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HAZARDS SUMMARY REPORT ON THE
OXIDE CRITICAL EXPERIMENTS

by
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Reactor Engineering Division

April, 1957

Operated by The University of Chicago
under
Contract W-31-109-eng-38

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ACKNOWLEDGMENTS

The authors of this report have been aided appreciably in its preparation by several members of the scientific staff. The procedures presented in Section IV are based on work by S. G. Kaufmann in planning for the actual operation of the facility. H. K. Richards performed some of the reactor safety calculations. J. Haugsnes contributed the appendix treating the external radiation hazard associated with a reactor disaster. R. O. Brittan and B. I. Spinrad have reviewed this report while in draft form and their experience and comments on problems related to critical experiments have contributed much to the improvement of the presentation.

Initiation of a program of critical experiments requires extensive efforts by many individuals. The authors are particularly appreciative of the contributions made by D. Nicoll as the design engineer in the formative stage of this project and by C. A. Pesce and R. L. Ramp on instrumentation and control problems. The contributions of members of the Metallurgy and Electronics Divisions, Central Shops and the Special Materials Department are gratefully acknowledged.

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I. SUMMARY

This report describes a new zero-power reactor facility, designated as ZPR-VII, which is under construction in the critical assembly building D316 at the Argonne National Laboratory. This facility will be used to supply physics design information required in connection with several phases of Argonne's boiling reactor program.

Also included in this report is a description of the first reactor system to be investigated in this facility and an assessment of the hazards associated with the planned program of measurements.

The first system to be investigated will employ fuel in the form of pellets of a fired mixture of enriched uranium oxide and thorium oxide. These pellets will be approximately $1/4$ in. in diameter, stacked to a height of 5 ft within an aluminum jacket of $5/16$ -in. outer diameter. These fuel rods will be grouped in a variety of arrangements within a $6\frac{3}{4}$ -ft diameter tank which can be filled with heavy water.

The construction, instrumentation, and control system of the ZPR-VII conform with past practice at Argonne. This reactor system is unusually safe as a consequence of large negative temperature and void coefficients of reactivity. Large reactivity effects are virtually impossible to achieve other than through deliberate sabotage, and the inherent safety mechanisms make the reactor completely self-limiting with no damage for a 1% excess reactivity step and probably for a 2% step. In addition, the public hazard associated with an appreciably greater reactivity step is extremely small.

II. GENERAL DISCUSSION

A. Motivation

The history of Argonne's developmental program on boiling reactors is well known. Exploratory studies and experiments by the Laboratory staff on the possibility of constructing a power reactor of simplified design, utilizing steam produced directly in the core, led to the construction of a series of "Borax" reactors at the AEC National Reactor Testing

Station in Idaho. This series of experiments established the feasibility of boiling water reactors and provided some of the data required for the design of the Experimental Boiling Water Reactor (EBWR).

Borax-I, the first boiling test reactor, was an experimental facility in which the stability and self-regulatory aspects of a boiling reactor were investigated. When all of the desired experiments were complete, Borax-I was deliberately destroyed in a transient nuclear reaction. The final destructive test showed that extraordinary measures are necessary to produce a damaging excursion. Tests with Borax-I proved that this type of boiling water reactor is self-regulatory and inherently safe. Borax-I was followed by a larger experimental assembly, termed Borax-II, capable of operating at a greater pressure. This facility became Borax-III when a new core was substituted and a turbine and generator were added in 1955. Borax-I, II, and III provided information on stability and control, corrosion and erosion, radioactive transfer, leakage, and other information required for the design of the EBWR.

EBWR, one of the five reactor types included in the AEC's five-year reactor development program initiated in 1954, has now been placed in operation. It is a reactor built for steady power generation to demonstrate the feasibility and safety of large-scale power plants utilizing the boiling principle. Its present loading is metallic uranium of low enrichment. More recently the Borax experiments have shifted to an investigation of the behavior of cores composed of ceramic fuel elements, with the use of such materials in mind for subsequent loadings in EBWR.

It is clear from the above summary that the boiling reactor program to date has been primarily concerned with engineering tests related to reactor performance. There is a growing need for a better understanding of the physics of boiling reactors, and to date a program of measurements of a fundamental nature has been lacking in the Argonne program on boiling reactors. To provide the physics support necessary for the optimum design of subsequent boiling systems, a related experimental physics program has been initiated.

B. Facilities

The investigation of the physics of boiling reactors will utilize two experimental facilities. One, encompassing exponential or subcritical experiments, is underway in the heavy water exponential facility of the International School of Nuclear Science and Engineering. This program will involve the exploration of a range of concentrations among materials of potential interest in the boiling reactor program. Initially the principal constituents of the exponential systems will be thorium oxide, uranium oxide, aluminum, and heavy or light water. The program will include measurements at reduced water densities and elevated temperatures. The

exponential facility will be used to explore a range of concentrations and numerous geometrical arrangements of these materials. These results will help narrow the range of lattice arrangements and loadings of interest for detailed investigation in connection with the design of a second loading for EBWR.

Critical and mockup experiments are necessary to achieve higher precision for those configurations found to be of special interest and also for the determination of results not readily measurable in exponentials, such as reactor power distribution patterns, effects of non-uniform void distributions, kinetic behavior, control rod effectiveness, importance of streaming, and localized substitution effects. A facility for these experiments is now under construction. It will be the seventh zero-power reactor facility constructed by Argonne, designated as ZPR-VII.

The immediate objective of the program of measurements to be performed in ZPR-VII is the exploration of a second loading for the EBWR composed of uranium oxide, thorium oxide and heavy water. This immediate use has influenced heavily the design details of ZPR-VII, but sufficient flexibility has been incorporated to allow for subsequent research and testing in connection with other reactor systems using light or heavy water. However, this report deals only with the ZPR-VII facility as it will be constructed and used in the initial program of measurements concerned with oxide-heavy water systems.

C. Program with Oxide Fuels

In order to provide a workable design for a second loading of the EBWR, to predict its performance and simultaneously to provide the basic physics information of value for the understanding and design of improved systems, a comprehensive program of experiments is planned. It is recognized that those measurements of a long-range nature may be postponed by the need for operational testing of other systems, but the entire program as presently visualized will be included here for evaluation and approval. For convenience, the program is summarized in terms of two classes of experiments, those concerned with one particular loading and those involving changes in loading.

For a particular loading, the types of experiments of interest are as follows:

1. critical sizes, with and without a top reflector;
2. reactivity variations for a range of temperatures and fraction of voids;
3. worth of control rods for a range of temperatures and voids, measured by rod bump, rod drop and/or distributed poison methods;

4. axial and radial flux distribution measurements and determinations of associated reflector savings;
5. effect of non-uniform dispersal of voids on flux distributions and control rod worth;
6. measurement of the migration area, M^2 , by the poison and/or enrichment methods;
7. determination of lattice cell parameters, such as fast effect, thermal utilization, resonance escape probability and conversion ratios by foil techniques;
8. investigation of neutron streaming and other effects of void inhomogeneity;
9. reactor response to transient voids:
 - a. a "bubble drop," analogous to a rod drop, in which reactivity losses associated with the introduction of an air bubble stream or a control rod of styrofoam into a reactor are measured;
 - b. oscillatory introduction of voids;
10. worth of light water in the vicinity of fuel elements and as an impurity in the heavy water moderator;
11. Doppler effect in thorium;
12. worth of U^{233} and U^{235} in various concentrations;
13. reactivity coefficients of various materials; and
14. neutron spectrum measurements.

The second category of experiments is of that type involving changes in the core loading such as:

1. measurements with various lattice spacings for a given cluster of fuel elements;
2. various oxide-to-water ratios for a particular lattice spacing;
3. effect of uranium enrichment on reactivity; and
4. investigation of multizone cores.

D. Subsequent Use of ZPR-VII Facility

The future program at Argonne will require a continuing study of the physics of water reactors. It is planned that the ZPR-VII facility will be a semi-permanent installation and so all components have been designed for a long life. The possible studies utilizing this facility in the near future include pre-operational testing of the ALPR core prior to its installation in Idaho, generalized studies of uranium oxide-thorium oxide-light water systems, design studies and pre-operational core tests for the ARBOR experiments, criticality studies in connection with the design of a high flux research reactor, and investigation of a system for a third loading of the EBWR.

This Hazards Summary Report is concerned only with the facility as it will be used in the oxide-heavy water critical program. Reports summarizing significant modifications to the facility and changes in the component materials under study will be submitted to the Hazards Evaluation Staff when appropriate.

III. DESCRIPTION OF THE ZPR-VII FACILITY

A. Site and Building

The ZPR-VII Facility will be located in the Critical Assembly Building at the DuPage County, Illinois, site of the Argonne National Laboratory. The regional map (Fig. 1) indicates the location of this building with respect to other buildings in the area. The make-up of the territory surrounding the DuPage site and degree of population of this land is presented in Appendix A.

The system will be constructed in cell number one (1) of the Critical Assembly Building. Figure 2 shows the location of this experimental set-up in relation to the other two reactor cells in this building. Presently cell number two (2) is occupied by the internal exponential experiment, ZPR-V, and cell number three (3), presently under construction, will house the rebuilt thermal source reactor, ZPR-IV'.

Cell number one (1) is a reinforced concrete and brick structure, 30 feet long, 26 feet wide, and 25 feet high. A 3-foot thick shield wall protects personnel in adjacent rooms, and entrance is made through a baffled doorway. An emergency door permits quick exit to the outside. The cell is normally ventilated by a 2000-cfm system. During reactor operation the ventilating system is shut off.

B. Reactor Structure

Figure 3 shows the arrangement of the main components of the reactor structure to be located in the previously described cell. Items 1, 2, 3, 4, 5, 6 and 7 may be seen on a larger scale in Fig. 4.

1. Main Tank

The upper tank section (Item 1) and lower tank section (Item 2) are flanged and bolted together to provide an assembly 12 feet, 8 inches in depth. The inside diameter of both these sections is 81 inches and the wall thickness $1/4$ inch. Bolted to the top flange of the top section is a tank member that provides cover for the beams which support the control rod pulleys. It also supports the reactor tank cover, in which are located several covered openings which will be used for mounting

PARTIAL BUILDING PLAN

FIG. 2
REACTOR CELLS IN
BUILDING 316

SECTION A-A

KEY PLAN

FIG. 3
THORIUM-URANIUM
OXIDE CRITICAL FACILITY

experimental equipment within the reactor tank. The diameter of this member is 94 inches and its length is 7 inches. All the tank sections are protected against corrosion by a baked-on phenolic resin coating applied to the interior surfaces.

FIG. 4
CUTAWAY VIEW OF TANK SHOWING
CORE AND CONTROL MECHANISM

2. Core and Tank Support

The core (Item 6) is elevated one foot above the bottom of the reactor tank and rests on a 3/8-inch aluminum bed plate. The load on the bed plate is transmitted through an egg-crate grid structure (Item 7) to a circular ring to which the side walls and bottom plate of the reactor tank are welded. The ring and bottom plate rest on a similar ring of a pedestal (Item 25) which carries the load to the floor. The plate on the pedestal is reinforced by gussets to insure a flat surface for the bottom of the reactor tank.

3. Core for the Oxide Critical

The core for the oxide critical is a heterogeneous assembly of fuel rods held in element clusters which are positioned by holes in the bed plate. The fuel is a uniform mixture of uranium oxide (highly enriched U^{235} as either U_3O_8 or UO_2) and thorium oxide (ThO_2) which has been molded and fired to make pellets of high density. These pellets are between 0.230 and 0.236 inch in diameter and have a length of approximately 0.600 inch. They are inserted into an aluminum tube, 0.312 to 0.306 inch outside diameter and 0.246 to 0.241 inch inside diameter, to form a core length of 60 inches. Prior to insertion of the pellets, the lower end of the tube is sealed with a support adapter. An end adapter for the removal of the elements from the fuel cluster is brazed into the tube after insertion of the pellets. Approximately six thousand fuel elements will be available for insertion into the core. Seven thousand void elements will also be available. The void tube has the same length and outside diameter as the fuel element tube; however, the wall is thinner (0.014 inch). The lower end is closed by an adaptor brazed in the same manner as the fuel element. The upper end is plugged by an "O" ring-fitted stopper. This stopper, being removable, provides access to the inner cavity of the void element so that the water content may be varied as required by the nature of the experiment. The diameter of the stopper was determined by test procedure so that vibration cannot cause it to work free. The buoyancy of the void elements is counteracted by a restraining grid placed above these tubes and attached to the fuel clusters.

The fuel cluster (Fig. 5) consists of a support plate for the fuel elements and two aluminum plates perforated with holes that are spaced in a triangular pattern 3/8 of an inch between centers. The four corner holes are fitted with parts which unite the support and separating plates into a semi-rigid construction. The cluster dimensions are 3-7/8 inch square by 63 1/2 inches long. Up to one hundred and one fuel elements can be placed in a fuel cluster. These clusters permit a very flexible arrangement for the fuel and void elements. Various lattice spacings are available at the discretion of the experimentalist.

FIG. 5
FUEL CLUSTER

Holes for the flow of water and the positioning of fuel element clusters are provided in the bed plate (see Fig. 4). They are in a square array with 4-inch spacing. Two hundred fifty-five clusters can be located within the reactor tank. One hundred forty-eight clusters are required to match the outline of the present EBWR loading, which is slightly less than 5 feet in diameter.

C. Reactor Control

1. Control System

For the safety of the reactor, two events are to be initiated simultaneously to provide emergency shutdown. These are the rapid insertion of the control rods and dumping of the moderator through a quickly opening valve.

The core must be accessible for experimental purposes. Therefore, it is desirable to keep the top of the reactor clear of control rod mechanisms. This has been done by mounting the control rod drives on brackets supported at the periphery of the upper tank section. A separate drive is provided for each control rod.

Two groups of three rod-drive units are mounted diametrically opposite each other, and an additional three units are located in the other quadrants (see Item 5 of Fig. 3). Two of these latter three units are mounted on the same side and are spaced 75 degrees apart, while the other unit is mounted on the opposite side. The center unit of each three-group mounting is elevated above the other two drives. These units are equipped with an adjustable pulley bracket in order to accommodate this variation in mounting height.

Below the control drives and under the reactor tank cover is an arrangement of beams for support of guide pulleys that control cables attached to the control rods. All guide pulleys are fitted with close guards that prevent the cable from becoming wedged between pulley and support should the cable tend to bounce during a scram of the control mechanism (Fig. 4). The pulley guide beams are bolted to the upper tank flange. The beams are $3/4$ inch wide, corresponding to the width of the control rod guides, and are arranged on $12-3/4$ -inch centers for the initial experiments. Clearance holes are drilled in the beams for the passage of the drive cables. When the cover is removed, the reactor can be loaded without any interference from the control rod drives or pulley arrangement except for the cables that are directly over the rods.

2. Control Rod Drive (Fig. 6)

The rod drive unit was originally designed and used for operation with the ZPR-I. Except for a renovation and redesign of the

FIG. 6
CONTROL ROD DRIVE MECHANISM

position-indicator mechanism and an increase in the length of stroke, this control rod drive unit is the same as that used successfully in several critical experiments at Argonne during the past seven years.

The 1/8-inch diameter cable attached to the control rod passes over a series of pulleys through two horizontally mounted guide pulleys to a drum which is coupled to a gear motor through an electromagnetic clutch. The drum is mounted on ball bearings and is free to rotate on the shaft when the clutch is de-energized automatically at scram, or by power failure or at will by the operator. With the clutch de-energized, the control rod falls with an acceleration of about one "g" (16 ft/sec²). The rod is decelerated smoothly when very near the bottom position by bringing a Houdi-type shock absorber into action as a snubber. The shock absorber is coupled with the drum by a cable loop system, one end of which winds up while the other unwinds as the drum turns. When one end is unwound it starts rewinding so that both ends of the cable are winding up on the drum and the loop shortens, pulling the shock absorber arm back. The rod comes to rest fully inserted (about 1 inch above the base plate). The cable is capable of snubbing the rod without permanent set after a free fall of five feet, even if the shock absorber fails to operate. Mechanical stops prevent the rod from striking the base plate and also from being withdrawn beyond the rod guide. Limit switches control the power to, and direction of rotation of, the motor. Rod position is transmitted from the unit and received at the control panel by coupled synchros.

