rod-drop techniques. All of these methods are techniques which have been used on other reactors.

The prototype control rods will be evaluated by intercomparing them with the critical-assembly operational rods, by subcritical multiplication measurements, and by evaluating the rod in terms of the change in fuel mass required to maintain criticality.

Flux- and Power-Distribution Measurements

Axial and radial flux distributions will be measured by activating indium and/or manganese foil materials at various points in the core. Cadmium differences will be taken to separate the thermal from the epithermal fluxes. These foils will be mounted on aluminum or plastic holders when positioned in the water region and on the structural components of the elements themselves for activation within the coolant passages.

Detailed flux measurements will be made through typical unit cells and through fuel elements at a number of locations in the core. Detailed flux measurements may also be made in the vicinity of prototype control rods.

Power distribution will be measured through the core and in detail through selected fuel elements. Measurements will be made by counting the fission-product activity in irradiated uranium samples or by counting the fission-product activity in aluminum catcher foils which have been placed adjacent to the uranium fuel.

Reactivity-Coefficient Measurements

The reactivity coefficient for uranium-235 at various locations in the core will be determined as a means of evaluating the reactivity effects of changes in critical mass. The reactivity coefficients of structural materials and of insulating materials will be determined as an aid in optimizing the reactor design.

The reactivity coefficient of structural changes in the fuel elements will be measured by manually or remotely inserting and removing the elements. Changing elements manually will require that the reactor be completely shut down and water drained from the tank. Changing elements remotely can be accomplished by the apparatus described in Appendix F. The reactor system is interlocked so that the reactor must be subcritical by at least 1 per cent $\Delta k/k$ before elements can be changed. This amount of reactivity is considered sufficient to override the reactivity worth of any alteration to the fuel elements.

Optimization of Design

A limited number of experiments may be undertaken to optimize the design of the core. They will consist of testing a small number of fuel elements of modified design and possibly of testing prototype control rods of modified shape, material, or location. The fuel-element spacing and the core height will remain constant and the percentage of structural materials and voids will vary only within narrow limits.

Reactor Parameters and Operating Characteristics

By means of the detail flux- and power-distribution measurements described above, the thermal utilization and fuel-disadvantage factor may be determined. Partial-water-height experiments will be conducted to determine the migration area for the core. All the partial-water-height experiments will be done with nearly full water height since all increases in fuel loading will be accompanied by increases in the number or amount of insertion of prototype rods into the core.

The moderator temperature coefficient of reactivity will be measured for uniform core temperatures from room temperature to approximately 180 F. These temperature changes will be obtained by heating the moderator water.

The effect of flooding of the fuel elements may be determined at a number of locations in the core by adding water or water-simulating material such as polyethylene to individual elements and inserting these elements in the core. The reactor will be made critical and the effect of the change will be measured in terms of the change in position of a calibrated rod or by measuring the positive period. To facilitate these experiments, the modified elements may be interchanged with standard core elements via the element-exchange apparatus described in Appendix F.

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 these quantities may change slightly but not significantly. Composition of the first core is given in Table 1.

TABLE 1. REACTOR CORE COMPOSITION

Material	Volume Per Cent
Water	43
Structural	10
Aluminum	11
Uranium	1
Void	35
Insulation(a)	9

(a) The insulation will occupy an equivalent volume of the void.

The neutron-generation time has been calculated as 50×10^{-6} sec. A large error in this value will not appreciably affect the results of the hazards calculations or the conclusions drawn from them.

Some calculations have been made concerning the temperature coefficient of reactivity. The slow temperature coefficient, where the power rises slowly enough for appreciable heat to be transferred to the moderating water, is the difference between two rather large numbers. The difference is less than the uncertainties in the numbers which are being subtracted and therefore the over-all temperature coefficient is not well known. The calculations show that the slow temperature coefficient will be of the order of $10^{-5} \pm 1.5 \times 10^{-5} \Delta k/(k)(C)$ and is believed to be negative.

The prompt temperature coefficient, due to the heating of the fuel elements only, has been calculated to be $-5.9 \times 10^{-6} \Delta k/(k)(C)$. This coefficient is made up from five components. Individually, these are (1) fuel expansion, $-5 \times 10^{-6} \Delta k/(k)(C)$, (2) moderator expansion, $-0.37 \times 10^{-6} \Delta k/(k)(C)$, (3) structural expansion, $+0.5 \times 10^{-6} \Delta k/(k)(C)$, (4) uranium-235 Doppler effect, $+0.4 \times 10^{-6} \Delta k/(k)(C)$, and (5) uranium-238 Doppler effect, $-1.40 \times 10^{-6} \Delta k/(k)(C)$.

These coefficients are not important in normal operation since the reactor is not producing significant amounts of heat. In an accident the prompt temperature coefficient would be important, but it is of so small magnitude that it would not contribute significantly to the course of the accident.

PLAN OF NORMAL OPERATION

The general administrative control and operating responsibility for the Critical-Assembly Facility are outlined in the organizational chart of Figure 14. The detailed operating rules and the general laboratory rules for the critical-assembly operation are included in Appendix E.

Procedures are discussed below which cover the pertinent operation of the GCRE critical assembly. These are the operations which are required to perform the experiments described in the previous section.

General Limitations Affecting Operations

A basic limitation on all experiments and in all operations will be a rate of change of reactivity limitation of 0.04 per cent $\Delta k/(k)(\text{sec})$. This limitation will apply to all controls which may be operated remotely. It will be applied in operating the control rods and in changing reactivity-coefficient samples and fuel elements remotely. It will also be applied in adding water when the reactor is less than 5 per cent subcritical.

The normal procedure for obtaining criticality will be by withdrawing control rods and not by increasing the water height. This limitation is used because the direction of motion of the control rods can be reversed in milliseconds while the direction of flow of water may take time of the order of a second to reverse.

Normally six safety rods will be used in the experiments and no experiment will be conducted with less than three safety rods. The actual number will be determined so that the core will always be subcritical with the rod of greatest reactivity worth in the full-out position.

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

Core Assembly 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. Procedures and controls will be adopted 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 No. 3 to Procedures Manual BMI-PM-607. These procedures will not be 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 700 g in one batch, with the exception of the reactor assembly itself.

FIGURE 14. ORGANIZATION CHART

- (2) The location of the uranium and the amount assembled at any one point will be carefully controlled by having a flow and balance record at each point at which the material is handled.
- (3) A supervisor will be appointed who will have responsibility for each procedure being carried out.

In addition to these precautions, the multiplication of the reactor will be measured, recorded, and evaluated during assembly of the core. Since assembly will be carried out with the water completely drained from the reactor tank the multiplication is expected to be very small.

Precritical Operation Checkout

Prior to each day's reactor operation, and more often if the supervisor deems it necessary, a complete check will be made of the instrumentation and safety mechanisms of the reactor. Each radiation detector will be checked for calibration with a neutron source. Each scram circuit will be checked and each scram mechanism will be operated using a source to initiate the scram. The interlocks will be checked by attempting operation contrary to the 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 checkout 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 occasion 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 subsequent approaches to criticality will be made with less data taking and plotting.

Initial Approach to Criticality

Each approach to criticality will be preceded by the operational checkout described above. Following the checkout, everyone will leave the assembly room and the assembly room door will be closed. The source will be inserted and the control rod which is interlocked into the water pumping system will be fully withdrawn. At least one of the remaining control rods will be partially withdrawn to a point where it will be immediately effective on scram. A number of control rods, sufficient to make the reactor subcritical with the full water height, will be interlocked so that they must be in their full-in position before the water can be pumped. Normally this rod full-in interlock will be satisfied with one of the wide control rods. This will leave the positioning of several control rods to the option of the operator.

With the control rods set up in the positions demanded by the interlock circuits and with the remaining control rods partially withdrawn, the dump valve will be closed

and the water pump and filling valve operated. The counting rates and the meter readings will be taken on all of the nuclear detectors with no water in the tank. The water will be pumped to the level of the bottom of the active core region and data again taken. The water height will then be raised in steps of approximately 5 in. each, with detector readings taken at each height. The plot of the inverse of the readings as a function of water height will be extrapolated after each change to predict when the reactor will go critical with the chosen rod configuration. Normally full water height will be reached without criticality being attained. The remaining control rods will be withdrawn stepwise with the continuation of the plotting of the inverse of the detector readings with each step. The steps used in raising water height or withdrawing rods will be made smaller as criticality is approached.

When the reactor power rises into the operating range the source will be withdrawn in steps and a control rod adjusted to keep the power at a nearly constant level. When the source is fully out and the power remains level, the reactor will be critical.

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 will be used. In the subsequent approaches to criticality, the changes in water level and rod position will be continuous rather than stepwise, and counts on the pulse-counting channels will not be plotted. However, the pulse-counting channels will be in operation so that the operator will have an additional visual and an aural signal of the power level. The progress in this approach to critical will be founded on the power-level recorder and the meters on the console, which will be in operation during all of the reactor experiments.

Measurement of Reactivity Changes

There will be a number of experiments in which the reactivity of the reactor will be changed by a small amount. In conducting such experiments, the general policy will be to make the reactor subcritical before introducing a change having an unknown effect on the reactor.

In measuring the temperature coefficient of reactivity, the reactor will be made critical at a given temperature and then a control rod will be inserted a small distance to make the reactor slightly subcritical. With the neutron source moved into the core, the heating system and the stirring motors will be turned on to raise the temperature and to maintain it uniform across the core. The behavior of the temperature patterns in the reactor will be observed on the several temperature indicators and the reactivity behavior of the reactor will be noted by the multiplication changes in the subcritical core. After a temperature rise of 3 to 10 C has occurred, the heat will be turned off and the core allowed to come to equilibrium. The source will be withdrawn and the control rod adjusted to again make the reactor critical. If the temperature coefficient is small or strongly negative and if the temperature rise is slow and uniform, subsequent temperature measurements may be carried out with the reactor at or near critical at all times.

If the reactivity changes are caused by interchanging fuel elements, the sequence of operation described in Appendix F will be followed. The exchanging mechanism is interlocked so that the reactor must be at least 1 per cent subcritical before any exchange can be made. Further details of the normal plan of operation are described in Appendix F.

To determine the effect of flooding, fuel elements containing water or water-simulating material will be inserted in the core. The elements may be inserted by the fuel-element-exchange mechanism following procedures outlined in Appendix F if only a single element is flooded. To investigate the effect of an entire region of "flooded" elements, the elements will be manually inserted in the core with the water drained from the tank.

Processing and Disposal of Radioactive Materials and Solutions

The only processing of material planned for the Critical-Assembly Laboratory is the preparation of small uranium foils. All other processing will be done in the uranium area at the main Battelle Laboratories, which are equipped and experienced in all phases of material handling.

The present program plans to use only water as the moderator. This water will be demineralized and is not expected to become radioactive. However, the water will be checked for radioactivity approximately once per month and prior to being dumped from the system.

Since all experiments will be done at low power (0.01 to 10 w) there will be no high-level induced radioactivity. Examinations for fission-product contamination in the assembly room will be made by taking periodic air samples and smear tests. Under normal operations, there is no possibility of fission-product contamination in the water since the water never comes in contact with the fuel elements.

SAFETY MECHANISMS

This critical assembly does not have the inherent safety of the usual water-moderated reactor where there is good heat transfer from fuel to moderator. Consequently, considerable emphasis has been placed on designing the facility to minimize the possibility of accidents resulting either from malfunctioning of equipment or from operator error. The principal features incorporated to make the assembly safe are described below.

Critical-Assembly Control Rods

For the control of the critical assembly, a total of six wide control blades and two narrow blades have been constructed. The wide blades are estimated to have 2 per cent reactivity control each, and the narrow blades 1/2 per cent each. Normally six wide blades and only one narrow blade will be used in the operations. Critical operations will be permitted with as few as three wide blades, with the restriction that the reactor is subcritical with the control rod having the greatest worth in the full-out position. The fulfillment of this last requirement will insure that there will be greater than 2 per cent control of the reactor at all times.

The control rods are each attached to a rod drive unit but only one power supply is provided. Thus each rod is independent of the others and only one can be operated at a time. In a reactor scram, all the rods fall under the action of gravity; it is estimated that they will have an acceleration of 0.7 g and will require approximately 1/2 sec for full insertion. The operation of the rod drive and scram mechanism will be tested daily in the checkout procedure.

The maximum rate of reactivity change for the withdrawal of a control rod is estimated to be 0.04 per cent $\Delta k/(k)(\text{sec})$. If a rod were moved continuously starting with the reactor just critical, it would take about 18 sec for the reactor to reach prompt critical. Thus the scram circuits would have more than enough to operate if the operator had not taken corrective action.

Water-Filling and Dump System

A curve estimating the reactivity effect of water height and the differential worth of water height changes is given in Figure 15. In the present experiments, a low capacity pump is used in the water filling system. When the water is approaching the top of the core, the filling rate is approximately 20 gal per min. If the core is critical with full water height, the reactivity will be increasing at the rate of 0.02 per cent per sec and the core will be approximately 1 per cent subcritical when the water is 1-1/2 in. below the top of the core. When the water is 8 in. below the top of the core and rising, the reactivity will be increasing by only 0.04 per cent per sec and the core at this point is about 5-1/2 per cent subcritical.

Normally criticality will not be attained by the addition of water but by withdrawing rods. The interlock system is arranged so that rods sufficient to maintain the reactor subcritical must be in their full-in positions while water is being added.