The drive motor is 1/30 HP capable of operating at various speeds from a 90-volt, d-c power supply. The gear drive allows a maximum speed of rod withdrawal of 12.9 inches per minute. The drive circuits are so interlocked that only one rod, as determined by a selector switch, can be withdrawn at a time. If the circuits were rewired to allow withdrawal of two rods simultaneously, the limited power available would reduce the speed to one-half that for a single rod.

3. Control Rod (Item 3 of Fig. 3 and Fig. 4)

The ZPR-VII facility is provided with nine control rods that can be arranged into any configuration by relocating the guides on the reactor grid plate and the guide pulleys at the support beams.

The 60-inch control rod to be used in the initial measurements is fabricated in the form of a square cross 10 inches across the blades (see Fig. 4). Cadmium is the neutron-absorbing material and it is sandwiched between 1/16-inch aluminum plates. All seams are welded for water tightness. An attachment for a 1/8-inch diameter cable is made an integral part of the rod.

4. Control Rod Guide (Item 4 of Fig. 3 and Fig. 4)

For the initial program, nine guides, one for each control rod, will be bolted to the reactor grid plate in a pattern corresponding to the rod locations in the EBWR. Five guides are placed on the reactor center lines and the other four are placed at the corners of a square lattice spaced 12-3/4 inches from the center lines (see Fig. 4). They are united into a rigid assembly by connecting links. The height of the guide is 6 feet.

D. Water Circulating System (Fig. 7)

The heavy water is of pure quality, but not degassed. The system will be operated at atmospheric pressure and at room temperature, except for experiments involving temperature coefficient measurements.

FIG. 7
FLOW DIAGRAM

RE-8-19320-B

1. Dump Tank (Item 20)

D₂O is stored in the dump tank located in a pit as seen on Figs. 2 and 3. It is 5 feet 7 inches wide, 7 feet 3½ inches long and 8 feet 4 inches high. The material is mild steel. Interior surfaces are protected against corrosion by a baked-on phenolic resin coating. The storage capacity of the tank is 2500 gallons.

2. Circulating Pump (Item 21)

A seal-less, all stainless steel Chempump, with a maximum capacity of 160 gallons per minute at a head of 40 feet, is submerged in the dump tank. This pump is a centrifugal pump designed for general chemical service with absolutely no leakage or contamination of the pumped fluid. It has no mechanical seal or stuffing box. Lubrication is automatically provided by the fluid being pumped. Any need for maintenance is therefore practically eliminated. It is driven by an integral 3-horsepower, 3-phase, 60-cycle, 220/440-volt, 1750-rpm, canned motor with class "H" insulation which permits a maximum temperature of 450°F in the stator windings. The pump suction nozzle is 2-inch and the discharge nozzle is 1½ inch NPS. The pipe connecting to the discharge is increased immediately to 2 inches.

3. Reactor Feed Lines (Fig. 8)

The circulating pump delivers the water through a manually operated main 2-inch shut-off valve (Grinnel-Saunders Diaphragm Globe type). The flow is directed by a series of solenoid-operated, air-assisted valves, remotely controlled at the panel board.

During initial filling of the reactor tank, the flow is through a 2-inch, solenoid-operated, air-assisted, normally closed valve (ASCO 8336). At this time the water flow is at maximum pump capacity (160 gallons per minute). During this phase of the operation, both the by-pass valve, which is similar to the above valve except that it is normally open, and the small volume, 1-inch, normally closed valve (ASCO 8336) are closed. After the reactor tank has filled to a level determined by the nature of the experiment, the main solenoid-controlled valve is closed and the by-pass and small volume valves are opened. There is a one-inch Globe Needle valve (Crane No. 222X) in the small volume line. This permits the flow to the reactor tank to be adjusted from a trickle to full opening. The flow for the final phase of filling of the tank will be adjusted to less than 10 gallons per minute. Excess flow is recirculated through the by-pass valve to the dump tank. This permits the pump to run continuously or be shut off during an experiment.

A pressure-operated switch in the air-supply line for the solenoid-operated control valves will actuate a light on the control-room panel board. This light indicates to the operator that compressed air at proper pressure is available to operate the control valves.

4. Adjustable Stand-Pipe (Item 8)

The reactor tank will fill to a level predetermined by an adjustable apparatus shown in Fig. 9. The height of water can be varied from 23 inches to 96 inches, measured from the bottom of the tank. The

FIG. 8
REACTOR FEED LINES

rate for varying the height of the adjustable over-flow is 19 inches per minute. During initial assembly, and prior to start-up, an operator can verify the height of the overflow weir by reading two sets of dials on the remotely operated drive mechanism (Item 9) located at the top of reactor. This information is also indicated at the panel board in the control room. The height of the overflow weir, and thus the water level, can be read within 0.02 inch. A sight gauge (Item 26), outside the reactor tank, may be read through a shielded window from the control room to verify the height of water shown by position indicators. During reactor operation, the water level within the reactor tank can be raised or lowered to satisfy experimental conditions. Surplus water passes over the weir and is conducted to the dump tank.

A special open end flange secures the standpipe bellows to the bottom of the reactor tank, while the upper end of the bellows is secured to a flange which forms the weir for water overflow. Spacing bushings connect the weir flange to an acme nut that raises or lowers the bellows. The nut is prevented from rotating by two guide rods that extend from the bottom of the reactor tank to its upper flange. Power is supplied to the acme screw through a double-ended, universal-jointed shaft by a motor-driven gear reduction unit. A fixed $2\frac{1}{2}$ -inch diameter pipe, connecting to the reactor tank, completes the flow circuit at the spool piece, which is located between the dump valve and an elbow at the dump tank.

5. Air Vent Line (Item 13)

A vent line connects the reactor tank with the dump tank and will serve as an overflow when operating at the maximum water level of 8 feet.

6. Dump Valve (Item 24 and Fig. 10)

Water is removed from the reactor tank by tripping the dump valve. This is a "Rockwell" butterfly valve located in the 12-inch discharge pipe below the reactor tank.

A circular vane, hinged along its diameter, seats against a rubber-lined body. It is held closed against the weight of the water above by an electromagnetic clutch secured to a motor-driven gear reduction unit. When the clutch releases, the vane swings down, accelerated by a weighted lever and the force of the water above it.

The operating mechanism consists of a weighted lever (approximately 35 lb at 18 inches) secured to the butterfly valve stem. The electromagnetic clutch armature is secured to the lever and the magnet side of the clutch attaches to the reduction unit shaft. The $1/4$ -horsepower, Janette motor-driven gear reduction unit is capable of delivering 708 inch-lb

FIG. 10
DUMP VALVE AND ACTUATING MECHANISM

of torque to the valve stem. The valve will close through its angular rotation of 78 degrees in 1.1 seconds. Micro-switches, actuated by the lever, control the power to the motor and clutch and operate lights that indicate valve position at the control room panel board.

Power to the motor and clutch is supplied by depressing a control switch at the panel board. Circuits are electrically interlocked with reactor circuits for safe operation. The valve is closed by the motor which rotates the clutch and valve stem until the lever trips the micro-switch which automatically stops the drive. A pre-set mechanical stop prevents the valve vane from being jammed against its seat. The clutch remains energized during operation of the reactor. It is de-energized either automatically at scram, or on power failure, or at will by the operator in the control room.

The water-circulating circuit is completed by a spool piece between the dump valve and an elbow which is bolted to the dump tank. A point of connection has been provided for future installation of an Ion Exchanger System in the event that soluble poison experiments are programmed. This provision may be seen on Fig. 8.

7. Heating of Water

For temperature coefficient experiments the D_2O will be heated by two stainless steel-sheathed, 36-kw immersion heating elements (Item 22) (Chromalox TMI-1236). Experiments at elevated temperatures may make it desirable to heat as high as $210^{\circ}F$. These heaters plus a Fenwall thermoswitch (Item 23) and a float switch (Revere F8300) mounted in the dump tank are electrically interlocked to control water temperature. The float switch protects the heating elements against burnout in case the water drops below a pre-set level. The float switch consists of three major parts: body assembly, float assembly and shell. The body assembly contains the mounting flange, mounting fitting and a hermetically sealed, magnetically-operated Glaswitch, potted in a sealed tube extending through a hole in the float assembly. The float assembly is fabricated from a non-absorbing material, contains two permanent magnets, and is protected by an aluminum outer shell which is designed to prevent random operation caused by sloshing liquid. Nylon-jacketed lead wires of suitable length complete the unit.

A number of thermocouples located in the dump and reactor tanks will be utilized to indicate the temperature of the water during an experiment. These temperatures will be read at the control panel.

E. Start-Up Source and Drive Mechanism (Item 11 and Fig. 11)

A Po-Be source producing about 5×10^7 neutrons per second initially is located below, and on the axis of, the reactor tank. In preparation

for reactor start-up it will be raised into a stainless steel thimble which penetrates the reflector 12 inches to within 2-3/8 inches of the bottom of the core. When withdrawn, it is dropped into an oil-shielded storage well to a depth of 100 cm below floor level. The source travel is limited by micro-switches and is interlocked with the control rod release mechanism so that the rod-actuating clutch cannot be energized initially unless the source is in the inserted position. The source strength is sufficient that, with no multiplication in the assembly, the startup instruments will receive sufficient neutron flux to give a signal well above background (a one-liter BF_3 counter at atmospheric pressure 50 in. away from source will register of the order of 200 counts per sec).

FIG. 11
START-UP SOURCE POSITIONING MECHANISM

F. Instrumentation

Five neutron-sensitive devices and one gamma-radiation detector are provided for operation and control of the ZPR-VII facility. These information channels are shown in block diagram form in Fig. 12. Their functions are as follows:

- a. Two BF_3 proportional counter channels (1,2) including separate amplifiers, HV supplies and scalers will indicate the level of neutron flux during startup operation. The reactor period will be derived from a logarithmic count rate meter during this part of the operation, and the period amplifier will supply a trip signal if the period becomes less than 5 sec. The counting rate information will be both displayed and audible at the control console. As soon as the counters start jamming, the high voltage will be turned off and the startup channels thereby inactivated.
- b. Power level will be monitored through two complete and independent ion chamber-vibrating reed channels (3,4). Information on excessive power level is supplied to a common trip circuit from these units.
- c. One ion chamber circuit (5) containing HV supply and log-period networks will take over the period information as the upper limit of the range of the BF_3 chambers of (a) is approached. It supplies a scram signal to a trip circuit whenever the period becomes shorter than a preassigned value.
- d. A scintillator circuit will supply power information based only on gamma intensity. This circuit has an extremely rapid response and will be used for a high power level trip signal.
- e. Health physics monitors will be distributed throughout the reactor and the control rooms.

The instrument circuits will be powered by the same power line which activates the magnetic clutches on the control rod drive units. Linear and logarithmic level information derived from these instruments will be displayed on strip charts at the control panel. The minimum complement of monitoring channels will comprise 1 period and 2 level circuits for any given operating range. Most nuclear detectors and some of the circuits will be built by Argonne's Reactor Engineering and Electronics Divisions; the rest will be bought from manufacturers with established reputations.

FIG. 12
EBWR - OXIDE CRITICAL
NUCLEAR INSTRUMENTATION

G. Ancillary Devices

With the reactor cover removed, access within the reactor is provided by a ladder that can hang from the upper tank. Two semi-circular rings are located inside the upper tank and provide the means for placing beams for the support of personnel working within the reactor.

An aluminum tube (Item 10) for positioning a neutron detector inside the tank during the initial start-up is bolted to the side of the upper tank. Electrical connections are so placed as to facilitate removal of instrument.

Access to the top of the reactor is provided by a working platform fabricated in an octagon shape (Item 27). Metal stairs lead from the floor to the platform.

A fireman's pole (Item 28) is also provided for quick retreat from the reactor in case of an emergency.

A convenient summary of the mechanical features of the ZPR-VII facility and its cell is presented in tabular form as Appendix B.

IV. REACTOR OPERATION

A. Pre-operational Testing

After the ZPR-VII facility has been assembled, but prior to the time at which fuel elements are available in the reactor cell, a systematic testing of all components will take place. This will include testing the system for water leaks; measurement of water flow rates; positioning, withdrawal, and slow and rapid insertion of the control rods; tests of all valves for proper functioning, including the rate of drainage of water through the dump valve; positioning of the detectors for proper sensitivity; and observation of instrument response during source motion.

After any malfunctioning of the components has been rectified, loading to critical will be simulated, using the element clusters loaded only with void tubes. This will serve as a period for training of the operating personnel and familiarizing them with the operating procedures and safety regulations. Several of the personnel to be associated with the program of the ZPR-VII facility have had significant experience with existing and/or prior zero-power reactors operated by Argonne since 1950.

Only during these pre-operational activities, prior to the movement of any fuel elements into the ZPR-VII cell, will water be pumped into, or contained in, the core tank when the access door is open.

B. Loading Regulations

Fuel will be stored in the assembly room. Fuel rod clusters will be suspended from special supports fastened to the cell walls. The depth of the stored fuel will be no more than two complete clusters. Removal of fuel will be done under the supervision of a designated member of the operating crew. He will be charged with the responsibility of keeping an independent check on the amount of fuel withdrawn and incorporated into the assembly.

The assembly will be loaded according to a loading pattern determined in cooperation with the analytical group. A crew member who has not participated in the loading will make a final check on the agreement between the proposed and the actual loading before it is approved for operation. The approach to criticality shall proceed in stepwise increments. The initial portion will be selected on the basis of knowledge acquired in prior exponential and critical experiments involving the constituents and as agreed upon by representatives of both the experimental and theoretical personnel associated with the program. Subsequent additions will be further guided by multiplications observed in previous steps and will proceed only upon agreement between two qualified operators and objection by no crew member. No change in loading will be made when the core tank contains water.

C. Operating Procedures

Power for operation of the facility may be turned on through the combination of two keys. Only two keys will be in existence, and the number of personnel authorized for possession of a key will be limited. Presence of the key in its proper switch will indicate knowledge and approval of the current operation by the qualified operator supplying the key.

The following checks will precede the daily operation of the assembly:

- a. inspection of the assembly and control system for evident infractions on the safety rules;
- b. functioning of the dump valve checked by use of the manual "scram" button;
- c. period circuits checked as the source is inserted and a period simulated through source approach of suitable speed; and
- d. power trip circuits checked through a combination of detector operation on source alone and a simulated trip signal.

A separate source operated manually will be used for this check on operation of the less sensitive circuits. A safety rod, selected in rotation, will be withdrawn for each test of a safety circuit as an indication of the proper functioning of the safety circuit and rod drive.

Operation of the assembly may not begin as long as interlocks are shorted out. Any exception must be approved in the log book by both of the qualified operators. One crew member will check the cell for the presence of personnel, and the start-up of the reactor will be announced over the intercom system. During the operation at least two crew members will be present, of whom one will be a staff person qualified for possession of an assembly key.

In general, the reactor will be brought to criticality by withdrawal of control rods after the water has been pumped into the core tank. Two safety rods will always be held in a cocked position at a height at least as great as that of the prospective core. When the reactivity worth of the water level is to be investigated, the last step toward criticality will be made through carefully controlled changes in water level. The procedure is outlined below.

When the reactor is brought to criticality on rod motion, the water level will be permitted to rise at its maximum rate while the control rods are in the core and the safety rods are suitably cocked. Water flow will be stopped periodically and the counting rate as function of water level will be recorded and plotted. This will be standard procedure during start-up until it is felt that the worth of the water in inhours per cm of water level is known well enough to give confidence in the safety of an uninterrupted start-up. After the water has reached the maximum height, the control rods will be withdrawn at a rate of no more than 15 ih per sec. Rod motion will be stopped at intervals determined by the operator so that the reactivity of the system may be examined.

When the final step toward criticality is to be taken through adjustment of water level after control rods have been withdrawn, the procedure followed will be this:

The reactor will first be taken to criticality by removal of control rods with the water level fixed at a predetermined height. After criticality has been achieved the water level will be lowered by a small amount and the reactivity restored through the withdrawal of rods. This procedure will be repeated until criticality has been reached with all control rods above the water level. The final adjustments of water level will be made by varying the level at a rate of about $1/20$ of the maximum rate of level rise ($1/20$ of 0.16 inch per sec).

At the end of the working day the reactor keys will be removed from their switches, thereby removing the electrical power to the control system, and stored by their custodians.

D. Interlocks

Three interlocks are provided in the reactor control circuitry as a reminder and check that the established operating procedures are being followed. Provision is made for deliberately bypassing these interlocks in the event of an emergency or non-routine situation. Any such action will be recorded and justified in the log book and approved by two qualified operators. The fact that any interlock is shunted will be indicated continuously by a flashing red light on the control console.

The first interlock insures that power for the start-up of the reactor will not be available so long as the door to the ZPR-VII cell is open. A second interlock is provided which prevents startup before the source has been inserted. A third interlock circuit will require that a minimum of two safety rods be cocked to at least the height of the core under study before power to remove the remaining rods or to close the dump valve is available.

A minimum of three additional interlocks will always be operative. These are associated with the minimum complement of monitoring channels which must be available for operation and control of the facility for any operating range. Two of these will serve to limit the power level of the assembly and the third will limit the reactor period which may exist without shutdown.

Opening the interlocks will remove power from the magnetic clutches and will thus release the safety rods into the assembly, retract the source, and remove the moderating D_2O by opening the dump valve.

V. REACTOR SAFETY EVALUATION

A. General Safety Considerations

The ZPR-VII facility is located in a building where the population density of Laboratory personnel in the vicinity is very low. Only those individuals whose normal work assignments require access to the building or facilities have standing authorization for admittance to the building. Access to the area containing the fuel, reactor cells and control rooms is further restricted. Visitors are admitted to the building only by special authorization and only when there is sufficient reason for their entry. This serves to minimize the distractions and thereby enhance the safety of the operation of the zero-power experiments.

The reactor in its cell will be separated from the operating personnel in the control room by a 3-foot concrete wall and a distance of approximately 20 feet. The radiation level in the control room during normal operation will be surveyed periodically by a representative of the Industrial Hygiene and Safety Division. During the initial startup of this facility, radiation surveys will be made to determine whether any restrictions on operation of adjacent facilities are required. Survey of the radiation level in the reactor room following operation of the assembly will be done as a matter of routine operating procedure by the personnel engaged in the experimental program. In addition, the intensity level associated with fuel elements will be checked frequently and appropriate restrictions placed on the handling of fuel elements as required in re-arranging the core loading.

The nature of the program planned for the oxide critical is such that the accumulated fission products will be extremely small. During much of the work week the reactor will be in a shutdown condition for changes in loading. For most of the measurements the reactor will be in a critical or near critical condition at a power level of the order of a watt. Occasionally an experiment might involve an integrated energy liberation of as much as a kilowatt hour in connection with the determination of flux distributions or lattice cell parameters by foil techniques.

The operating procedures have been outlined in the preceding section. The primary basis for safe operation is the agreement on all operational steps by a minimum of two individuals thoroughly familiar with the details of the program and with adequate background knowledge of and experience with critical experiments. In addition, certain interlocks provided in the control system insure compliance with safe operating procedures.

The safety of the entire program of critical experiments will be enhanced appreciably by knowledge obtained in preceding exponential experiments.

B. Nuclear Characteristics of the Reactor

Initially the ZPR-VII facility will be used to explore the physics characteristics of a large variety of lattices composed of uranium oxide, thorium oxide and heavy water. These, of course, behave differently from one another in situations involving hazards, and there is no unique calculation of behavior possible, as in the case of a specific reactor design. Lattices to be studied will generally be of the type suitable for boiling or pressurized water power reactors using enriched oxides in heavy water. Hence, it is felt that by a judicious choice of a typical lattice, a set of nuclear characteristics can be calculated which should be adequate for hazard estimations. It is felt that the results and conclusions obtained from

the ThO_2 - UO_2 - D_2O lattice below pertain approximately to almost all heavy water-moderated, uranium-enriched thorium oxide or enriched uranium oxide lattices of practical interest.

Figure 13 shows a cross-sectional view of a fuel element in its $4\frac{1}{4} \times 4\frac{1}{4}$ " cell for the typical lattice chosen. The volume ratios are summarized in Table I.

FIG. 13
HALF SCALE TOP VIEW
OF A FUEL ELEMENT

Two-group theory has been used to compute the behavior of a cylindrical core composed of such fuel elements surrounded by a heavy water reflector. However, inaccuracies of this theory, and even of more elaborate ones, may amount to a few percent reactivity in some cases, and the purpose of the program of measurements in ZPR-VII is to reduce these theoretical uncertainties. The parameters used for these calculations are given in Table II.

Several quantities of interest in connection with inherent reactor safety and shutdown mechanisms have been evaluated for the representative reactor assembly described above. The prompt temperature coefficient, due primarily to the Doppler effect and, to a lesser extent, to dimensional and density changes, is $-2.55 \times 10^{-5} \Delta k/k$ per $^{\circ}\text{C}$. The reactivity effect associated with voids formed or introduced in the fuel zone, but computed for all of the heavy water present in the lattice cell, is -0.541% reactivity per $\%$ void. Nine cross rods of 10-inch span, positioned in the EBWR pattern, are worth approximately thirty percent when fully inserted, and the central cross rod alone controls about 5% reactivity. The two-foot thick top reflector of heavy water is worth 1.5%.

Other information is of interest in evaluating the safety of reactor startup procedures. The maximum differential worth of the most effective rod, the central cross rod, is calculated as $0.067\% \Delta k/k$ per centimeter of motion. For the selected lattice cell a maximum rate of change at criticality of reactivity with water level of $0.567\%/cm$ is found for a core with the same buckling (0.001902 cm^{-2}), but with dimensions of 3.07-ft radius and 1.793-ft height of fuel. A minimum rate at criticality of $0.0574\%/cm$ for this lattice would occur in a core of 1.192-ft radius having no top reflector ($B^2 = 0.001902 \text{ cm}^{-2}$ still).

For convenient reference, these pertinent reactivity effects are summarized in Table III.

Table I

TYPICAL REACTOR VOLUME RATIOS

	<u>Relative Volume</u>	<u>Density in gm/cm³</u>
Fuel		
ThO ₂	0.05168	9.69
U ²³⁵ O ₂	0.001467	10.78
U ²³⁸ O ₂	0.000110	10.90
Void (due to packing)	0.01994	0
Fuel Tube		
Aluminum	0.05167	2.706
Void (space around fuel)	0.003814	0
D ₂ O		
Fuel Zone	0.02805	1.109
Moderator Zone	0.84326	1.109

Table II

TWO GROUP CONSTANTS AND DIMENSIONS FOR
A TYPICAL REACTOR

ϵ	1.0178	Radius	1.55 ft
p	0.9080	Height	5 ft
η^{25}	2.073	Radial reflector savings	23.4 cm
f	0.7321	Top reflector savings	36.7 cm
k_{∞}	1.4025	Bottom reflector savings	28.8 cm
τ	141.8 cm ²	B^2	0.001902 cm ⁻²
L^2	40.16 cm ²	Neutron Lifetime	1.692 x 10 ⁻⁴ sec
k_{eff}	1	Delayed neutron fraction	0.755%

Table III

REACTIVITY EFFECTS

Prompt temperature coefficient	$-2.55 \times 10^{-5} \%/^{\circ} \text{C}$
Void coefficient, $d\rho/dV$	$-0.541\%/ \%$ void
Worth of 9 cross rods (EBWR pattern)	30%
Worth of central cross rod alone	5%
Maximum differential worth of central cross rod	0.067%/cm.
Maximum differential worth of water	0.567%/cm.

C. Possible Sources of Excess Reactivity

The nature of critical experiments require deliberate changes in the reactivity of the system. These situations form a part of standard operating practices and subsequently are excluded here. Only unexpected sources of reactivity occurring either suddenly or at a slow rate are under consideration. These may result from equipment failure, operational errors, experiments of an unusual nature, external catastrophes and sabotage.

1. Equipment Failure

Occasional malfunctioning of the mechanical and electrical components of the system can be expected in spite of a systematic program of checking and preventive maintenance. Wherever possible, all components of the system have been designed to be fail-safe. Examples where failure of the components could result in the unwanted addition of reactivity are stuck switches, shorted wires, and inoperative valves. It is conceivable that the switch controlling rod motion could become stuck while the last rod is being withdrawn at a maximum rate and criticality is achieved while the rate of addition of reactivity is a maximum and sustained. A similar situation could result if the flow system does not respond to the switching from high to low flow rate, due to either a defective switch or malfunctioning valve. Furthermore, it is conceivable that a component in a trip circuit fail in such a way that a scram signal from a level or period circuit fails to produce the desired shut-down of the reactor.

2. Operational Errors

Operational errors attributable to faulty judgment or inattention on the part of reactor operators have occurred in the past. Failure to observe the increased reactivity of an assembly while a very effective control rod is being continuously withdrawn, or failure to switch to a low water flow rate, are examples of such operational errors. The resulting rates of addition of reactivity would be identical to those associated with

failure of certain equipment as discussed above. From the nuclear characteristics of the reactor, summarized in Table III, and the mechanical characteristics, tabulated in Appendix B, the maximum rate of reactivity addition due to rod withdrawal and water addition are 0.0366% and 0.231% $\Delta k/k$ per second, respectively. The comparatively high rate associated with full flow is the reason underlying the special procedure outlined in Section IV for approaching a water level critical in steps involving criticality by rod motion.

3. Unusual Experiments

Certain experiments associated with measurements on the effect of voids involve unusual conditions regarding the possible accidental addition of reactivity.

Static void coefficient experiments will involve loading the assembly with special void element tubes filled with heavy water to a predetermined level. The void volume above the water level in these elements will be maintained by top plugs fitted with "O" rings. Leaks in the gasket seals and floating of the elements out of the reactor core are two ways in which excess reactivity may be added to the assembly during this type of experiment. The effect of leaks in the "O" ring seals represents a minimum hazard for several reasons. It is unlikely that such a leak will occur, and especially unlikely that it will occur simultaneously in a large number of plugs. It has been observed that, even when no stopper is used to close off the upper end of a void tube, water will not enter the tube spontaneously when the water level rises over the top of the open void element. Hence it is unlikely that water would enter a void tube through a leak in the gasket. Since the void volume of a single void element is extremely small in comparison with the core volume, only a minute amount of reactivity is associated with each void tube. Any entry of water into a void element will be slow in rate and small in effect. Reactivity would be added more rapidly if the void elements should float out of the reactor. Void elements are held in place by holddown plates covering the top of the element clusters. These plates are attached rigidly to the element clusters by nuts screwed onto the corner posts of the clusters. Thus the buoyancy of the void tubes is counteracted by weight of the clusters and fuel elements.

Dynamic experiments associated with the effect of voids offer another means for the accidental addition of unwanted reactivity. Reasons for giving consideration to dynamic experiments are twofold: there should be an experimental check on theoretical predictions of the effect of void movements in order to obtain confidence in similar calculations in power reactors; and there is reason to believe that this type of experiment could offer advantages of rapidity and/or accuracy over static experiments.

One type of experiment will determine the reactor response to the sudden formation of voids. Two ways of introducing voids rapidly are planned. One involves the introduction of a void-filled container, such as a sheet of Styrofoam. The other involves the release of a fixed volume of air at pressure into an inflatable container, such as a balloon, positioned in the core or into a tube simulating a shroud around a fuel element. Both of these methods result in an approximate step insertion of negative reactivity. The amount of void volume introduced will always be kept less than that associated with a reactivity of $0.7\% \Delta k/k$, such amount having been determined by preceding static experiments. The potential source of unexpected reactivity in this type of experiment is that due to the collapse of the voids and the associated reactivity step, or the breaking off of the void container with an attendant slow addition of the associated reactivity as the material floats out of the reactor. Since these experiments will always be initiated from a reactor supercritical by no more than 0.3% (10-second period) and control rods will never be withdrawn with these voids present, an addition of reactivity by collapse of the voids or fracture of the container would never make the reactor supercritical by more than 0.3% , an amount readily curbed by the safety circuits before an excessive power level would be reached.

A second type of dynamic experiment* will involve fluctuations in void density. Oscillatory introduction of voids will be accomplished either by mechanical motion of a void-containing material over a short distance, or by an oscillation in the volume of air contained in an inflatable device or shrouded region. Such measurements, performed to establish the effect of fluctuations in void density on the stability of operation, will be accomplished with a reference reactivity of no more than 0.3% , and here also control rods will not be withdrawn while voids are present. Associated reactivity swings will always be from a maximum of $+0.30\%$ to $-0.4\% \Delta k/k$. For this type of measurement the period trip circuit will be adjusted to a value much lower than that used normally and may have to be by-passed for rapid oscillations. However, even in the event that the oscillation should stop at the maximum positive reactivity effect, the resulting rise in power level with an asymptotic period of 10 seconds will be readily handled by the power level trip circuits before any damage can occur.

Measurements of the worth of light water as a coolant and as an impurity in the heavy water moderator will also involve the use of the special void tubes. A decrease in the purity of the heavy water is expected to increase the reactivity of a typical system under study up to perhaps a 10% concentration of light water. In order not to contaminate the heavy water, all light water used for these measurements will be introduced in containers of the void element type. Such additions will be made in small increments and the total volume of light water contained in the reactor core with no D_2O present will never be greater than an amount calculated to give

*This type of experiment has been done successfully on JEEP by V. Erickson (private communication).

an assembly having a $k_{\text{eff}} = 0.9$ without rods. During the loading of H_2O -filled void elements the safety and control rods will be inserted for an added margin of safety.

Temperature coefficient experiments involving the heating of the water at atmospheric pressure to slightly less than its boiling point are planned. The water temperature will be increased by immersion heaters in the dump tank. During the heating process the water will be pumped through the by-pass line in the flow system in order to achieve the necessary mixing of the water. When the desired temperature is reached the water will be pumped into the reactor.

Large reactivity effects, possibly as high as 3%, are associated with a reactor temperature change of 80°C . However, the capacity of the two heaters, rated at 36 kilowatts each, will limit the rate of temperature increase to a maximum of 8.7°C per hour. No provision has been made for forced cooling and so the increase in reactivity associated with cooling will take place at even a slower rate.

The mixing of the water during heating and pumping into the tank prevents the addition of a quantity of water at a temperature significantly different from the average temperature of the total water present. Because both the conductivity and bonding of the oxide fuel is poor, the temperature coefficient experiments will of necessity be conducted very slowly. The limited capacity for heating, the time delay in achieving equilibrium, and the dependence upon convective, radiant and conductive losses for cooling will all serve to limit the rate of change in reactivity associated with the temperature coefficient experiments to an extremely low value.

4. External Catastrophes

Possible effects of external catastrophes on the reactivity of the oxide critical experiment should be considered. As was pointed out earlier, the reactor will normally be in operation for only a fraction of the working day. At other times the water will be in the dump tank so only a situation whereby the upper core tank would become flooded, presumably with light water, need be considered for the shutdown reactor. To minimize contamination of the heavy water, the cover plate will be kept on the reactor tank except when some change is being made in the configuration of the assembly. In addition the dump valve will always be open during shutdown. The topography of the area in the immediate vicinity of the critical assembly building precludes the possibility of a flood. Since the building is of fireproof construction and the ZPR-VII structural materials and loading are not inflammable, no large quantity of water would be used in combating a fire in the reactor cell. Thus there is no likely situation which would result in the leakage of a sufficient quantity of water into the assembly to

cause the reactor to achieve a subcritical state. Other conditions under which the shutdown reactor would become supercritical as a consequence of an external catastrophe are not readily imagined.

The addition of reactivity to the system while operating is unlikely to result from any imagined external catastrophe. Any event resulting in an interruption of power would, of course, cause the reactor to shut down. Outbreak of a fire should not be so sudden that the operator would be unable to press the shutdown button. There has never been recorded in this area an earthquake of sufficient intensity to damage a building of the constructional quality of D316. An unexpected external blast sufficient to destroy a wall or ceiling of the reactor cell would probably rupture the 1/4" thick reactor tank with an attendant loss of water (and reactivity). It is highly improbable that such a blast would simultaneously rearrange the fuel in the core into a more reactive array, distort the rods or rod guides sufficiently to hold them out of the assembly, and render the dump valve inoperative yet maintain the tank intact. An explosion associated with one of the reactors operating in an adjacent cell would give a radiation signal sufficient to operate certain of the trip circuits and initiate shutdown before any effects due to blast would occur at the ZPR-VII facility.