FIGURE 15. REACTIVITY EFFECTS OF WATER HEIGHT

Note: the curves are developed from the relation

$$\frac{\partial \rho}{\partial h} = \frac{2\pi^2 M^2}{h^3 k_{\infty}}$$

where:

- ρ = reactivity
- h = water height plus extrapolation lengths
- M^2 = migration area, taken as 70 cm²
- k_{∞} = infinite multiplication factor, taken as 1.45.

However, if there were a miscalculation in the core criticality or a failure in the interlock system, the rate of addition of reactivity with the increasing water height is not great enough to present a hazardous situation as shown above.

On a reactor scram two 4 in. dump valves are opened completely. The valves will open in a small fraction of a second and will uncover the core in an estimated 20 sec. This is a relatively slow scram and is considered as a backup scram to the operation of the rods. It also insures that there will be no water in the reactor tank while people are in the room and makes it safe to work on the core.

The dump valves open upon failure of either the building electrical system or the building compressed-air system.

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 the reactor
- (3) Reaching critical by increasing the water height
- (4) Closing dump valves or pumping water without a safety rod cocked
- (5) Moving rods and pumping water at the same time
- (6) Removing or reinserting fuel elements remotely without sufficient rods in the core to make the core subcritical.

Instrumentation

The safety instrumentation includes four detectors, three of them ionization chambers and one a photomultiplier-scintillation detector. Each of the detectors has a level scram originating from it, and two of them have period scrams. All instruments are d-c devices minimizing the possibility of scram failure from overloading.

With the exception of the contact meters originating the scrams, all relays are closed during normal operation and open on a scram signal or power failure.

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

MAKEUP 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 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 2.

TABLE 2. POPULATION DISTRIBUTION WITHIN
VARIOUS DISTANCES FROM SITE

Distance Radius	Population
1500 ft	0
2500 ft	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 about 4000 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 3.

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 400 families. Future expansion is to include several industries; the information regarding these industries has not 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 the 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 A.

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 deg 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 three 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-wind 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.

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, 1957. The bulk of the summer rainfall comes in frequent thunderstorms, and tornadoes are not unknown. However, local storm records show that only four tornadoes have 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 are 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.

TABLE 3. INDUSTRIES WITHIN A 10-MILE RADIUS OF PROPOSED REACTOR-FACILITY SITE

Industry	Products	Number Employees ^(a)	Distance From Site, miles
Terndstedt, Columbus Division, GMC	Auto parts	3931	8
Westinghouse Electric Corporation	Refrigerators, appliances	~4000	8
Stokely's Canning Company ^(b)	Canned sweet corn	75-85	2
Hartley Printing and Publishing Company	Printing and publishing	27	10
Columbus General Machine, Inc.	Dies, tools, fixtures	12	10
H. J. Upperman and Sons	Lumber	12	9
Columbus Stationery Company	Stationery	Less than 10	9
Five Manufacturing Company	Farm wagon unloaders	Less than 10	9
Georgitons Candy and Ice Cream Company	Ice cream, candy	Less than 10	10
Merriman Cement Products, Inc.	Cement blocks	Less than 10	2
Stiles Gauge Pin Company	Gauge pins	Less than 10	9
West Jefferson Sand and Gravel Company	Sand and gravel	Less than 10	3

(a) May, 1956.

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

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 and is underlain by the major aquifer of sand, gravel, and limestone. Yields up to 300 gal per min 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 ft 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 Big Darby Creek; hence, water entering the ground on the Battelle property is already near its place of discharge.

Big Darby Creek accounts for the principal surface-water flow. The mean flow is 420 ft³ per sec, based on a 24-year record. Ground-water seepage from the impermeable deposits in Madison County adds little to stream flow. The water of Big 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 C. It is concluded that in case of liquid spillage, most of the liquid would flow overland to the Big 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 include the possibility of airborne fission products, since the reactor fuel will be uranium metal strips, direct radiation through the assembly room walls, and sky shine around and over the shielding walls. The sky shine may make an appreciable contribution to the neutron level in the remainder of the building since the fuel elements are located in axial voids, permitting the fast neutrons to leak upward without encountering any shielding or moderating material.

Fission-Product Hazard

It is estimated that less than 1.5 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 Teflon-type coating 0.0002 in. thick, which is approximately 1.5 mg per cm^2 . If the fission products are assumed to have an average range of 3.2 mg per cm^2 , the depth from which fission products may escape from the uranium will then be the difference in these two values, or 1.7 mg per cm^2 . This density corresponds to approximately 0.04 mil of uranium. Since the uranium strips are 1 mil thick and fission products may escape from either side, fission products may be expected to escape from approximately 8 per cent of the uranium, providing the fission products are traveling perpendicular to the surface. However fission products will be traveling in all directions not just perpendicular to the surface. An estimate has been made of the number which escape from the region close to the surface. The results indicate that approximately 1.5 per cent escape from the fuel strips. Since several layers of the foil will be wrapped on a single metal tube, the actual amount of fission products escaping will be only a few tenths per cent.

The fuel will be contained in capped aluminum cans, and therefore there will be no fission-product escape in normal operation. However the cans will be uncapped periodically for checking for water leakage and for modifying the fuel element. At these times, air samples will be taken and smear tests made to check for any contamination problems.

Direct-Radiation Hazard

Estimates have been made of the fast-neutron flux and gamma intensities at the doorway to the control room, the closest point to the reactor which can be occupied during operation. The direct neutron radiation at this point is estimated to be 1.6×10^{-2} neutrons/ $(\text{cm}^2)(\text{sec})(w)$ of reactor power. However sky shine will contribute an additional amount to the neutron level in the control room.

Sky shine from a reactor without the axial voids has been estimated to be equal to the direct radiation. If the axial voids in which the fuel elements are located are considered, calculations indicate an estimated 30 per cent increase in the sky-shine effect.

However, the neutrons in the control room resulting from sky shine have been degraded in energy due to multiple scattering and therefore a higher flux can be tolerated. It is estimated that a flux of 66 neutrons/(cm²)(sec) can be tolerated for a 40-hr week. Thus on the basis of fast neutrons alone a dose of only 0.001 MPE* will be received for 40 w-hr of operation.

The gamma intensity in the control room was estimated, taking into account instantaneous and delayed gammas from fission and the captured gammas resulting from neutron absorption in the reactor and in the shielding wall. A buildup factor was also included for the attenuation through the concrete shield and through the reactor core and reflector. The sky shine was estimated as 130 per cent of the direct radiation and was assumed to be equally hazardous. These calculations resulted in 0.047 MPE being received for 40 w-hr of operation.

Normal critical assembly operation is at a fraction of a watt power with occasional operation at from 1 to 10 w for foil irradiation. These occasional irradiation runs at 10 w or under total less than 4 hr weekly. Considering our power levels and our average operating time per week, we conclude that there is no hazard in the normal operation of the critical assembly.

Radiation surveys will be made during early stages of the reactor operation to insure that normal tolerances are not being exceeded.

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.