5. Sabotage

This zero-power reactor does not differ from the majority of such systems in regard to sabotage. A well-informed person familiar with reactor instrumentation and the nature of the system would have no trouble in devising and carrying out a scheme to damage the facility, destroy a portion of the building, and disperse radioactive materials over the surrounding area.

An act of sabotage of this magnitude would require free, continued and solitary access by an individual to the assembly room and control room, and under certain conditions also to the vault area. This is prevented in numerous ways.

The assembly room, control room, and vault are kept locked during nonworking hours and under constant guard by the Argonne Security Division. Written records of each individual entry into and exit from this portion of the building are kept during working hours and checked at the end of the work day, so that the chance of a person secreting himself in this area is virtually eliminated. Access to the control room is limited to relatively few of Argonne's personnel and these individuals must be admitted by a guard. The security system places emphasis upon recognition of authorized individuals through a photo-badge system and is in reality augmented by personal recognition. During each start-up two qualified operators must be present, and during all operation of the assembly the presence of a minimum of two, a qualified operator and a crew member,

is required. With three critical facilities being operated in the same immediate vicinity, an additional check exists on the possibility that a person determined to commit an act of sabotage would have his activities go undetected.

D. Countermeasures

In the above discussion concerned with possible sources of excess reactivity, it is pointed out that such incidents resulting from external catastrophes are highly improbable and that the rigid security measures in operation at the Laboratory, supplemented by operational rules and the number of personnel in the vicinity, tend to reduce the possibility of sabotage to zero. Consequently, the safety system provided has been designed primarily to counteract any sizable, positive reactivity effects resulting from equipment failure and/or human failure, either associated with operational errors or the conduct of unsafe experiments.

There are several lines of defense in the safety system associated with the oxide critical experiments to be run in the ZPR-VII facility. These represent the combined effects of supervisory control, operational control, and inherent self-control on the part of the reactor.

1. Supervisory and Operational Control

The routine maintenance of the equipment and pre-operational check procedures serve to minimize the likelihood of mechanical failure of the components of the facility. Pre-planning of the experiments and agreement in regard to details by two responsible individuals serve to prevent the introduction of unsafe experiments or faulty operating procedures. The existence of certain interlocks serve to alert the operator during the crucial stages of the startup operation. In addition, the need for changing range on the linear power amplifiers to prevent level trips serves to maintain the operator's attention as the reactor proceeds to criticality and the desired operating level. The detection circuits and their associated meters and records provide a continuous and nearly instantaneous measure of the behavior of the system and provide advance notice to the operator of any need for corrective measures. Should the operator fail to heed the warning given by the detection circuits, the functioning of the trip circuits will prevent any undesirable incident resulting from the operator's inattention.

2. Trip Circuits

During all phases of the reactor operation there will be a minimum of two independent power level trips and one period trip. A total of four independent trip circuits are available. One of these is activated by a log count rate meter which derives its signal from the selected one of two startup counters. A second trip circuit is used in connection with the

log power level channel. A third trip circuit is associated with the very rapid linear level channel. The fourth trip circuit can be activated by signals from either or both of the linear power channels.

The trip circuits function by the opening of relays which interrupt power to the magnetic clutches holding the safety and control rods and water dump valve. Removal of reactivity by the release of the safety and control rods is fast and very effective. Removal of the water upon opening of the dump valve is appreciably slower but even more effective. Even if there should be a complete breakdown of supervisory control and complete disregard of the prescribed operating procedures for maintenance, checking, planning of experiments and netting of safety trip levels, there still would be two very effective safety mechanisms present as inherent features of the systems under study. These are the large, negative temperature and void coefficients of reactivity.

3. Safety Rods

The control and safety rods will be driven by ZPR-I-type drive units. Over the past seven years these units have been used effectively in critical assembly experiments both here and at other installations and are well known for their reliability, ruggedness, and speed of response. For use in the ZPR-VII they have been modified slightly to allow a 5-foot rod drop. In order not to exceed the capacity of the shock absorbers, the rod weight has been held to 24 pounds. The rods will drop in cross-shaped guides having water-filled channels $1\frac{1}{2}$ " wide. Each rod will present an area approximately $1\frac{1}{8}$ " x 20" to move against the drag forces exerted by the water during the rod's relatively free fall. The retarding force exerted by friction in the pulley associated with the cabling and the inertia of the cable takeup drum will be small. The four edges of the cross-shaped guide will be open to allow free flow of water as displaced by the released rod. On the basis of earlier measurements with the ZPR-I drive unit it is estimated that a delay of only 0.02 second will occur between the time that the trip level is reached to the time at which the rod begins to fall, and that it will have fallen successively through 1, 2, 3, 4 and 5 feet in 0.37, 0.52, 0.63, 0.73, and 0.85 second.

In the discussion of the nuclear characteristics of a typical reactor an estimate of 30% shutdown associated with the nine cross rods was given. It was not intended to imply that all nine rods will be used in all experiments. A minimum of 2% reactivity held by a minimum of two safety rods will always be present as well as an additional amount of reactivity associated with a minimum of one control rod.

4. Water Dump

The dump valve is held closed against the weight of the water in the reactor core tank by an electromagnetic clutch in a motor-driven gear reduction unit. When the clutch releases, the vane swings

down under the action of a weighted lever and the force of the water above it. For a tank containing no core structure and filled to the height of the permanent overflow-vent line, the water level will fall as follows:

Height from bottom: 8 7 6 5 4 3 2 1 0 feet.

Time to reach height: 0 3.4 6.1 9.1 12.2 15.5 19.1 23.2 27.8 seconds.

These values will not differ appreciably for the case of a core present in the tank, since the reduction in the volume of water due to the presence of the core will be offset by the impedance of the core to water flow. The dropping of the water by 2 feet corresponds to the removal of the top reflector from the core. This is calculated to contribute a negative reactivity effect of 1.5% for the fuel lattice arrangement chosen as representative.

5. Prompt Temperature Coefficient

The prompt temperature coefficient associated with the oxide critical is due primarily to the Doppler effect. In order to establish the magnitude of this inherent safety mechanism, a calculation was made to establish the amount of excess of reactivity insertion which could be compensated for by this prompt temperature coefficient without damaging the reactor in any way. To obtain a conservative estimate of this reactivity, the following problem was performed on Argonne's AVIDAC. A positive reactivity step was suddenly inserted at low power, and the only inherent shutdown mechanism was taken to be the prompt temperature coefficient, $-2.55 \times 10^{-5} \text{ } ^\circ\text{C}^{-1}$ (appropriately statistically averaged). Radiation losses were negligible, although the small conduction losses through an assumed 0.005" air film were taken into account.

Figure 14 shows the results for a 1% reactivity step and an average fuel pellet. Peak power occurs when the excess reactivity is reduced to prompt criticality, as one expects. The excess reactivity is not zero until some time later. Until conduction losses are noticeable, the temperature distribution within the pellet is uniform. Figure 14 shows that the average pellet temperature will remain below 660°C , the melting point of aluminum. If a typical maximum to average power density is 2, then by doubling the average actual temperature curve one finds that at 1.04 seconds the hottest pellet has reached 660°C . However, a reaction between aluminum and water cannot be induced for several reasons:

- a. Beyond 1.04 seconds the pellet temperature rise e-folding time never exceeds 0.5 second, and the average temperature of the hottest pellet never exceeds 960°C ;

- b. The outer surface of the aluminum jacket is in contact with cold water;
- c. The film temperature drop between the pellet and aluminum is appreciable.

The total energy of the power excursion calculated is 71.3 kilowatt seconds per average fuel tube, 77% of which occurs after peak power.

Figure 15 shows results for a 2% reactivity step. Here the average pellet reaches 1680°C, and the hottest pellet would reach the melting point and start to melt. Hence, barring a reaction between aluminum and water, a 2% reactivity step can also be safely terminated by the prompt temperature coefficient of the reactor. Even for this case the reaction between aluminum and water is not too probable because of the large film temperature drop at the inner surface and the contact with cold water at the outer surface of the aluminum. The total energy of this power excursion is 172 kilowatt seconds per average fuel tube.

Lattices deviating from this typical one will, of course, have a larger or smaller degree of inherent safety. In particular, lattices having a larger resonance escape will have a Doppler effect smaller by the ratio of $1-p$ to $1-0.908 = 0.092$. These, however, will have smaller amounts of fuel.

6. Void Coefficient of Reactivity

A second inherent safety feature associated with water reactors is the void coefficient of reactivity. For the typical lattice studied, the void coefficient is not only negative, but also more than twice as large as any of the series of reactors studied in the BORAX and SPERT safety programs. Consequently, boiling is a very effective shutdown mechanism. However, because of the poor bond between the oxide fuel element and the aluminum jacket it is not possible to estimate reliably how much reactivity in addition to that handled safely by the Doppler effect could be tolerated as a consequence of the additional safety due to the boiling shutdown mechanism.

It is possible that some lattices could have void coefficients less negative than that of the typical lattice chosen for calculation, and the extent to which that quantity aids shutdown is lessened. In the event a lattice with a positive void coefficient is constructed, several facts should be considered. First, there is no sharp discontinuity in the inherent safety of a reactor as the void coefficient changes sign. Secondly, as voids form, the void coefficient becomes more negative, regardless of the sign of the initial value. Finally, in the event that it is possible, in spite of the high leakage, to construct a lattice with an objectionably large positive void coefficient, this would not logically fall into the proposed program which

FIG. 14
TERMINATION OF A 1% EXCESS REACTIVITY INSERTION
BY PROMPT TEMPERATURE COEFFICIENT

FIG. 15
TERMINATION OF A 2% EXCESS REACTIVITY INSERTION BY PROMPT TEMPERATURE COEFFICIENT

envisions lattices suitable for practical power reactors. If, for some reason not presently foreseen, it becomes desirable to study such a lattice, the proposed experiment will be submitted for review and approval by the Laboratory's Reactor Safety Review Committee prior to its execution.

The actual lattices to be investigated are expected to be have more like the one chosen for analysis rather than like the possible exceptions noted above. Consequently, it can be concluded that certainly 1%, and very probably 2%, reactivity insertion can be handled by the reactor's inherent shutdown mechanisms of negative temperature and void coefficients without fission product dispersal.

E. A Nuclear Accident

On the basis of the above discussion concerned with possible sources of excess reactivity, the procedures followed to minimize the likelihood of their occurrence, and the countermeasures available to offset their effect in the event of their occurrence, it is difficult to imagine a set of circumstances other than sabotage which would result in an excursion damaging to the reactor. The several lines of defense presented by the safety system against a power excursion make a runaway of any magnitude a very improbable event.

1. Possible Cause of a Significant Excursion

In the above discussion it was apparent that the maximum rate of addition of reactivity was associated with the flow of water into the reactor tank at the maximum rate of 160 gallons per minute. Other possible rates of sustained addition of reactivity are appreciably smaller. Consequently, a sequence of events involving malfunctioning of equipment and poor judgment on the part of the operator during the addition of water at the maximum rate of water flow is the most likely manner in which a fast period could result. It is believed that the most unfavorable set of circumstances, exclusive of compounding highly improbable situations, is as follows:

With the safety rods out and only one control rod present in the reactor, the reactor is being flooded with water in preparation for going critical on motion of the one control rod. As a result of a poor estimate of the approximate loading required (an unlikely situation because of the existence of the results of exponential experiments as well as calculations), the reactor actually will go critical on water level. As the water level is approaching that corresponding to criticality, the operator notices that the instrument readings are becoming appreciable and proceeds to remove the source. While the operator fixes his attention on the source removal operation, the water, flowing at maximum rate, reaches the critical level and continues past, adding reactivity at the rate of $0.231\% \Delta k/k$ per second. It is conceivable that the source be removed just at that time, in relation to

the $k_{\text{effective}}$ of the assembly and its rate of change, which will cause the most pronounced decrease in the flux level at the detectors. This decrease, though only of momentary duration, will delay the period channel's detection of the true period. Assume that only the minimum number of trip circuits are in operation, and these are one low power level, one high power level and one period circuit. It is not impossible that a component in one of these, for example the low level circuit, should have failed since its morning check and therefore be inoperative. Then a scram signal due to excessive power level would be delayed. Whether the high power level or period circuit would ultimately furnish the power interruption to the magnetic clutches is immaterial. This imagined set of circumstances could delay shutdown until perhaps as much as $3/4\%$ of excess reactivity had been added by movement of the water level. The scram signal from either the level or period trip circuit will cause the safety rods to drop in and the water dump valve to open, and simultaneously will divert the water flow from the inlet to the by-pass line. Either the safety rods or the water dump is adequate for controlling the amount of excess reactivity added, so only if malfunctioning of both of these devices should occur simultaneously would a hazardous situation ensue.

In the situation imagined above a pessimistic estimate of a gradual accumulation of as much as $3/4\%$ excess reactivity was made. As was pointed out earlier, a 1% step of reactivity can be readily handled by the prompt temperature coefficient alone without any damage, and possibly this coefficient could also control a 2% step. Void formation and the attendant large negative void coefficient of reactivity provides an additional margin of safety.

2. Energy and Radioactivity in an Excursion

In spite of an inability to imagine any set of circumstances which would result in a runaway reactor, it is instructive to consider the magnitude of the energy release associated with inputs of excess reactivity which could be achieved deliberately and to speculate on conditions which might result in a chemical reaction among the constituents of the reactor.

The energies and amounts of radioactivity associated with certain events involving the operation and safety of the oxide critical are summarized in Table IV. The results are expressed in terms of a single average fuel rod.

The accumulation of fission products associated with the operation of the assembly at a milliwatt per average fuel element for 40 hours a week over a period of one year is seen to be an order of magnitude less than the number of fissions resulting during an excursion associated with a 1% step in reactivity. This amount of operation certainly will represent an upper limit and so the accumulation of fission products in the

Table IV

REACTION ENERGIES AND ACTIVITIES PER AVERAGE FUEL ROD

Event	Energy	Number of Fissions	Curies of Activity	
			β	γ
Operation at 10^{-3} watt per element for 1 year (40 hours per week).	7.5 kw sec	2.25×10^{14}		
1% $\Delta k/k$ step	71.3 kw sec	2.14×10^{15}	$2.32 \times 10^5 t^{-1.21}$	$1.30 \times 10^5 t^{-1.21}$
2% $\Delta k/k$ step	172 kw sec	5.15×10^{15}	$5.58 \times 10^5 t^{-1.21}$	$3.13 \times 10^5 t^{-1.21}$
Chemical reaction of 25% Al	915 kw sec	-		

experimental program may be assumed negligible in connection with considerations of potential hazards. The results of the AVIDAC calculation for a 2% reactivity step are also included in Table IV. Increasing the size of the reactivity in excess of prompt critical by a factor of 5 has resulted in an increase in the number of fissions by a factor of 2.4. For higher reactivity additions, however, the number of fissions should be almost directly proportional to the reactivity in excess of prompt critical.

For comparison with the energy liberation associated with reactor transients of sizable magnitude, there is included in Table IV the amount of energy released in the chemical reaction of 25% of the aluminum associated with a fuel rod and the subsequent reaction of the deuterium released with oxygen in the air within the reactor cell. These reactions are

and

The chemical energy potentially available is seen to be well beyond the magnitude of that associated with any conceivable reactivity effect in the reactor. The chemical reaction requires molten aluminum in a dispersed state for its initiation, and no sure mechanism can be presented for achieving this chemical reaction condition in the oxide critical assembly. Nevertheless, if this state were attained it doubtlessly would be accompanied by appreciable amounts of previous boiling, sufficient to have terminated a nuclear excursion induced by several percent excess reactivity.

The occurrence of a chemical reaction between the fuel pellets and the rod jacket is improbable. Reported* values for the heats of formation of ThO_2 , UO_2 and Al_2O_3 exclude the possibility that the ThO_2 contained in the fuel pellets will react with the aluminum jacket and indicate that a reaction between UO_2 and Al is possible only at temperatures below the melting point of aluminum. At such temperatures the contact between the UO_2 and aluminum will be very poor and the rate of reaction will be vanishingly small. UO_2 will react with the oxygen in the air which fills the gap between the fuel pellets and the jacket, and the rate of reaction increases with increasing temperature. U_3O_8 will not react with aluminum, but UO_3 will. However, the latter reaction is considered unlikely because the very small amount of oxygen present will favor the formation of U_3O_8 rather than UO_3 .