* MPE is an abbreviation for maximum permissible exposure.

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 (1) direct irradiation by a cloud of fission-product gases, (2) direct irradiation by contaminated ground surfaces due to a fall-out or rain-out from the cloud, and (3) 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 4.

TABLE 4. PARAMETERS FOR DOSAGE CALCULATIONS^(a)

h	n	c^2		
		u=1	u=5	u=10
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
 c = virtual diffusion coefficient, $m^{n/2}$
 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 m

and 8 m, respectively. The results are shown graphically in Figure 16 for several meteorological conditions.

From Figure 16 it may be noted that at the exclusion-area limit, the dose is approximately 0.03 r per megawatt-sec. To attain a LD-50 at this location, i. e., a radiation dose expected to kill about one-half of the persons exposed to it (~ 400 r), would require an accident of 1.3×10^4 megawatt-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.

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 1 m 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 17.

From Figure 17 it may be noted that at the exclusion-area limit, the integrated dose is approximately 2.5 r per megawatt-sec and at the nearest village, the dose is approximately 0.1 r per megawatt-sec. It should be emphasized that these doses are based upon continuous long-term exposure following fall-out. If a shorter, more practical, length of 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 per megawatt-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 18 and 19.

The maximum permissible amounts of the various isotopes of major concern are given in Table 5.

TABLE 5. MAXIMUM PERMISSIBLE AMOUNTS OF SOME ISOTOPES

Isotope	Maximum Permissible Amounts in Total Body, μ curies
Iodine-131	0.3
Strontium-39	2.0
Strontium-90 + yttrium-90	1.0
Barium-140 + lanthanum-140	5.0
Cerium-144 + praseodymium-144	5.0
Yttrium-91	15.0

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

FIGURE 16. DOSES DUE TO PASSING RADIOACTIVE CLOUD

FIGURE 17. DOSES DUE TO FALL-OUT FROM PASSING RADIOACTIVE CLOUD

FIGURE 18. INHALATION ACTIVITIES FROM RADIOACTIVE CLOUD (LARGE LAPSE)

FIGURE 19. INHALATION ACTIVITIES FROM RADIOACTIVE CLOUD (LARGE INVERSION)

The most critical isotope is iodine-131. From Figure 19 the activity of this isotope for the worst diffusion conditions is 2.5×10^{-3} μ curies per megawatt-sec at the exclusion-area limit and 7×10^{-4} μ curies per megawatt-sec at the nearest village. Thus an accident of greater than 100 megawatt-sec is necessary to exceed the permissible tolerance value at the exclusion-area limit, and an accident greater than 4000 megawatt-sec is necessary to exceed the permissible tolerance value at the nearest village.

Hazards to Operating Crew

In the section on Hazards During Normal Operation it was found that 40 w-hr contributed approximately 0.05 MPE and that practically all of this dose was produced by gamma radiation. These results cannot be carried directly over to an accident analysis because, in an accident, some of the fission products would be airborne and the moderating and reflector water would be dumped, thus removing the self-shielding. The delayed gammas from fission account for approximately one-third of the total dose and these delayed gammas would be increased in effectiveness by a factor of 14 by the removal of the moderating and reflector water.

Applying the above changes to the hazard analysis it is found that 0.03 MPE results from the prompt gamma rays and 0.22 MPE from the delayed gammas during 40 w-hr of operation (when the moderator water-shielding effect is removed). Thus 40 w-hr of reactor operation under accident conditions contributes 0.25 MPE in gammas. The neutron effect is small in comparison and has been neglected. These numbers can be transformed to units of megawatt-seconds. An accident of 1 megawatt-sec will produce an exposure of 1.8 MPE or 0.54 r of gamma energy.

The maximum hypothetical accident results in an energy release of 120 megawatt-sec if a most pessimistic shutdown mechanism is assumed. In such an accident, the operating crew would receive an estimated dose of 65 r if the water shielding in the reactor tank is lost following the prompt radiation burst. This dose would be reduced by evacuating the building so that the crew would not be continuously exposed to fission-product gammas.

In more "credible" accidents the dose to the operating crew is reduced by a factor of two.

ACCIDENT CONSIDERATIONS

Procedures used in these studies predict the beginning of any hazardous condition and permit the operator to take corrective action. Operator error or equipment failure are protected against by having a number of parallel safety mechanisms which control the situation and prevent a hazardous condition. However, cases can be imagined where a combination of serious equipment failures and operator error could produce a hazardous situation.

The maximum hypothetical accident describes an upper bound on the hazards associated with this reactor. This accident, considered hypothetical and not credible, assumes a step increase in reactivity of 5 per cent $\Delta k/k$ with all available fuel in the reactor. It is further assumed that 100 per cent of the fission products are released. Choosing an absurd accident and proceeding to show that even this hypothetical case leads to no fatalities or substantial hazards to the public has become a standard method for showing that the more "probable" and less serious accidents will not be hazardous to the public.

The value of maximum step reactivity addition was arrived at by considering the instantaneous removal of the three-blade control assembly with the greatest worth from the critical reactor. No method is postulated for the instantaneous removal.

Other accidents are considered which have a finite probability. These are loosely classified as "credible" accidents even though all of them require multiple failures. Considering these accidents leads one to conclude that all "credible" accidents are the result of reactivity additions of less than 2 per cent $\Delta k/k$.

Failure of a Prototype Control Blade

The "poisoned" core will mock-up 11 to 13 of the GCRE control blades. These blades will not have drive units and will be moved manually. If a blade were fractured, part of the blade might fall out of the core region, causing a reactivity increase. The removal of the most important of these blades represents an estimated 2 per cent reactivity increase which could be produced as a rapid ramp addition if it fell to the bottom of the core through the water.

Fracturing a blade in operation due to internal stresses is not likely since the blades are handled enough during grinding, shipping, and assembling to pretest them to a certain extent. These blades are mounted with their small dimensions horizontal and will have a negligible bending movement due to their own weight. The upper support structure protects them from falling objects.

Addition-of-Water Accident

The reactivity changes associated with various water heights in the core are shown for a typical reactor configuration in Figure 15. It can be seen that with the water height at 20 in. (8 in. from the top of the core), the rate of reactivity change is 0.04 per cent per sec. This is the normal operating limitation of the system. When the water is at

this height the core would be 5-1/2 per cent subcritical if the core was just critical with the water at full height. With the water at half height, at 14 in., the rate of reactivity change on adding water is 0.09 per cent per sec, a little more than twice the normal operating limitation.

To have any accident caused by raising the water level, it would be necessary to have the core overloaded so seriously that criticality is attained with the water at less than 8 in. Even then the interlocks would have to be inoperative to permit adding water with the rods withdrawn, the operator would have to be completely inattentive to the instruments, and the scram systems would have to fail. However, if all of these improbable conditions were fulfilled at the same time, the resulting accident would be a slow excursion rather than a fast one.