To convert the energies and activities listed in Table IV to a form useful for estimating the associated hazard requires a knowledge of the number of fuel elements present in the reactor. The maximum number available will be approximately 6000 (having two or possibly three different fractions of uranium) although the average number present in the assembly will be appreciably less than this maximum. The fuel cluster used in the reactor taken as representative in an earlier portion of this section (see Figure 13) will contain 31 fuel rods at each lattice point and the lattice points are in a square array with a $\frac{1}{4}$ -in. spacing. The maximum rate of change of reactivity with water level was associated with a core having a 3.07-ft radius, becoming critical at a height of 1.793 ft. In such a core the equivalent number of full-length fuel rods would be 2650. This figure is also approximately the average number of fuel rods which will be present in the reactor core during the experimental program.

3. Results of an Imagined Transient

a. Mechanism for Dispersal of Fission Products

The AVIDAC calculations of the power excursion associated with a 2% reactivity step showed that the prompt temperature coefficient of reactivity was sufficient to terminate such an excursion. Radiation losses were neglected and only the small conduction losses through an assumed 0.005-inch air film were taken into account. Using a maximum to average power density of 2, it was found that the hottest pellet in the runaway reactor would reach the melting point of thorium oxide, $3,220^\circ\text{C}$.

Under actual conditions, some heat transfer to the water and aluminum will occur by radiation. Also, the pellets will make at least point contact with the wall of the aluminum jacket. Thus the oxide fuel pellets will lose heat during an excursion by both radiation and conduction. This will decrease the rate of reactivity loss due to the Doppler temperature coefficient of reactivity. However, the flow of heat into the water will bring the void coefficient shutdown mechanism into action, although the time delay may result in a higher energy liberation in the fuel elements.

*Bureau of Mines Bulletin 542: Heats and Free Energies of Formation of Inorganic Oxides, by James P. Coughlin, U. S. Government Printing Office, 1954.

If all of the nuclear energy liberated in a 2% excursion were to result in heating the aluminum and the reactor water to 100°C and vaporizing a portion of the water, the resultant pressure in the reactor cell would be only 4.6 psi (gauge). Hence, rupture of the building as a consequence of a 2% reactivity step is considered unlikely.

Because of the very high melting and boiling points of the oxide fuel elements, a chemical explosion would be necessary to disperse the radioactive material resulting from a nuclear excursion associated with even several percent reactivity. Only in the event that there is melting, and hence rupture of, the aluminum casing containing the oxide pellets would the fission products escape from the fuel rods. Only if a chemical reaction involving an appreciable quantity of the aluminum should occur would there be a way of dispersing the material beyond the confines of the reactor cell.

It is likely that the good contact between the aluminum jacket and the heavy water moderator will preclude a high temperature in the aluminum for a 2% reactivity step. However, for reactivity steps of greater magnitude, the energy liberation will be sufficient to melt the oxide fuel pellets and thereby result in good contact between the fuel and the aluminum jacket. Under these conditions the possibility of aluminum reaching its melting point of 660°C cannot be excluded. Not only must the nuclear energy released in the excursion melt the aluminum but the aluminum must, in addition, be dispersed into fine particles in order for the chemical reaction to proceed.

If the nuclear energy released in a large reactivity step should raise the aluminum to its melting point and during the process of steam formation in the water a dispersal of fine particles of aluminum should occur, the chemical energy release associated with the chemical reaction of the aluminum in the fuel rods and subsequent reaction of the heavy hydrogen would occur at a rapid rate. Assuming that 25% of the aluminum is oxidized and that the energy released in this chemical reaction is all absorbed in vaporizing the water, an excess pressure of approximately 52 psig would develop rapidly and the blast would rupture some part of the reactor cell. In this way the accumulated fission products would have a chance to escape from the reactor vessel and the cell and thereby constitute a potential hazard to personnel in the immediate vicinity.

b. Contamination of the Surrounding Area

For an excursion resulting from a 2% reactivity step in a reactor containing 2650 fuel rods, the number of fissions produced is 1.36×10^{19} . It is estimated that 4.65×10^{19} fissions would occur with a 5% reactivity step and 1.0×10^{20} for a 10% step. These steps far exceed any imagined situation which might result in the oxide critical experiments, excluding sabotage.

Since a chemical reaction would probably be necessary to achieve a dispersal of fission products beyond the confines of the reactor cell, the external hazard associated with an incident involving 10^{20} fissions has been evaluated. These calculations are included as Appendix E. Assuming that 50% of the products formed in 10^{20} fissions escape from the reactor cell in the form of a plume or cloud, the significant results from the hazards viewpoint are as follows:

- (1) The inhalation hazard due to a cloud formed in a violent explosion and expulsion of gaseous and particulate debris is negligible.
- (2) For slow leakage through minor fissures in the reactor cell walls an observer one mile distant who remained in the path of the plume would receive less than one third of the "once in a life-time" dose from I^{131} , and would accumulate much less than the maximum permissible dose of Sr^{89} , Ba^{140} or Rh^{105} .
- (3) The ingestion hazard from contaminated water supplies is negligible.
- (4) The external radiation dose from airborne activity at 1 mile is only 228 rep under the most pessimistic meteorological assumptions.
- (5) For total instantaneous washout at two miles, the infinite γ -dose is 161 rep to an exposed observer, 80 rep of which occurs in the first three hours. The β -dose is 1610 rep. The radius within which this activity is confined to achieve this level is only 289 meters, however. For optimum scavenging rates, the dose is down by a factor of more than twenty.

It should be emphasized that the above results apply to a disaster involving 10^{20} fissions, while only 1.36×10^{19} fissions are associated with a 2% reactivity step, and that a 2% reactivity step far exceeds that excess reactivity associated with the most pessimistic situation imagined for the oxide critical.

In a disaster involving 10^{20} fissions it is not unrealistic to assume that the reactor cell would be destroyed. In this case a cloud would be formed and the hazard associated with a cloud is considerably less than that for a plume. For a disaster in which only leaks in the cell resulted there would probably be considerably less than 10^{20} fissions and

correspondingly less fission product activity would be present in the plume extending downwind. Furthermore, time would be available for minimizing the effects by sealing of the cell and evacuation of the area in the downwind direction.

The Laboratory has a well-defined system of procedures in the event of a reactor incident. These are put into effect by dialing an emergency phone number. A report entitled "The ANL Emergency Plan" has been drawn up and copies distributed among administrative and supervisory personnel here and to the Atomic Energy Commission. This report includes evacuation plans, procedures for contacting both civil and military authorities, and security measures. This report has no document number but is a part of the Administrative Guide.

Clearly the public hazard associated with nuclear excursions of the magnitude considered here is extremely small, since the Laboratory site extends a minimum of one mile in all directions from the location of the oxide critical, and the nearest town, Lemont, is over two miles away.

F. Conclusions from Safety Evaluation

This review of the design, program and operating procedures for the oxide critical experiments has served to reveal the potential hazards associated with the operation of this facility. The risk to both the staff and public has been shown to be minimal. While this evaluation has resulted in increased confidence in the soundness of the plans, it is clearly recognized that there is a history of reactor incidents involving systems which were subjected to the same close scrutiny and in which the operating crew were equally confident. Consequently, a positive effort to keep the personnel always "safety-minded" will be sustained throughout the life of the ZPR-VII facility.

APPENDIX A
MAKE-UP OF SURROUNDING AREA

The map of the territory surrounding the Reactor Site (see Fig. 24) is useful for orienting the DuPage site insofar as this region, in general, is concerned. It is evident that a higher density of population occurs, because of communities, in the directions starting northwest and moving clockwise to the southeast.

Table IVa below shows population distributions, using the DuPage site as a focal point, for areas up to 20 miles in diameter. Also, Fig. 24 shows various industrial installations worthy of note and the Hinsdale Sanitarium.

Table IVa

POPULATION DISTRIBUTION
 (DuPage Site as Center)

Item No.	Region	Estimated** Population (1950 Census)	Area (Sq. Miles)	Density (Av. Population per Sq. Mile)
1	Total within 1-1/2-mile radius*	3,500	6.9	507
2	In the band between 1-1/2- and 5-mile radius	13,000	71.6	182
3	In the band between 5- and 10-mile radius	84,000	235.7	357
4	Total within the 5-mile radius	16,000	78.5	204
5	Total within the 10-mile radius	100,000	314.2	319

*During normal working hours.

**The population in this area is increasing at about 10% per year.

APPENDIX B

SUMMARY OF THE MECHANICAL FEATURES OF ZPR-VIIBUILDING CONSTRUCTION DETAILS

Building construction material:	Concrete and brick
Shield wall material:	3-ft thick reinforced poured concrete and concrete block
Floor load limit:	3000 lb/sq ft
Crane capacity:	4000 lb
Auxiliary cable hoist:	500 lb
Shielded viewing windows:	2
Doorways:	3: 1 shielded, 1 emergency and 1 large service
Cell size:	30 ft long x 26 ft wide x 25 ft high
Ventilation:	2000 cfm

REACTOR COMPONENTSReactor Tank

Size of lower tank:	81 in. ID x 1/4 in. wall x 8 ft - 8 in. high
Size of upper tank:	81 in. ID x 1/4 in. wall x 47 in. high
Upper ring:	94 in. ID x 1/2 in. wall x 7 in. high
Lower tank capacity, total:	2100 gal
Tank capacity with 148 fuel cluster core loading:	1730 gal
Minimum time to fill:	11 min
Maximum time to empty:	28 sec
Maximum water velocity through tank and core:	0.16 in./sec
Rate of circulation:	0 to 10 gal/min (adjustable by needle valve)

Core

Maximum possible loading:	255 clusters
Maximum number of elements per cluster:	101
U ²³⁵ content per fuel element:	12.2 ± 0.2 gm*
Weight ratio of thorium to U ²³⁵ :	15 ± 0.3%*
Pellet size (in.):	0.230 minimum x 0.236 maximum x 0.600 long (approx.)

*Quantities applicable to first 30% of the fuel rods. Fraction of U²³⁵ will be reduced in the remaining portion.

Core (Cont'd.)

Stacked pellet length:	60 in. + $\frac{3}{16}$ - 0
Fuel tube size (in.):	0.312 - 0.306 OD x 0.246 - 0.241 ID
Cluster Dimensions:	$3\frac{7}{8}$ in. sq. x $63\frac{1}{2}$ in. long

Control Rod

Size:	10 in. across blades x 60 in. long
Weight:	24 lb
Neutron absorber:	Cadmium
Number available:	9
Maximum speed of withdrawal:	12.9 in./min
Scram time:	1 sec
Drive unit:	ZPR-I type

Control Rod Guide

Length:	72 in.
Number available:	9

Dump Tank

Size:	5 ft - 7 in. wide x 7 ft - $3\frac{1}{2}$ in. long x 8 ft - 4 in. high
Storage capacity:	2500 gal

CIRCULATING SYSTEM

<u>Feed Line Size:</u>	2 in. NPS
<u>Dump Line Size:</u>	12 in. NPS
<u>Maximum Flow:</u>	160 gpm
<u>Adjustable Flow:</u>	0 - 10 gpm
<u>Circulating Pump</u>	

Power input:	3 phase, 220 - 440 volts
Horsepower:	3
Capacity:	160 gal at 40 ft TDH
Speed of rotation:	1750 rpm
Suction:	2 in. NPT
Discharge:	$1\frac{1}{2}$ NPS, 150-lb flange
Standard pressure limit (suction):	150 psi
Materials of construction:	304 Stainless Steel

Adjustable Overflow Standpipe

Limits of adjustment:	23 - 96 in.
Speed of adjustment:	19 in./min
Size of overflow line:	2½ in. NPS

Air Vent Line

Size of line:	6 in., Sch. 10 pipe
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Dump Valve

Size:	12 in. NPS (flanged)
Torque to close:	630 in. - lb
Drive unit:	1/4 - hp gear reducer
Speed for closing:	12 rpm
Angle of rotation:	78 degrees

APPENDIX C

HYDROLOGY AND SEISMOLOGYA. Soil and Hydrology1. Surface Characteristics and Drainage

Topographically, the DuPage site of Argonne National Laboratory is a rolling and partly rugged, partly forested plateau, rising quite sheerly from the Des Plaines River. The site is located on soil of a composition differing from the locality in general. Approximately 80% of the site is covered with yellow-gray silt loam (upland timber soil) and most of the remaining surface is exposed limestone, glacial river bottom, and river silt soils. The moraine overlies a Niagaran limestone formation to a depth varying from zero to 160 ft. Less than 5% of the soil is of the typical DuPage County composition, which is brown and black silt loams. There is very little precipitation penetration because, due to the type of soil, almost all rainfall is surface drained.

Bounding the DuPage site (elevation plus 750 ft) on the south is the Des Plaines River (elevation plus 590 ft), which flows southwesterly. The Des Plaines River, the Sanitary and Ship Canal, the Illinois and Michigan Canal, and the Sag Channel have all been interconnected by the time they reach Joliet, which lies about 14 miles southwest of the DuPage site (Fig. 23).

The water flow in the Des Plaines River just prior to Joliet varies between 10 and 100 cfs; the excess is diverted to the Sanitary Canal, at the point just south of Lemont, before reaching Joliet. The flow rate of the Sanitary Canal for 1944 and 1945 averaged about 3400 cfs; of this about 1600 cfs is water which has been diverted from Lake Michigan, and the balance is made up of domestic and industrial discharge from this district. Both the river and the canal flow approximately 20 miles per day. Any surface drainage, therefore, from the DuPage site would be diluted first by the river, bordering the site, and then further diluted by the canal within 14 miles, normally within 17 hours, unless such drainage occurred during a wet period when it would then be diluted more extensively and sooner.

Canal water is used prior to and in Joliet for some industrial uses, such as hydro-electric generators and condensers, as well as for irrigation at the state prison. After leaving Joliet, the Des Plaines River carries these waters southwest to a point approximately 14 miles further where the river is joined by the Kankakee River forming the Illinois River. The first use made of this water for drinking purposes is at Alton, Illinois, which is about 275 miles from the DuPage site and 10 miles downstream from the point where the Illinois River merges with the Mississippi River. Both Alton and East St. Louis treat about five million gallons per day of this Mississippi River water.

FIG. 17

FLOW OF WASTES
FROM CHICAGOLAND (INCLUDING ARGONNE)

2. Sub-surface Water Conditions

The geology of the DuPage site including the sub-surface water conditions is described in a report entitled "Geology and Ground-Water Hydrology of the Argonne National Laboratory Area, DuPage County, Illinois" by Allen, Drescher and Foley, Consultants to the U. S. Atomic Energy Commission. Presumably, copies of this report are available for the AEC. The following statement concerning any possible hazard due to water-borne activity has been prepared by Dr. F. C. Foley, who is the senior author of the above-mentioned report.

Ground water in the Laboratory area is obtained from two principal aquifers, the Niagara dolomite and deeply buried Ordovician and Cambrian sandstones. The two shallow supply wells in the Laboratory area obtain water from the Niagara dolomite while the deep well is cased and sealed through the Niagara and gets its water from the deep sandstones. Only the Niagara dolomite needs to be considered in connection with possible contamination from an accidental spill of radioactive liquid, as the deep aquifers are separated from the surface by hundreds of feet of rock formations of which the Maquoketa shale is particularly impermeable.

The uppermost formation at the Laboratory site is glacial till of Late Wisconsin age, which overlies the Niagara dolomite. It covers the whole area except along the bluff on the valley of the Des Plaines River, Chicago Sanitary and Ship Canal and along Mill Creek below Bluff Road. The till has a maximum thickness of about 180 ft. It consists of clay and silty clay containing fragments of dolomite and other consolidated rock, and a few thin lenses of sand.

The glacial till is relatively impermeable and water moves through it very slowly. No experimental work has been done at the Laboratory site to determine the exact rate of movement, but such work is planned. At the site of the Argonne Research Reactor, the glacial till is about 110 ft thick. It is estimated that about one year would be required for water introduced near the surface to penetrate to the top of the Niagara dolomite. All the time, and even if it did, it would be diluted with normal ground water descending with it. Once it entered the Niagara it would move through open fissures down the hydraulic gradient, which is generally southeastward and not toward the present supply wells. The closest well in use is the meteorology station well. It might seep gradually into the Des Plaines River over an extended period of time. In view of the great thickness of relatively impermeable glacial till at the reactor site, and the direction of the hydraulic gradient in the Niagara dolomite, it is very doubtful that radioactive liquids would enter the Laboratory supply well or private wells. It is probable that such liquids would be extremely diluted by natural water.