Withdrawal of Rods

The rate of withdrawal of control rods will be adjusted so the maximum change in reactivity will be 0.04 per cent per second. This value is the peak value and occurs over only a small range of rod travel. The rate of increase of reactivity will diminish with continued rod motion and will take approximately 30 sec to insert 1 per cent reactivity into the core.

The withdrawal of more than one rod is not possible in normal operation, since a selector switch chooses the rod to which power is applied. Furthermore, the system is fused to fail if more than the rod drive motor is connected. However if all six rods were ganged, a maximum rate of increase of reactivity of less than 0.24 per cent per sec would occur. It is shown in Appendix D that this ramp-rate reactivity addition corresponds to an instantaneous addition of 0.2 per cent $\Delta k/k$. To achieve an accident it would be necessary to have the core overloaded so that it would go critical with these six rods in their most effective region and would require the failure of interlocks, operator, and safety circuits.

Flooding of a Fuel Element

Each fuel element is individually sealed, as shown in Figure 6, in an aluminum can which has a welded bottom. Therefore the simultaneous and sudden flooding of several elements is so highly improbable that it will not be considered.

The failure of an aluminum weld so that a single fuel element can be flooded is a possibility and procedures are established to check for this.

The most likely condition is for an element to flood so slowly that the operator corrects for the reactivity drift and it is not noticed immediately during normal operation. The sudden flooding of an element is not anticipated.

However, the reactivity effect of flooding of an element has been calculated and is estimated to have a maximum effect of 0.3 per cent in reactivity. This is less than prompt critical and would not cause a serious problem.

FIGURE 20. NUCLEAR ENERGY RELEASE

Dropping Fuel Element Into the Core

In the measurement of reactivity coefficients, fuel elements may be handled by a remotely operated handling device. This device will remove and insert fuel elements into various locations along one radius in the core. However in its operation, it will not be possible to have more than one element removed from the core at any one time. It will also be necessary to have a control rod inserted so that the core will be 1 per cent or more subcritical while the change is taking place.

If a fuel element were dropped off the changing mechanism into the reflector region immediately adjacent to a critical core, it could add as much as 0.3 per cent reactivity. If the changing mechanism failed so the element being tested was allowed to drop free into the core, the increase in reactivity would be about 0.45 per cent. This figure assumes that a double-loaded fuel element is dropped instantaneously into a hole vacated by a standard element. The reactor is not subcritical by the required 1 per cent rod prior to the exchange. This requires a sequence of operator errors and interlocks malfunctioning.

These additions would actually be a fast ramp addition since the element would be falling under gravity and retarded only by the resistance of the water. However, for analysis purposes it can be considered a step function. These accidents amount to less than 20 per cent of the reactivity change which occurs in the maximum accident which has been postulated.

Such accidents will be prevented by rigidly fixing the elements to the changing mechanism and by filling the unused holes in the spacing plates with plugs. The element spacing plate and supporting ring cover the reflector area immediately adjacent to the core. In addition, the interlock requires that the reactor be 1 per cent subcritical and neither accident adds sufficient reactivity to bring the reactor to a critical state.

Energy Releases

The method for calculating the energy released in a nuclear accident is given in Appendix D. The shutdown mechanism was assumed to be loss of vaporized fuel from the system. The calculations, used to obtain the curves of Figure 20, assume unrestrained expulsion of the vapor from both ends of the fuel tube. A correction for the restraint imposed by the Teflon end cap (see Figure 6) and for expulsion of the fuel vapor from only one end of the fuel tube is also discussed in Appendix D. This correction was calculated for only the maximum hypothetical accident and the result plotted on Figure 20.

For the hypothetical accident, Figure 20 predicts a nuclear-energy release of 55 megawatt-sec. This value is estimated to be 120 megawatt-sec if the fuel escapes from only one end of the fuel tubes. For average atmospheric conditions an accident of 120 megawatt-sec would result in a direct irradiation dose of only 3.5 r (Figure 16) at the exclusion-area limit. The dose acquired in infinite time due to fall-out would be 120 r at the exclusion limit and 11 r at the nearest village (see Figure 17).

Since the maximum hypothetical accident is considerably less than 4000 megawatt-sec, the tolerance dose for inhalation of iodine-131 is not exceeded at the nearest village.

APPENDIX A

LETTERS ON EARTHQUAKES

APPENDIX A

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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

Enclosure

XAVIER UNIVERSITY
CINCINNATI 7, OHIO

C
O
P
Y

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

APPENDIX B

METEOROLOGY REPORT

APPENDIX B

METEOROLOGY REPORTIntroduction

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, 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 B-1 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

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

TABLE B-1. PERCENTAGE FREQUENCY OF WIND DIRECTION, WEATHER BUREAU AIRPORT
STATION, COLUMBUS, OHIO

(Based on hourly observations January, 1943, through December, 1953)

Wind Direction	Winter	Spring	Summer	Fall	Annual (All Observations)	Annual (8 am - 4 pm)	Annual (5 pm - 7 am)	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

TABLE B-1. (Continued)

Wind Direction	Winter	Spring	Summer	Fall	Annual (All Observations)	Annual (8 am - 4 pm)	Annual (5 pm - 7 am)	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

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 B-1 also compares the wind frequencies for two periods of the day - 8 am to 4 pm and 5 pm to 7 am. 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 periods of precipitation in order to consider the effect of wash-out of possible waste contaminants. Table B-1 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 B-2 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 to 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,

TABLE B-2. PERCENTAGE FREQUENCY OF WIND-SPEED GROUPS, WEATHER BUREAU AIRPORT STATION,
COLUMBUS, OHIO

(Based on hourly observations January, 1948, through December, 1953)

Wind Speed, mph	Winter	Spring	Summer	Fall	Annual (All Observations)	Annual (8 am - 4 pm)	Annual (5 pm - 7 am)	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	0.7	1.0	(a)	(a)	0.5	0.7	0.4	0.9
32-46	(a)	(a)	(a)	(a)	(a)	(a)	(a)	(a)
46	-	-	-	(a)	(a)	-	(a)	(a)
Mean wind speed	9.7	9.7	6.4	7.9	8.4	10.5	7.1	10.8

(a) Only a few observations.

but a rare severe storm may cause winds in excess of 50 mph, 1 to 3 in. of rain in an hour, and hailstones 1/2 in. 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 four tornadoes in the immediate Columbus area.

Precipitation

The Columbus area receives approximately 38 in. of precipitation annually, which is 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 hr was 3.91 in. Columbus has an average snowfall of 22 in., which falls on approximately 6 days per year. The greatest amount ever recorded for a 24-hr period was 11.9 in., and for 1 month was 29.2 in.

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 a 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.

APPENDIX C

CONCLUSIONS OF GEOLOGY AND HYDROLOGY REPORT

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).

** BMI-ACRS-600.

"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."