B. Seismology

Realistic prediction regarding earthquakes or earthshocks, their frequency and severity, can only be based upon the seismic history of the area concerned. Apparently, new trouble areas seldom occur and this is particularly true of the United States. Insofar as the Argonne sites are concerned, it is expected that no building of sound construction on firm ground will suffer any damage as a result of the infrequent and small earthquakes which may occur anywhere within 150 miles from the sites. It is conceded that earthquakes occurring as far as 500 miles away may disturb sensitive laboratory apparatus.

APPENDIX D

CLIMATOLOGY

The climatology of the DuPage site of the Argonne National Laboratory and of the adjacent region is well documented, and so this material will not be reproduced here. The most recent report by Argonne's Meteorology Group is ANL-5592, "Five Year Climatological Summary, July 1949 to June, 1954" by Harry Moses and Jack Willett.

A summary of the climatology pertinent to reactor hazards evaluation, based on these results reported by Argonne's meteorology group, has been submitted recently to the Reactor Hazards Evaluation Staff, Division of Civilian Application, USAEC as Appendix H, pp. 76 to 106, of ANL-5647, "Summary Report on the Hazards of the Argonaut Reactor" by D. H. Lennox and C. N. Kelber.

APPENDIX E

RADIATION HAZARD FROM A REACTOR DISASTERThe Disaster

In the following hazard estimates, it is arbitrarily assumed that a reactor incident occurs which results in the fission of 10^{20} U^{235} atoms in a time interval of the order of a millisecond. The nuclear energy liberated instantly will be about 185 Mev per fission or about 7.1×10^8 calories for the 10^{20} fissions. No mechanism will be postulated for the release of the fission products, but the consequences of two situations will be evaluated. One situation, arising perhaps as a consequence of the nuclear energy release initiating a chemical reaction among the constituents of the assembly, imagines a violent explosion which destroys the reactor cell and results in gaseous and small particulate debris in the form of a hot cloud escaping immediately. The other case assumes a less violent explosion which results in minor fissures in the walls and doors of the cell and so the radioactive material leaks from the cell slowly, forming a plume in the downwind direction. Incidents in which the material is confined to the cell are not considered. Furthermore, it is assumed that the reactor involved has a negligible accumulation of fission products from its prior operation.

Amount of Activity:

Because the reactor considered here is a zero-power reactor, all of the fission product activity which could give rise to radiation hazards will be generated by the 10^{20} fissions. In the following estimates, it will be assumed that 50% of each fission product isotope escapes from the reactor building and travels downwind in a cloud or plume.

The total beta activity due to 10^{20} fissions will be taken to be $1.6 \times 10^{20} \times t^{-1.21}$ Mev/sec where t is in seconds (Meteorology and Atomic Energy, p. 157). The preceding expression also gives the total gamma activity due to 10^{20} fissions. External beta and gamma radiation hazards will be due to 50% of this activity or $8 \times 10^{19} \times t^{-1.21}$ Mev/sec (beta or gamma).

Estimates of the ingestion and inhalation hazards are made only for those radioisotopes for which data is given in ADC-65 and also in NBS-52 and for which the critical organ is the thyroid, kidneys, or bone. ADC-65 gives the activity in disintegrations per minute for each fission product isotope at various times after the simultaneous slow neutron fission of 10^4 U^{235} nuclei. Because the various fission product activities build up and decay at different rates, it complicates hazard estimates considerably to consider decay. Therefore, the amount of each radioisotope present in the cloud or plume is taken to be 50% of the maximum amount of that radioisotope present in the fission products at any time. In the case of Te^{127} ,

Te¹²⁹, and Y⁹¹, the amount of isotope present in the plume or cloud is taken to be 50% of the maximum amount of the long-lived member of the isomeric pair present in the fission products. Table V lists the radioisotopes considered in this report; it gives the maximum activity of each isotope in disintegrations per minute per 10²⁰ fissions (taken from ADC-65), and it gives the amount of each isotope in the cloud or plume which escapes from the reactor building in microcuries.

Biological Effect of Activity:

Inhalation and ingestion hazard estimates are based upon the data and calculations in National Bureau of Standards Handbook 52 (NBS-52). Columns 4 through 8 of Table V give some of the values from NBS-52 which are used in the following hazard estimates. As explained in NBS-52, the maximum permissible amount of radioisotope in the total body is calculated by determining the uniform concentration of the radioisotope in the critical tissue that will irradiate it at a rate of (0.3/RBE)rep/week continuously for a lifetime. As far as present knowledge goes, a dosage rate of (0.3/RBE)rep/week over the whole body for a period of years is not supposed to cause any detectable biological injury. The maximum permissible concentration of the radioisotope in water is calculated on the basis of continuous ingestion of the water for a lifetime.

Estimated radiation doses may be compared to several numbers. One such number is a total exposure of 300 roentgens over the whole body. This is a quite serious exposure and will produce a mortality of about 15%. This exposure is the limiting dose used by the Reactor Safeguard Committee in defining the exclusion radius for a power reactor (cf. Wash III). Another number is a total exposure of 25 roentgens (measured in air) over the whole body. This is the "once in a lifetime" dose defined in National Bureau of Standards Handbook 59; this dose is supposed to produce no detectable clinical effects. In NBS-59, it is assumed that an exposure of 25 roentgens will have no effect upon the radiation tolerance status of the observer.

The Hazard Estimates:

Argonne National Laboratory controls the area within one mile of the reactor building. Therefore, the primary concern of this appendix is the radiation dose which an observer one mile or more from the disaster might receive.

The radiation hazards due to a radioactive cloud or plume which escape from the reactor building after a reactor disaster will be considered as follows:

A. Inhalation Hazard. If an observer stands in the plume until the building stops leaking, or under the cloud until it has passed overhead, he

Table V
HAZARDOUS FISSION PRODUCTS FOR INHALATION AND INGESTION

Radioisotope	Maximum Activity of Isotope Produced by 10 ²⁰ Simultaneous Slow Neutron Fissions of U ²³⁵ (AFC-65) in (disintegrations/minute)	Amount of Isotope which Escapes from Reactor Building = Q (in microcuries)	Effective Half-life (in Days)	Maximum Permissible Amount of Radioisotope in Total Body for Continuous Exposure (in microcuries)	Maximum Permissible Concentration of Radioisotope in Water for Continuous Exposure (in microcuries/milliliter of water)	Fraction in Critical Organ of that in Total Body	Fraction Reaching Critical Organ by Inhalation
Critical Organ: Kidneys							
Ru ¹⁰⁶ + Rh ¹⁰⁶	6.6×10^{11}	1.49×10^5	19	4	0.1	0.04	0.01
Rh ¹⁰⁵	2.3×10^{14}	5.18×10^7	1.4	9	1.5×10^{-2}	0.08	1.7×10^{-2}
Te ¹²⁷	1.5×10^{11}	3.38×10^4	13	4	3×10^{-2}	0.2	0.02
Te ¹²⁹	2.7×10^{12}	6.08×10^5	10	1.3	10^{-2}	0.2	0.02
Critical Organ: Bone							
Ga ⁷²	2.2×10^9	4.95×10^2	0.59	8	9	0.82	0.1
Sr ⁸⁹	4.2×10^{13}	9.46×10^6	52	2	7×10^{-5}	0.7	0.22
Sr ⁹⁰ + Y ⁹⁰	2.7×10^{11}	6.08×10^4	2.7×10^3	1	8×10^{-7}	0.7	0.22
Y ⁹¹	4.4×10^{13}	9.91×10^6	51	15	0.2	0.65	0.15
Mo ⁹⁹	1.0×10^{15}	2.25×10^8	2.8	50	14	0.5	1.8×10^{-4}
Ba ¹⁴⁰ + La ¹⁴⁰	2.3×10^{14}	5.18×10^7	12	5	2×10^{-3}	0.96	0.2
Ce ¹⁴⁴ + Pr ¹⁴⁴	8.9×10^{12}	2.00×10^6	180	5	4×10^{-2}	0.8	0.10
Pr ¹⁴³	1.4×10^{14}	3.15×10^7	11	29	0.4	0.6	0.063
Pm ¹⁴⁷	8.8×10^{11}	1.98×10^5	140	120	1	0.7	0.09
Sm ¹⁵¹	3.3×10^{10}	7.43×10^3	3.9×10^4	420	0.2	0.65	0.05
Critical Organ: Thyroid							
I ¹³¹	1.7×10^{14}	3.83×10^7	7.7	0.3	3×10^{-5}	0.2	0.15

will inhale a certain amount of each fission product isotope. Estimates are made of the maximum amount of each radioisotope listed in Table V which could be accumulated in the observer's body and these amounts accumulated are compared with the maximum permissible amounts as given by NBS-52.

B. Ingestion Hazard. Ingestion hazards arise when the activity in the cloud or plume is deposited on the ground from whence it can get into the water supply or into edible plants and animals. Estimates of the maximum ingestion hazard to downstream communities are made by assuming that all of the activity in the cloud or plume is washed into the Des Plaines River, by calculating the concentrations of activities in the River, and by comparing these concentrations with the maximum permissible concentrations given in NBS-52.

Estimates are also made of the maximum ground concentrations of activity which could occur and of the maximum concentrations of activity in nearby wells. The concentrations in the well water are compared to the maximum permissible concentrations given in NBS-52.

C. External Radiation Hazard. If an observer stands in the plume until the building stops leaking, or under the cloud until it has passed overhead, he will get an external beta and gamma dose. If the observer remains in an area upon which airborne activity has been deposited, he will also get an external beta and gamma-radiation dose. The maximum air and ground concentrations of beta and gamma activity which could occur are estimated and the radiation doses from these concentrations are calculated.

Inhalation Hazard:

Following a reactor disaster, gaseous and small particulate debris in the reactor room could escape into the atmosphere within a few minutes. In this case a hot cloud would be formed which would rise to some height and then travel downwind at that height, diffusing as it travels. The height to which the cloud would rise depends upon the atmospheric conditions. In radiation hazard estimates, a cloud height of 1300 meters (4300 ft) will be taken as the expected average daytime height. A cloud height of 300 meters (1000 ft) will be taken as the expected average night-time height and as the expected average daytime height during adverse daytime conditions. Because the primary interest is in the hazards which exist one mile or more from the disaster, the cloud can be considered as originating from an instantaneous point source directly above the reactor building and at the assumed cloud height. Let

$Q \equiv$ amount of radioisotope which escapes from reactor building
in microcuries (column 3 in Table V)

$B \equiv$ breathing rate = 2.32×10^{-4} (meters)³/sec (NBS-52)

f $\equiv$ fraction of inhaled isotope which reaches critical organ
(column 8 in Table V)

$\frac{X(d,t)}{Q}$ $\equiv$ relative dilution of material in cloud at distance d downwind
at time t in microcuries per cubic meter per microcurie

Then the total amount of a specific isotope accumulated in the critical organ during passage of the cloud overhead should be less than

$$Q B f \int_0^{\infty} \frac{X(d,t)}{Q} dt \equiv Q B f \text{ (T.I.D.) in microcuries.}$$

The total integrated dose (T.I.D.) is obtained from O. G. Sutton's expression for the diffusion of a cloud coming from an instantaneous point source by calculating the cloud distribution at the instant that it is overhead and then letting a cloud with this distribution pass over the observer from horizon to horizon. By differentiating and maximizing the T.I.D. expression, one obtains the following expressions for the maximum T.I.D. (for a $1\text{-}\mu\text{C}$ source, $Q = 1\text{ }\mu\text{C}$) and the distance downwind (from the disaster) at which this maximum T.I.D. occurs:

$$\text{T.I.D.}_{\text{max}} = \frac{2}{\pi \bar{u} h^2 e}; \quad d_{\text{max dose}} = \left(\frac{h}{C} \right)^{\frac{2}{2-n}}$$

where h is the cloud height, n is Sutton's stability parameter, $\bar{u}$ is the mean wind speed in meters per second, and C is Sutton's generalized diffusion coefficient (cf. Meteorology and Atomic Energy, p. 154).

For an unfavorable wind speed of 1 meter/sec, a stability parameter of 0.5 (indicating stable atmospheric conditions), cloud heights of 300 meters and 1300 meters, and diffusion coefficients of 0.45 and 0.1, the $\text{T.I.D.}_{\text{max}}$ and the $d_{\text{max dose}}$ are given below:

n	h (meters)	C	$\bar{u}$ (meters/sec)	$\text{T.I.D.}_{\text{max}}$ (sec/meter ³)	$d_{\text{max dose}}$
0.5	300	0.45	1	2.62×10^{-6}	5.82 km = 3.62 miles
0.5	300	0.10	1	2.62×10^{-6}	43.3 km = 26.9 miles
0.5	1300	0.45	1	1.39×10^{-7}	41.2 km = 25.6 miles
0.5	1300	0.10	1	1.39×10^{-7}	306 km = 190 miles

Using these values of $\text{T.I.D.}_{\text{max}}$, the maximum amounts of the radioisotopes listed in Table V which an observer could accumulate in the critical organ have been calculated and are given in Table VI. Table VI also gives the

Table VI

ACCUMULATION OF FISSION PRODUCTS FROM A CLOUD BY INHALATION

Radioisotope	Maximum Permissible Amount of Radioisotope in Critical Organ (in microcuries)	Cloud Height 300 Meters		Cloud Height 1300 Meters	
		Maximum Amount of Radioisotope Accumulated in Critical Organ (in microcuries)	Maximum Amount Accumulated ÷ Maximum Permissible Amount	Maximum Amount of Radioisotope Accumulated in Critical Organ (in microcuries)	Maximum Amount Accumulated ÷ Maximum Permissible Amount
Critical Organ: Kidneys					
Ru ¹⁰⁶ + Rh ¹⁰⁶	0.16	9.01 × 10 ⁻⁷	5.6 × 10 ⁻⁶	4.78 × 10 ⁻⁸	3.0 × 10 ⁻⁷
Rh ¹⁰⁵	0.72	5.34 × 10 ⁻⁴	7.4 × 10 ⁻⁴	2.84 × 10 ⁻⁵	3.9 × 10 ⁻⁵
Te ¹²⁷	0.8	4.10 × 10 ⁻⁷	5.2 × 10 ⁻⁷	2.18 × 10 ⁻⁸	2.7 × 10 ⁻⁸
Te ¹²⁹	0.26	7.37 × 10 ⁻⁶	2.9 × 10 ⁻⁵	3.92 × 10 ⁻⁷	1.5 × 10 ⁻⁶
Critical Organ: Bone					
Ga ⁷²	6.56	3.01 × 10 ⁻⁸	4.6 × 10 ⁻⁹	1.60 × 10 ⁻⁹	2.4 × 10 ⁻¹⁰
Sr ⁸⁹	1.4	1.26 × 10 ⁻³	9.0 × 10 ⁻⁴	6.70 × 10 ⁻⁵	4.8 × 10 ⁻⁵
Sr ⁹⁰ + Y ⁹⁰	0.7	8.15 × 10 ⁻⁶	1.2 × 10 ⁻⁵	4.32 × 10 ⁻⁷	6.2 × 10 ⁻⁷
Y ⁹¹	9.75	8.44 × 10 ⁻⁴	8.7 × 10 ⁻⁵	4.48 × 10 ⁻⁵	4.6 × 10 ⁻⁶
Mo ⁹⁹	25.0	2.46 × 10 ⁻⁵	9.8 × 10 ⁻⁷	1.31 × 10 ⁻⁶	5.2 × 10 ⁻⁸
Ba ¹⁴⁰ + La ¹⁴⁰	4.80	6.31 × 10 ⁻³	1.3 × 10 ⁻³	3.35 × 10 ⁻⁴	7.0 × 10 ⁻⁵
Ce ¹⁴⁴ + Pr ¹⁴⁴	4.0	1.22 × 10 ⁻⁴	3.1 × 10 ⁻⁵	6.46 × 10 ⁻⁶	1.6 × 10 ⁻⁶
Pr ¹⁴³	17.4	1.20 × 10 ⁻³	6.9 × 10 ⁻⁵	6.39 × 10 ⁻⁵	3.7 × 10 ⁻⁶
Pm ¹⁴⁷	84.0	1.08 × 10 ⁻⁵	1.3 × 10 ⁻⁷	5.74 × 10 ⁻⁷	6.8 × 10 ⁻⁹
Sm ¹⁵¹	273	2.26 × 10 ⁻⁷	8.3 × 10 ⁻¹⁰	1.20 × 10 ⁻⁸	4.4 × 10 ⁻¹¹
Critical Organ: Thyroid					
I ¹³¹	0.06	3.48 × 10 ⁻³	5.8 × 10 ⁻²	1.85 × 10 ⁻⁴	3.1 × 10 ⁻³

ratio of the maximum amount of the radioisotope which could be accumulated in the critical organ to the maximum permissible amount of the radioisotope in the critical organ. It can be seen that the inhalation hazard due to the radioactive cloud is small; the only radioisotope accumulated in appreciable amount is I^{131} , which has a 7.7-day effective half-life. For a cloud height of 300 meters, the observer could accumulate only 5.8% of the maximum permissible amount of I^{131} in the thyroid.