APPENDIX D

HAZARDS CALCULATIONS

APPENDIX D

HAZARDS CALCULATIONSDose 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) (megawatt-sec)

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

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 , curies per megawatt-sec

σ = cloud radius at distance X_o , m

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

*Fitzgerald, J. J., Hurwitz, H., Jr., and Tonks, L., "Method of Evaluating Radiation Hazards From a Nuclear Incident", KAPL 1045 (1954).

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

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 $\approx 2.66t^{-1.2}$ mev per sec-fission for time between 10 sec and 100 days after shutdown. Hence, the rate of energy release in the cloud per megawatt-sec is

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

or

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

where

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

Hence,

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

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

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

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

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

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

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} , $\text{Sr}^{90} + \text{Y}^{90}$, $\text{Ba}^{140} + \text{La}^{140}$, $\text{Ce}^{144} + \text{Pr}^{144}$, 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

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

where

J = inhalation rate, 17 liters per min

v = cloud velocity, m per sec

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

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

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

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

*Landahl, H. D., and Tracewell, T. N., "On the Retention of Air Borne Particles in the Human Lung, II", 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.

$$\begin{aligned}
 Q &= f e^{-\lambda \left(\frac{X}{V} \right)} \lambda \frac{3 \times 10^{16}}{3.7 \times 10^{10}} \frac{\text{curies}}{\text{megawatt-sec}} \\
 &= 8.11 \times 10^5 f e^{-\lambda \left(\frac{X}{V} \right)} \frac{\text{curies}}{\text{megawatt-sec}} .
 \end{aligned} \tag{D-7}$$

Table 4 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 (D-6), where

$$Q = 2.16 \times 10^6 \left(\frac{V}{X} \right)^{1.2} \frac{\text{curies}}{\text{megawatt-sec}} . \tag{D-8}$$

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 by fuel vaporizing and flowing out the ends of the fuel tubes and, hence, being lost from the system. 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 30 kg of fuel, the energy required to reach the boiling temperature is 20.3 megawatt-sec. An additional 49.5 megawatt-sec would vaporize the fuel.

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

$$\nu(t) \bar{\rho}(t) - \frac{N(t)}{A} + L \frac{\partial \bar{\rho}(t)}{\partial t} = 0 \tag{D-9}$$

$$\nu^2(t) \bar{\rho}(t) - 1/3 g P_o(t) + L \left[\frac{\nu(t)}{2} \frac{\partial \bar{\rho}}{\partial t} + \frac{\bar{\rho}(t)}{2} \frac{\partial \nu(t)}{\partial t} \right] = 0 \tag{D-10}$$

$$2/3 \frac{C_p}{R} \nu(t) P_o(t) + \frac{1}{2Jg} \nu^3(t) \bar{\rho}(t) - \frac{h\ell}{A} N(t) - \bar{\rho}(t) \bar{Q}(t)L \quad (D-11)$$

$$+ L \left\{ 8/9 \frac{C_v}{R} \frac{\partial P_o(t)}{\partial t} + \frac{1}{2Jg} \left[\frac{\nu^2(t)}{4} \frac{\partial \bar{\rho}(t)}{\partial t} + \frac{\bar{\rho}(t)\nu(t)}{2} \frac{\partial \nu(t)}{\partial t} \right] \right\} = 0$$

$$\Delta k(t) = \Delta k_o - 2A\alpha \int_0^t \bar{\rho}(t')\nu(t')dt' \quad (D-12)$$

$$N(t) = \frac{\bar{Q}(t)}{C_\ell} \left[\frac{M_o}{2} - \int_0^t N(t')dt' \right] \quad (D-13)$$

$$\frac{d\bar{Q}(t)}{dt} = \frac{\Delta k(t)}{\ell} \bar{Q}(t) \quad (D-14)$$

$$H = \int_0^t Q(t') \left[M_o - 2A \int_0^{t'} \bar{\rho}(t'')\nu(t'')dt'' \right] dt' \quad (D-15)$$

t = time after vaporization begins, sec

$\nu(t)$ = velocity of gas leaving the core, ft per sec

$\bar{\rho}(t)$ = average density of gas in core, lb per ft³

$N(t)$ = rate of gas formation, lb per sec

A = cross-sectional flow area for gas, 0.417 ft²

L = half length of fuel tube, 1.29 ft

$P_o(t)$ = maximum pressure in core, lb_f per ft²

C_p = heat capacity of gas at constant pressure, 3.34×10^{-2} Btu/(lb)(F)

C_v = heat capacity of gas at constant volume = 2.59×10^{-2} Btu/(lb)(F)

R = gas constant = 5.78 ft - lb_f/(lb)(F)

h_ℓ = enthalpy of gas at the boiling point, 1007 Btu per lb

$\bar{Q}(t)$ = average rate of heat generation per unit mass, Btu per lb-sec

g = gravitational constant, 32.2 lb - ft/(lb_f)(sec²)

$J = 778 \text{ ft-lb}_f \text{ per Btu}$

$C_\ell = \text{heat of vaporization, } 715 \text{ Btu per lb}$

$\alpha = \text{reactivity coefficient of fuel mass, } 6.07 \times 10^{-3} \text{ per lb}$

$\Delta k(t) = \text{prompt excess multiplication at time } t = k_{\text{eff}}(1-\rho) - 1.$

$k_{\text{eff}} = \text{effective multiplication factor}$

$\Delta k_0 = \text{initial value for } \Delta k(t)$

$M_0 = \text{initial fuel loading - taken as } 66 \text{ lb}$

$H = \text{total nuclear energy of the excursion, Btu}$

$\ell = \text{neutron generation time} = 3.5 \times 10^{-5} \text{ sec.}$

Equation (D-9) is the continuity equation. Equation (D-10) is a momentum balance, and Equation (D-11) is an energy balance. In Equations (D-10) and (D-11) the pressure dependence in space was taken from Mills.* In all three equations the velocity is assumed to vary linearly in space, and the perfect gas law has been applied.

Equation (D-12) relates the loss of fuel to the reactivity.

Equation (D-13) relates the rate of vaporization of fuel to the power level and the amount of fuel already vaporized. It is assumed that only fissions which occur in the solid phase are effective in vaporizing fuel. Fissions which occur in the gas are assumed to increase the energy of the gas.

Equation (D-14) relates the power to the prompt excess multiplication. Delayed neutrons are neglected since it is prompt critical conditions which are of primary interest.

Equation (D-15) gives the total nuclear energy of the excursion.

Equations (D-9) through (D-15) were solved by an electronic operational analogue. The initial conditions were chosen so that for a given value of Δk_0 , the power level was such that 20.3 megawatt-sec had already been supplied, i. e.,

$$Q_0 = \frac{20.3}{M_0} \frac{\Delta k_0}{\ell} \frac{\text{megawatts}}{\text{lb}} = 8.35 \times 10^6 \Delta k_0 \frac{\text{Btu}}{\text{lb-sec}} .$$

*Mills, M. M., "On the Hazards Due to Nuclear Reactors", Reactor Sci. & Technol., 1 (1), 55 (1951).

The transient was terminated when $\Delta k(t) = 0$, and the fission energy of the transient was obtained. The nuclear-energy release is plotted against Δk_0 in Figure 20.

It was assumed in developing Figure 20 that the shutdown is achieved by fuel vaporization and free expulsion from both ends of the fuel tube. The driving force for the expulsion was considered to be supplied by nuclear energy alone and all the energy generated was assumed to go into fuel heating. This is a reasonable assumption since the cases leading to large nuclear-energy releases are the step and fast ramp reactivity additions where the power rises very rapidly (in a few milliseconds), large temperature gradients are built up, and heat transfer is negligible during the time of the power rise and shutdown. Consideration of the driving force from vaporization of structural materials and from the chemical-energy release would unnecessarily complicate the analysis. These latter forces will clearly facilitate fuel expulsion and thus cause a more rapid shutdown, but do not contribute to fission-product activity. The inclusion of chemical energy would reduce the calculated magnitude of the accident since no credit is taken for building containment and 100 per cent fission-product release has been assumed. Thus, even though the total energy release may be greater if the chemical energy is considered, this energy can contribute nothing further except to expel fuel from the tube.

The assumption that shutdown occurs because of free expulsion of the fuel vapor from both ends of the tube may be a conservative one. Accordingly, corrections were developed for the maximum hypothetical accident case to estimate what effect the pessimistic assumption of fuel expulsion from only one end may have. These corrections were based upon the following reasoning: the accident proceeds along the energy release-time scale curve of the previous calculation until the end cap, which seals the tube, is lifted enough so that it offers no resistance to gas flow. During this period no fuel is assumed to escape from the tube. After the cap is lifted free, the fuel-vapor release from one end proceeds at one-half the rate as had been calculated when both ends were assumed to be open. The equation of motion of the end cap included an inertial-resistance term and a friction term proportional to the velocity. The driving force was taken to be the time-dependent pressure buildup in the fuel tube. The solution showed that in a time on the order of 10^{-4} sec the cap would be lifted free of the fuel-vapor flow and the accident could proceed as outlined. At this point, only half the excess reactivity had been removed at the time corresponding to shutdown in the previous calculations. The accident was taken to proceed at a constant fuel-vapor removal rate until the remaining fuel was expelled. This approximately doubled the time of the accident. These calculations resulted in the integrated energy release of 120 megawatt-sec which appears on Figure 20.

Solutions to the coupled-energy, momentum, and reactor-kinetics equations indicate that the rate of fuel-vapor expulsion increases very rapidly as the accident proceeds. Therefore, the time interval required to expel the shutdown quantity of fuel from one end of the tube is certainly less

than twice the time taken to expel this amount from both ends. It follows that the time-doubling assumption is quite conservative. Assuming the fuel to leave only one end of the tube is also conservative, because during the excursion, some of the uranium vapor will be forced into the lower, fuel-free region of the tube. Consequently, a situation between free expulsion of the fuel vapor from both ends of the tube and from only one end can be assumed.

APPENDIX E

OPERATIONAL AND LABORATORY RULES

APPENDIX E

OPERATIONAL AND LABORATORY RULES

The following critical-assembly and laboratory operating rules have been distributed to all persons directly associated with the GCRE critical experiments. In addition, these rules are posted in the critical-assembly control room.

Critical-Assembly Operational Rules

On days of reactor operation these rules must be followed unless written permission for exceptions are posted with this notice.

- (1) Three people familiar with the facility to be operated or with reactor operation in general must be present during operation. Two of these must be qualified operators of the particular facility being operated.
- (2) Established checkout procedures must be followed and a check list signed off on operating days.
- (3) A minimum of three power levels and one period scram is to be in operation.
- (4) There must be a visual indication of power level on at least two instruments while approaching critical.
- (5) Reactivity must not be changed faster than 0.04 per cent per sec. Mechanical and electrical controls are to be adjusted to meet this condition.
- (6) Outside doors in the CA room are to be locked and sealed during operation.
- (7) All operators must be in agreement before proceeding with an experiment.
- (8) Data on which operating decisions are to be made must be plotted.
- (9) Each person assigned to the project and present in the control room during an experiment must be engaged in activities which are directly connected with the current operation.

- (10) The doors dividing the control room, vault, and critical assembly room from the rest of the building must be locked during any experiments.
- (11) Radiation warning signals must be posted prior to any reactor experiments.

Critical-Assembly Laboratory Rules

To be followed at all times unless written permission for exceptions are posted with this notice.

- (1) No core will be assembled which will go critical by operating rods alone.
- (2) Anytime fuel is handled in the critical-assembly room a physicist familiar with the handling procedures must be present.
- (3) Any time the vault is open a physicist familiar with fuel-handling procedures must be present.
- (4) There is to be a minimum of two physicists present at any time the assembly room or vault is open.
- (5) No one is to be in the assembly room when the reactor halves are assembled or water is in a reactor tank. When people are in the assembly room the reactor tables are not to be moved more than 6 in. from their full out position or water is not to be pumped into a reactor tank.
- (6) A detailed written plan of each experiment is to be presented to the scientist in charge and approved before changes are made in the reactor or criticality is attempted.
- (7) No fuel additions or rearrangements are to be made without adequate checks on the multiplication.
- (8) No flammable liquids or gases are to be left in the assembly room or vault unattended.
- (9) Security personnel are not to enter the assembly room or vault to fight fires unless a physicist of the operating crew is present.

APPENDIX F

DESCRIPTION AND OPERATION OF THE REMOTE
FUEL-ELEMENT EXCHANGING APPARATUS

APPENDIX F

DESCRIPTION AND OPERATION OF THE REMOTE
FUEL-ELEMENT EXCHANGING APPARATUSDescription

The mechanical operation of the remote fuel-element exchanging mechanism can be described by referring to the photographs of Figures F-1 and F-2. The apparatus shown is essentially a block-and-tackle arrangement to raise and lower four elements in line. A template over the reactor core which has a single hole in it large enough for an element to pass through allows one of the four elements being lowered to enter the core. The three other elements will stop their downward travel when they contact the template. Translating the carriage along the core radius will bring any of the four elements over this hole so that each has access to the core with this mechanism.

The vertical motion of each element on the apparatus is guided by vertical rods for the full length of its travel. This prevents the possibility of the elements tilting or falling over onto the core. The carriage is translated along the horizontal guide rods. The double-pulley arrangement (one above the other) on these rods prevents the carriage from swinging.

The major components of the apparatus are labeled on Figure F-2; their functions will be fully described in the section "Operation" below.