Because the meteorological parameters C and n drop out in the process of deriving $T.I.D._{max}$, the total integrated dose is quite insensitive to atmospheric conditions and should be correct to an order of magnitude for almost all atmospheric conditions. Doses will be less if the cloud puff meanders. If the cloud height were 100 meters instead of 300 meters, the amounts of radioisotopes accumulated would be greater by a factor of 9 but still not a serious radiation hazard. The accumulated amounts of radioisotopes could be greater by a factor of 10 if the wind speed were 0.1 meter per second instead of 1 meter per second. As before, this would still not be a serious radiation hazard; moreover, it would take about 4.5 hours for the cloud to travel 1 mile to the boundary of the territory controlled by the Laboratory. A wind speed of 0.1 meter per second persisting for more than 4 hours is extremely rare and 4.5 hours should be a sufficient time to move people in danger of being exposed to the radioactive cloud.

If the gaseous and small particulate debris in the reactor room escape slowly into the atmosphere over a period of $1/2$ hour up to several days, then a plume of radioactive material will extend downwind from the reactor building. Again, because the primary interest is in the hazard which exists one mile or more downwind from the disaster, the plume can be considered as coming from a continuous point source at ground level. If it is assumed that the observer remains in the center of the radioactive plume until the building stops leaking, then it is not necessary to specify leak rates. If the building should leak at a uniform rate for three hours and the observer should remain in the plume for only one hour, then his dose would be $1/3$ of the dose estimated here. Let Q , B , and f be defined as in the previous estimate.

$F(d) \equiv$ concentration of radioisotope in microcuries per cubic meter at a distance d meters downwind from a continuous point source on the ground which emits one microcurie per second.

$(\Delta Q / \Delta t) \equiv$ leakage rate of isotope from building in microcuries per second.

$Q(\Delta t / \Delta Q) \equiv$ exposure time, i.e., the time observer remains in the plume until the building stops leaking.

Then the total amount of a specific isotope accumulated in the critical organ by inhalation during exposure in the plume should be less than

$$B Q f (\Delta t / \Delta Q) F(d) (\Delta Q / \Delta t) = B f Q F(d) .$$

$F(d)$, the relative concentration of fission products in the plume, is taken from Table XXIII of Appendix H (Climatology) in ANL-5647.

Using the values of $F(d)$ for the stable case ($n = 0.5$; $C_y = 0.10$; $C_z = 0.06$; wind speed = 2 meters/sec), the amounts of the isotopes listed in Table V which an observer could accumulate in the critical organ have been calculated and are given in Table VII for downwind distances of 3280 ft, 1 mile, and 5 miles. Table VII also gives the ratio of the amount of isotope which could be accumulated in the critical organ to the maximum permissible amount of the isotope in the critical organ.

As would be expected, the internal radiation hazard is considerably greater in this case than in the case of a cloud passing overhead. At a distance of one mile an observer could accumulate as much as 17 times the maximum permissible amount of I^{131} in his thyroid. I^{131} has an effective half-life of 7.7 days (effective mean life of 11.1 days). The total dose to the thyroid resulting from the rapid accumulation of 17 times the maximum permissible amount of I^{131} followed by the 7.7-day decay is given by

$$\int_0^{\infty} (17) (0.3 \text{ rep/week}) (1 \text{ week}/7 \text{ days}) \exp (-t \text{ days}/11.1 \text{ days}) dt$$

$$= (17) (0.3) (1/7) (11.1) \text{ rep} = 8.1 \text{ rep} ,$$

and this is about one third of the "once in a lifetime dose" for exposure of the whole body as defined in NBS Handbook-59. This dose should produce no detectable clinical effects.

At a distance of one mile the observer could accumulate as much as 27% of the maximum permissible amount of Sr^{89} and as much as 39% of the maximum permissible amount of $Ba^{140} + La^{140}$ in his bones; he could accumulate as much as 22% of the maximum permissible amount of Rh^{105} in his kidneys. The radiation dose from the Sr^{89} , $Ba^{140} + La^{140}$, and Rh^{105} would not be serious provided the observer does not accumulate large amounts of other activities not considered here. The Sr^{89} , $Ba^{140} + La^{140}$, and Rh^{105} decay with effective half-lives of 52 days, 12 days, and 1.4 days, respectively.

The amounts of radioisotopes accumulated during exposure in the plume could be greater by a factor of 20 if the wind speed were 0.1 meter/sec instead of 2 meters/second. It would then take about 4.5 hours for

Table VII
ACCUMULATION OF FISSION PRODUCTS FROM A PLUME BY INHALATION

Radioisotope	Maximum Permissible Amount of Radioisotope in Critical Organ in microcuries	3280 ft Downwind		1 Mile Downwind		5 Miles Downwind	
		Amount of Isotope Accumulated in Critical Organ in microcuries	Amount Accumulated $\div$ Maximum Permissible Amount	Amount of Isotope Accumulated in Critical Organ in microcuries	Amount Accumulated $\div$ Maximum Permissible Amount	Amount of Isotope Accumulated in Critical Organ in microcuries	Amount Accumulated $\div$ Maximum Permissible Amount
Critical Organ: Kidneys							
Ru ¹⁰⁶ + Rh ¹⁰⁶	0.16	5.50 x 10 ⁻⁴	3.4 x 10 ⁻³	2.68 x 10 ⁻⁴	1.7 x 10 ⁻³	2.41 x 10 ⁻⁵	1.5 x 10 ⁻⁴
Rh ¹⁰⁵	0.72	3.26 x 10 ⁻¹	4.5 x 10 ⁻¹	1.59 x 10 ⁻¹	2.2 x 10 ⁻¹	1.43 x 10 ⁻²	2.0 x 10 ⁻²
Te ¹²⁷	0.8	2.50 x 10 ⁻⁴	3.1 x 10 ⁻⁴	1.22 x 10 ⁻⁴	1.5 x 10 ⁻⁴	1.10 x 10 ⁻⁵	1.4 x 10 ⁻⁵
Te ¹²⁹	0.26	4.52 x 10 ⁻³	1.7 x 10 ⁻²	2.20 x 10 ⁻³	8.5 x 10 ⁻³	1.97 x 10 ⁻⁴	7.6 x 10 ⁻⁴
Critical Organ: Bone							
Ga ⁷²	6.56	1.82 x 10 ⁻⁵	2.8 x 10 ⁻⁶	8.90 x 10 ⁻⁶	1.4 x 10 ⁻⁶	7.98 x 10 ⁻⁷	1.2 x 10 ⁻⁷
Sr ⁸⁹	1.4	7.72 x 10 ⁻¹	5.5 x 10 ⁻¹	3.76 x 10 ⁻¹	2.7 x 10 ⁻¹	3.38 x 10 ⁻²	2.4 x 10 ⁻²
Sr ⁹⁰ + Y ⁹⁰	0.7	4.96 x 10 ⁻³	7.1 x 10 ⁻³	2.42 x 10 ⁻³	3.5 x 10 ⁻³	2.17 x 10 ⁻⁴	3.1 x 10 ⁻⁴
Y ⁹¹	9.75	5.14 x 10 ⁻¹	5.3 x 10 ⁻²	2.50 x 10 ⁻¹	2.6 x 10 ⁻²	2.24 x 10 ⁻²	2.3 x 10 ⁻³
Mo ⁹⁹	25.0	1.50 x 10 ⁻²	6.0 x 10 ⁻⁴	7.34 x 10 ⁻³	2.9 x 10 ⁻⁴	6.58 x 10 ⁻⁴	2.6 x 10 ⁻⁵
Ba ¹⁴⁰ + La ¹⁴⁰	4.8	3.84	0.80	1.87	0.39	1.68 x 10 ⁻¹	3.5 x 10 ⁻²
Ce ¹⁴⁴ + Pr ¹⁴⁴	4.0	7.42 x 10 ⁻²	1.9 x 10 ⁻²	3.62 x 10 ⁻²	9.0 x 10 ⁻³	3.25 x 10 ⁻³	8.1 x 10 ⁻⁴
Pr ¹⁴³	17.4	7.35 x 10 ⁻¹	4.2 x 10 ⁻²	3.58 x 10 ⁻¹	2.1 x 10 ⁻²	3.22 x 10 ⁻²	1.8 x 10 ⁻³
Pm ¹⁴⁷	84	6.60 x 10 ⁻³	7.9 x 10 ⁻⁵	3.22 x 10 ⁻³	3.8 x 10 ⁻⁵	2.88 x 10 ⁻⁴	3.4 x 10 ⁻⁶
Sm ¹⁵¹	273	1.38 x 10 ⁻⁴	5.1 x 10 ⁻⁷	6.70 x 10 ⁻⁵	2.5 x 10 ⁻⁷	6.01 x 10 ⁻⁶	2.2 x 10 ⁻⁸
Critical Organ: Thyroid							
I ¹³¹	0.06	2.13	35	1.04	17	9.30 x 10 ⁻²	1.5

the plume to travel one mile to the edge of the area controlled by the Laboratory. As pointed out before, a wind speed of 0.1 meter/sec persisting for more than 4 hours is extremely rare, and 4.5 hours should be a sufficient time to move people out of danger of exposure to the radioactive plume.

It should be pointed out that the meteorological parameters for stable conditions which give these relatively high concentrations of activity downwind also give a very narrow plume. At a distance of one mile, the plume width would be 253 ft and the height would be 76 ft. The height and width are defined as the distances from the axis of the plume at which the concentration is 1/10 of that at the axis. Thus the plume being considered here could be a hazard only along a narrow path of the order of 200 to 1000 ft wide and several miles long. For meteorological parameters giving better diffusion and larger cloud widths, the concentration of activity would drop correspondingly.

Finally, in a disaster involving 10^{20} fissions it is likely that the building would be destroyed and a cloud, rather than a plume, would be formed.

The hazard due to a cloud is considerably less than that from a plume. For a disaster in which the building would not be destroyed and a radioactive plume would be formed, there would probably be considerably less than 10^{20} fissions and correspondingly less fission product activity formed.

Ingestion Hazard:

Estimates of the maximum ingestion hazard to downstream communities were made by assuming that all of the activity in the radioactive cloud or plume is brought down on the Laboratory site by precipitation and surface drained into the Des Plaines River. As explained in Appendix C of this report, almost all of the rainfall at the Laboratory site is surface drained by several small creeks into the Des Plaines River (minimum flow rate of 10 cu ft/sec), which passes about one mile to the south of the reactor building. There is very little precipitation penetration because of the type of soil on the Laboratory site. The Des Plaines River joins the Chicago Sanitary and Ship Canal (average annual flow rate of about 3400 cu ft/sec) below the Laboratory site.

The concentrations of the radioisotopes listed in Table V which might occur in the Des Plaines River and the Sanitary and Ship Canal have been calculated for uniform rates of draining of the activity into the River over periods of 30 minutes, 300 minutes, and 3000 minutes (2 days). Table VIII gives the ratios of the calculated concentrations to the maximum permissible concentrations for continuous ingestion for a lifetime as given in NBS-52.

Table VIII
ACCUMULATION OF FISSION PRODUCTS BY SUSTAINED INGESTION

Radioisotope	Maximum Permissible Con- centration in Drinking Water in $\mu\text{c}/\text{ml}$ for con- tinuous exposure	Des Plaines River			Sanitary & Ship Canal		
		$\left[\begin{array}{c} \text{Concentration} \div \\ \text{Maximum Permissible} \\ \text{Concentration} \end{array} \right]$			$\left[\begin{array}{c} \text{Concentration} \div \\ \text{Maximum Permissible} \\ \text{Concentration} \end{array} \right]$		
		Drainage Time			Drainage Time		
		30 min	(5 hrs) 300 min	(2 days) 3000 min	30 min	(5 hrs) 300 min	(2 days) 3000 min
Critical Organ Kidneys							
Ru ¹⁰⁶ + Rh ¹⁰⁶	0.1	2.9×10^{-3}	2.9×10^{-4}	2.9×10^{-5}	8.5×10^{-6}	8.5×10^{-7}	8.5×10^{-8}
Rh ¹⁰⁵	1.5×10^{-2}	6.8	0.68	0.068	0.020	2.0×10^{-3}	2×10^{-4}
Te ¹²⁷	3×10^{-2}	2.2×10^{-3}	2.2×10^{-4}	2.2×10^{-5}	6.5×10^{-6}	6.5×10^{-7}	6.5×10^{-8}
Te ¹²⁹	10^{-2}	0.12	1.2×10^{-2}	1.2×10^{-3}	3.5×10^{-4}	3.5×10^{-5}	3.5×10^{-6}
Critical Organ Bone							
Ga ⁷²	9	1.1×10^{-7}	1.1×10^{-8}	1.1×10^{-9}	3.2×10^{-10}	3.2×10^{-11}	3.2×10^{-12}
Sr ⁸⁹	7×10^{-5}	260	26	2.6	0.78	0.078	7.8×10^{-3}
Sr ⁹⁰ + Y ⁹⁰	8×10^{-7}	150	15	1.5	0.44	0.044	4.4×10^{-3}
Y ⁹¹	0.2	0.097	9.7×10^{-3}	9.7×10^{-4}	2.9×10^{-4}	2.9×10^{-5}	2.9×10^{-6}
Mo ⁹⁹	14	0.032	3.2×10^{-3}	3.2×10^{-4}	9.3×10^{-5}	9.3×10^{-6}	9.3×10^{-7}
Ba ¹⁴⁰ + La ¹⁴⁰	2×10^{-3}	51	5.1	0.51	0.15	0.015	1.5×10^{-3}
Ce ¹⁴⁴ + Pr ¹⁴⁴	4×10^{-2}	0.098	9.8×10^{-3}	9.8×10^{-4}	2.9×10^{-4}	2.9×10^{-5}	2.9×10^{-6}
Pr ¹⁴³	0.4	0.15	0.015	1.5×10^{-3}	4.5×10^{-4}	4.5×10^{-5}	4.5×10^{-6}
Pm ¹⁴⁷	1	3.9×10^{-4}	3.9×10^{-5}	3.9×10^{-6}	1.1×10^{-6}	1.1×10^{-7}	1.1×10^{-8}
Sm ¹⁵¹	0.2	7.3×10^{-5}	7.3×10^{-6}	7.3×10^{-7}	2.2×10^{-7}	2.2×10^{-8}	2.2×10^{-9}
Critical Organ Thyroid							
I ¹³¹	3×10^{-5}	2500	250	25	7.4	0.74	0.074

Although the concentrations of activity in the canal are rather large (for 30 minutes of drainage time) and the concentrations in the river are very large, there are a number of reasons why these concentrations do not seem to be a serious ingestion hazard. In the first place, the water is not used for drinking purposes until it has been diluted by the Kankakee River and the Mississippi River; this should reduce the concentration ratios by at least two orders of magnitude. Some of the water might leak into shallow wells along the banks of the river and the canal, but this would happen only for a length of time comparable to the drainage time of the activity at the Laboratory site. If appreciable amounts of activity could enter the ground water supply in this fashion, then appreciable amounts of river or canal water could enter (after the passing of the activity downstream) to dilute the activity in the ground water.

In the second place, the river and the canal travel about 20 miles per day. In the unlikely event that large concentrations of activity should occur in the water, it would take about 10 hours for the activity to reach the Illinois State Prison (where the water is used for irrigation purposes) and it would take about 14 days for the activity to reach Alton, Illinois (where the water is first used for drinking purposes). Thus there would be ample time to prevent use of any contaminated water. Furthermore, relatively short-lived activity will decay out during the time it takes the river to travel to the observer; e.g., $(14/8) = 1.7$ or more half-lives of I^{131} will elapse before the activity gets to Alton, Illinois. Furthermore, no serious amount of long-lived activity should be accumulated in the body because a particular observer could ingest the radioactive water only for a time comparable to the drainage time of the activity at the Laboratory site.