The apparatus will be supported rigidly from the reactor-stand superstructure. The vertical rods extending up from the exchange mechanism will be bolted (3/8-in. steel bolts) to 3-in. channel iron which is fastened to the 8-in. cross beams of the superstructure. Additional cross bracing of the apparatus to the reactor stand will be provided to prevent any sway. This cross bracing is required more for the purpose of facilitating fuel-element insertion and removal than for safety, but does provide double protection against the possible failure of any primary vertical support.

Heavy structural frame and support members are being used in the construction of the fuel-element exchanging apparatus. The main vertical supports are 3/4-in. stainless steel rods. The translating-carriage frame is 3/4 by 1/4-in. cold-rolled steel, and rolls on 3/4-in. drill rod. The elements are suspended from a 1/4 by 2-3/4 by 22-in. aluminum platform reinforced by two 1-in. aluminum angles. The fixed pulleys are supported by 3/16-in. dowel wire, while the pulleys attached to the elements are 1/4-in. dowel pins. A flexible cable with a test strength of at least 100 lb will be used to raise and lower the elements.

Operation

Figure F-3 is a schematic of the electrical circuit associated with the operation of the fuel-element exchanging mechanism. The major interacting parts of the circuit are (1) the source and rod interlocks, (2) the carriage-translating-drive motor, (3) limit switches, (4) the fuel-element-hoist motor, (5) the indicating lamps, and (6) the

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FIGURE F-1. FUEL-ELEMENT EXCHANGING APPARATUS

N47969

1. Horizontal position-locking pin
2. Carriage motor
3. Horizontal position indicator
4. Hoist motor
5. Vertical-position indicator (Veeder limit gear)
6. Upper-limit switches
7. Fuel elements (typical)
8. Vertical guide rods
9. Tension switch
10. Horizontal guide rods
11. Carriage-limit switch

FIGURE F-2. DETAILS OF OPERATING MECHANISMS OF FUEL-ELEMENT EXCHANGER

FIGURE F-3. ELECTRICAL SCHEMATIC OF REMOTE REACTIVITY MECHANISM

horizontal-position-fixing solenoid. The operation of these items along with the operation of the components indicated on the photographs of Figures F-1 and F-2 can best be described by following through one complete cycle of unloading and loading elements into the core.

The normal operating procedure to be followed using the element exchanger is to have a standard core element and three modified elements fixed in the exchanging apparatus via the cable and pulley which appear in Figures F-1 and F-2. The core is set up initially with the standard element completely inserted into the core. The reactor is made critical and the position of a calibrated control rod is noted. In this condition the neutron sources and control blades will be out of the core and the interlocks (Figure F-3) prevent driving either the carriage or hoist motors. Next the reactor is shut down.

Before exchanging elements, the source and rod interlocks require that the instruments be "on-scale" and that the reactor is shutdown by at least 1 per cent $\Delta k/k$. In addition, the rod interlock requires that enough rods be out of the core to give 2 per cent $\Delta k/k$ for shutdown safety. Closing this source and rod interlock permits the 24-v dc to be applied to the hoist motor so that the element in the core may be removed. However, in this example the carriage-translating motor cannot be operated since the initial conditions of operation had an element fully in the core and thereby had an upper-limit switch disengaged.

Operating the hoist motor to remove an element supplies tension to the cable which breaks a contact and extinguishes the "bind" lamp. The bind lamp indicates a relaxed cable and does not relight until the next element is fully in the core and the tension on the cable is again relaxed.

When the element is completely withdrawn, all the elements on the carriage eventually contact their respective upper-limit switches simultaneously. This turns off the hoist motor and lights the up lamp. The simultaneous contact of all the elements with the upper-limit switches allows the carriage to be translated so that a new element may be positioned over the hole in the template which is rigidly in position over the core.

Locating the next, modified, fuel element over the hole in the template is automatic. Once the carriage translation has been initiated (by push button at the reactor console) and the carriage moved sufficiently to close the microswitch located on the upper horizontal guide rod one element space behind the pinning solenoid, the carriage moves until the microswitch drops into the next groove. This cuts the power to the translating motor and releases the solenoid pin. The carriage coasts until the solenoid pin drops into the next hole (spaced along the guide rod at separation distances equal to the pitch of fuel elements along the test radius). This closes the solenoid-in limit switch so that current may now be supplied to the hoist motor. The horizontal-position-indicating Selsyn units are shown in the photograph and schematic. This position indicator provides the operator with accurate information regarding the relative location of the elements and the access hole to the core. This is intended to prevent lowering the wrong element into the core which, even though it is not unsafe, could not be tolerated experimentally.

Inasmuch as the source and rods are still interlocked, a modified element may now be lowered into the core by switching the hoist motor to the down position. The elements leave their upper-limit positions. This breaks the upper-limit switches, the up lamp is extinguished, and the translating motor is incapacitated (see Figure F-3). As

the hoist motor unwinds, all but one of the elements stop their downward travel when they contact the template plate over the core. The test element enters the core through the one hole in the template and is seated in position, closing the lower-limit switch and removing tension from the cable. This lights the down lamp and the bind lamp, stops the hoist motor, and closes the contact to the rod interlocks to allow the rods to be withdrawn. The reactor may now be taken critical and the effect of the new element investigated.

In addition to the up, down and bind lamps which indicate the vertical positions of the element, a reversing Veeder-Root counter is driven by a cam located on the exchange apparatus. Schematically, this is denoted the "Veeder-limit" switch on Figure F-3. Intermittent pulses produced by the rotating cam and Veeder-limit switch cause the counter register to turn. Since the counter is a reversing type, both insertion and withdrawn positions are monitored. The operator may couple the vertical-position indication with the bind lamp lighting to indicate that the element is stuck midway in its downward travel if this should ever occur.

In order to arrive at a critical condition by moving control rods with an element out of the core and poised above a vacant hole, both an operator error and a malfunctioning of an interlock would be required. This is so because the lower-limit switches (interlocked to the control rods in the core) will be disengaged and the control rods cannot be withdrawn. The limit switch must fail and the operator must disregard the lamps and position indicators which show that the fuel element is not in the core.

To summarize the safety features of the device:

- (1) Safety rods with a total reactivity worth of 1 per cent $\Delta k/k$ or more must be in the core and safety rods worth at least 2 per cent $\Delta k/k$ must be poised above the core before this apparatus can be operated.
- (2) The neutron source must be in the core so that the instruments are in an operating range before the apparatus may be operated.
- (3) Only one element can possibly enter the core at a time.
- (4) The safety rods cannot be withdrawn unless the test element is fully in the core. This prevents reaching a critical reactor condition with an element poised above the vacant hole in the reactor.
- (5) The carriage motor cannot be moved without all the elements fully withdrawn and engaging the upper-limit switches. This feature insures that the cable is tightly engaged to all the pulleys before an element is moved over the hole in the template and also that the element will not be pulled on a bias while still in the core.
- (6) The positions, both horizontal and vertical, of the fuel element being exchanged are relayed to the operator by lamps and by special position indicators.