Finally, it is unlikely that such large concentration ratios as those given in Table VIII would occur in the first place. It is possible that most of the activity in the reactor room could leak out and be deposited on the Laboratory site by precipitation within as short a time as 30 minutes. However, the time it takes for the rainwater to be drained into the river and the canal should be considerably longer. Then, too, the movement of fission product activity along the ground into the river and the canal should be considerably slower than the movement of the rainwater carrying this activity. Increasing the drainage time decreases the concentration in the river and canal water, allows short-lived activity to decay to a greater extent, and gives radiation safety personnel more time in which to warn downstream communities of a possible hazard.

In the case of activity which is deposited off the Laboratory site by precipitation, the activity will be spread over a larger area because the cloud or plume carrying the activity expands as it travels downwind. Not as much activity will be washed into the river or the canal and the time required to drain the activity into the river or the canal should be considerably longer.

Fission product activity could give rise to an ingestion hazard not only to downstream communities, but also to communities where the activity is initially deposited. In this case the hazard would be due to activity which gets into the wells supplying drinking water. Although this hazard does not appear serious, some idea of the extent of the hazard can be obtained by calculating the maximum ground concentrations of activity which could occur and then estimating the concentration of activity which might occur in the well water on the basis of some pessimistic estimate of the rate at which the surface activity gets into the well.

At considerable distances from the disaster, the largest ground concentrations of activity will occur as a result of precipitation (as opposed to dry deposition). If it is raining at a steady rate from the time of the disaster until some time after the cloud or plume has passed the observer, there will be a certain scavenging rate which produces the maximum deposition at a distance d downwind from the disaster. For this "optimum" scavenging rate, the maximum deposition due to steady washout is given by

$$W = \frac{Q}{e^{\pi^{1/2}} C_y d^{(2 - \frac{n}{2})}}$$

where W is the deposition rate in microcuries per square meter per second for a continuous source of Q microcuries per second or the total deposition for an instantaneous source of Q microcuries; C_y is Sutton's diffusion coefficient and n is Sutton's stability parameter.

An upper limit to the possible ground concentration of activity is obtained by assuming that the activity in a radioactive cloud is totally and instantaneously deposited in the area of interest. The maximum deposition due to total instantaneous washout is given by

$$W = \frac{Q}{\pi C^2 d^{2-n}},$$

where W is the ground concentration in microcuries per square meter for an instantaneous source of Q microcuries; C is Sutton's generalized diffusion coefficient and n is Sutton's stability parameter (cf. Meteorology and Atomic Energy, p. 156).

The relative ground concentrations of activity due to steady deposition at the "optimum" scavenging rate and due to total instantaneous washout have been calculated for various distances downwind with $n = 0.5$ and $C \approx C_y \approx 0.45$ and 0.1 . These concentrations are given as follows; they are quite pessimistic since it is unlikely that n would be as great as 0.5 and C and C_y would be as small as 0.1 under ordinary precipitation conditions.

d (meters)	W/Q (Steady washout at "optimum" scavenging rate) (meter ⁻²)		W/Q (Total instan- taneous washout) (meter ⁻²)	
	C = 0.45	C = 0.10	C = 0.45	C = 0.10
800m (1/2 mile)	3.83×10^{-6}	1.72×10^{-5}	6.93×10^{-5}	1.40×10^{-3}
1600m (1 mile)	1.14×10^{-6}	5.13×10^{-6}	2.46×10^{-5}	4.96×10^{-4}
3200m (2 miles)	3.39×10^{-7}	1.52×10^{-6}	8.66×10^{-6}	1.76×10^{-4}
6400m (4 miles)	1.01×10^{-7}	4.53×10^{-7}	3.06×10^{-6}	6.20×10^{-5}
12800m (8 miles)	2.99×10^{-8}	1.35×10^{-7}	1.09×10^{-6}	2.20×10^{-5}

For the case of total instantaneous washout with $C = 0.45$ and $n = 0.5$, the ground concentrations of the radioisotopes listed in Table V have been calculated and are given in Table IX. Concentrations of the radioisotopes in the ground water have been estimated, based on the arbitrary assumption that it takes one to two months for the activity to wash into a well and that the activity is diluted by one to two months of rain (say 10 centimeters of rainfall) in the process. The ratios of these estimated concentrations to the maximum permissible concentrations of the radioisotopes in water (NBS-52) are given in Table IX.

At a distance of 3200 meters (2 miles) downwind, the concentrations of Sr^{89} , $\text{Sr}^{90} + \text{Y}^{90}$, $\text{Ba}^{140} + \text{La}^{140}$, and I^{131} are 12, 6.6, 2.2, and 110 times the maximum permissible concentrations in water for continuous exposure. In order to estimate the radiation dose resulting from such concentrations, it has been arbitrarily assumed that it will take one month for the activity to get into a well and that contaminated water will be ingested for the following month. Furthermore, the decay of the radioisotopes, not previously taken into account, will also be considered here.

The effective half-life of $\text{Sr}^{90} + \text{Y}^{90}$ in the body is 2.7×10^3 days or 7.4 years; the effective mean life in the body is 3.90×10^3 days. If the maximum permissible concentration of $\text{Sr}^{90} + \text{Y}^{90}$ in water were ingested for a lifetime, one could accumulate the maximum permissible amount of $\text{Sr}^{90} + \text{Y}^{90}$ in the body. If 6.6 times the maximum permissible concentration were ingested for 30 days, one could accumulate $6.6 \times (30 \text{ days} / 3.90 \times 10^3 \text{ days})$ or 5.1% of the maximum permissible amount in the body.

Twenty-eight days after the disaster, radioactive decay will have reduced the concentration of $\text{Ba}^{140} + \text{La}^{140}$ to about 22% of the concentration given in Table IX. Thus the concentration would be 0.48 instead of 2.2 times the maximum permissible concentration for continuous ingestion, and it would continue to decrease with a 12.8-day half-life.

The physical half-life of Sr^{89} is 53 days; the physical half-life of I^{131} is 8 days. Since the physical half-lives are 1.4% and 4.5% respectively, of

Table IX

GROUND CONCENTRATION OF FISSION PRODUCTS FOR TOTAL INSTANTANEOUS WASHOUT

Radioisotope	Maximum Permissible Con- centration in $\mu\text{c}/\text{ml}$ of H_2O for Continuous Exposure	1600 Meters Downwind (1 Mile)		3200 Meters Downwind (2 Miles)		6400 Meters Downwind (4 Miles)	
		Ground Concentration of Activity in $\mu\text{c}/\text{m}^2$	Ratio of Estimated Con- centration in H_2O to Maximum Permissible Concentration	Ground Concentration of Activity in $\mu\text{c}/\text{m}^2$	Ratio of Estimated Con- centration in H_2O to Maximum Permissible Concentration	Ground Concentration of Activity in $\mu\text{c}/\text{m}^2$	Ratio of Estimated Con- centration in H_2O to Maximum Permissible Concentration
Critical Organ: Kidneys							
Ru ¹⁰⁶ + Rh ¹⁰⁶	0.1	3.64	3.64×10^{-4}	1.28	1.28×10^{-4}	0.453	4.53×10^{-5}
Rh ¹⁰⁵	1.5×10^{-2}	1.27×10^3	0.85	449	0.30	158	0.11
Te ¹²⁷	3×10^{-2}	0.831	2.8×10^{-4}	0.293	9.8×10^{-5}	0.103	3.4×10^{-5}
Te ¹²⁹	10^{-2}	15.0	1.5×10^{-2}	5.27	5.3×10^{-3}	1.86	1.9×10^{-3}
Critical Organ: Bone							
Ga ⁷²	9	1.22×10^{-2}	1.4×10^{-8}	4.30×10^{-3}	4.8×10^{-9}	0.00152	1.7×10^{-9}
Sr ⁸⁹	7×10^{-5}	2.32×10^2	33	82	12	28.9	4.1
Sr ⁹⁰ + Y ⁹⁰	8×10^{-7}	1.50	19	0.528	6.6	0.186	2.3
Y ⁹¹	0.2	2.44×10^2	0.012	85.9	0.0043	30.3	0.0015
Mo ⁹⁹	14	5.53×10^3	0.004	1950	0.0014	689	4.9×10^{-4}
Ba ¹⁴⁰ + La ¹⁴⁰	2×10^{-3}	1.27×10^3	6.4	449	2.2	158	0.79
Ce ¹⁴⁴ + Pr ¹⁴⁴	4×10^{-2}	4.92×10	0.012	17.3	0.0043	6.12	0.0015
Pr ¹⁴³	0.4	7.75×10^2	0.019	273	0.0068	96.4	0.0024
Pm ¹⁴⁷	1	4.86	4.9×10^{-5}	1.72	1.7×10^{-5}	0.606	6.1×10^{-6}
Sm ¹⁵¹	0.2	1.83×10^{-1}	9.2×10^{-6}	0.0645	3.2×10^{-6}	0.0228	1.1×10^{-6}
Critical Organ: Thyroid							
I ¹³¹	3.0×10^{-5}	9.4×10^2	310	331	110	117	39

the biological half-lives, the biological half-lives will be taken as infinitely long in the following estimates of the radiation dose from Sr^{89} and I^{131} .

Twenty-eight days after the disaster, radioactive decay will have reduced the concentrations of Sr^{89} and I^{131} from 12 and 110 to 8.3 and 11.6 times the maximum permissible concentrations in water, respectively. If an observer begins ingesting the contaminated water 28 days after the disaster, the amount of radioisotope accumulated in his body (in microcuries) at time t (in days) after he begins ingesting the water is approximately given by

$$\text{Sr}^{89}: A(t) = 8.3(\text{MPA}) \left(\frac{0.693 t}{53 \text{ days}} \right) \exp \left(- \frac{0.693 t}{53 \text{ days}} \right)$$

$$\text{I}^{131}: A(t) = 11.6(\text{MPA}) \left(\frac{0.693 t}{8 \text{ days}} \right) \exp \left(- \frac{0.693 t}{8 \text{ days}} \right)$$

where (MPA) is the maximum permissible amount of Sr^{89} or I^{131} (in microcuries) in the body as specified in NBS-52. If the observer ingests the contaminated water for a period of 30 days he will receive a total dose of 1.60 rep in his bones from the Sr^{89} and a total dose of 4.2 rep in his thyroid from the I^{131} . Moreover, he will receive doses of 7.2 rep and 1.1 rep, respectively, while the Sr^{89} and the I^{131} decay out after the 30-day ingestion period. These doses are less than half of the "once in a lifetime" dose defined in NBS-59; they should produce no detectable clinical effects.

The arbitrary assumptions that the deposited activity would be diluted by only 10 centimeters of rainwater, that the activity would take about one month to wash into a well, and that an observer would ingest the water for only one month must be questioned. Consideration of the movement of deposited activity into wells is quite complicated because of local variations in the predominant structure of the ground in the region around the Laboratory. For example, the soil on the Laboratory site is very impervious to the flow of ground water. Rough measurements indicate that it would take 100 to 150 years for water introduced at the surface to penetrate the 100 ft to 150 ft of yellow-gray silt loam and get into the Niagara Dolomite sandstone layer from which the Laboratory gets its water. On the other hand, there are villages and towns in the region around the Laboratory which lie on considerable thicknesses of gravel or other material which allows the relatively free flow of ground water. Towns such as Naperville (9 miles distant), Wheaton (13 miles distant), and Warrenville (13 miles distant) are in regions of permeable soils. Here it should be noted that the greater distances involved would reduce the estimated ground water concentrations of activity by at least a factor of ten.

Other local variations which complicate the picture are gravel pits, abandoned wells, and wells which are shallow or defective. There are some deep gravel pits in the region around the Laboratory. It might be possible for contaminated rainwater to be surface drained from a considerable area

into a gravel pit from which point the contaminated water could enter porous strata underlying relatively impervious strata above. In towns which have put in municipal wells, many abandoned wells may not be properly capped or sealed; if such wells exist, they could offer surface water a relatively easy access to the lower water-bearing strata from which the municipal wells are currently drawing water. Finally, wells which are not properly sealed as they pass through strata impervious to the flow of water may be drawing some of their water from the surface.

There would be no ingestion hazard to Chicago and many of its suburbs which lie to the east of the Laboratory site because their source of water is Lake Michigan. Many nearby communities have excellent deep wells (1500 ft to 2000 ft deep) and, here again, the ingestion hazard is negligible. Some nearby communities do not have such excellent or such deep wells. However, the distance from the disaster will cut the estimated concentrations by a factor of ten, or more. Then, too, any surface water which is pulled by a defective well will be considerably diluted by the water which the well pulls from the lower strata which are not contaminated. Private wells nearby range from excellent to miserable. Some private wells are effectively only 18 ft deep in sandy soil and occasionally get polluted by a neighboring septic tank.

The assumption that it would take about one month for activity to get into a well seems to be rather pessimistic for most of the wells in the region, with the exception of some poorly constructed nearby private wells. The assumption that contaminated water is ingested for one month seems reasonable since this ought to be ample time in which to form a radiation safety group to monitor the water supply in order to prevent ingestion of contaminated water in a community where there was a total instantaneous washout of activity from a reactor disaster. The assumption that the activity deposited on the ground is diluted by only 10 centimeters of water also seems pessimistic. In general, the activity would be diluted by more than one or two months of rain and also by the water already in the ground.

External Radiation Hazard:

In estimating the external radiation dose from exposure in a radioactive plume, the decay of the fission product activity will be neglected during the time the activity is confined in the reactor building; this is equivalent to assuming that the plume escapes quite rapidly from the building. It will also be assumed that at the observer's position the fission product energy absorbed per second per meter³ of air is equal to the fission product energy emitted per second per meter³ of air. The absorption (and by assumption, the emission) of 6.77×10^{10} Mev per meter³ of air at the observer's position corresponds to a dose to the observer of 1 rep (the difference between the air density at the observer and that at sea level will be neglected).

The assumption that the energy emitted is equal to the energy absorbed per unit volume per second is quite good if the concentration of fission product activity in the plume is uniform over distances of the order of the "range" of the beta and gamma rays. The range of the beta rays is of the order of 5 ft and the concentration of activity in the plume is probably constant over this distance. However, the nominal range of the gamma rays is of the order of 300 meters, while the width and height of the radioactive plume could be 253 ft and 76 ft at a distance of one mile under stable conditions. Therefore the gamma dose will be estimated by use of a nomogram given in Meteorology and Atomic Energy, p. 104.

In the case of beta radiation let

$Q \equiv 8 \times 10^{19} \times t^{-1.21}$ Mev/sec = beta activity which gets into the radioactive plume where $t = d/\bar{u}$.

$(\Delta Q/\Delta t) \equiv$ leakage rate of activity from the building in Mev/sec/sec.

$Q(\Delta t/\Delta Q) \equiv$ exposure time, i.e., the time an observer remains in the plume until the building stops leaking.

$F(d) \equiv$ concentration of beta activity in Mev/sec/meter³ at a distance d downwind from a continuous point source on the ground emitting 1 Mev per second of beta activity each second;

$K \equiv$ conversion factor = 1 rep/(6.8×10^{10} Mev per meter³)

$G \equiv$ geometry factor: For beta radiation, a value of $G = (0.5)(0.64) = (0.32)$ will be used because the beta flux at the surface of the skin is 1/2 that in air and because the beta particles will enter the observer's feet from a solid angle of 2π and his head from more nearly a solid angle of 4π (an average value of 0.64 is given in Meteorology and Atomic Energy).

Then the beta dose from exposure in a radioactive plume is given by

$$F(d) (\Delta Q/\Delta t) Q (\Delta t/\Delta Q) K G = F(d) Q K G \quad \text{in rep}$$

The beta dose has been calculated for several distances downwind and is given in Table X. The values of $F(d)$ used are those for which $n = 0.5$, $C_y = 0.10$, $C_z = 0.06$ and the windspeed is $\bar{u} = 2$ meters/sec (4.5 miles/hour). Because the dose at a given distance downwind increases as $(\bar{u})^{0.21}$

for short-lived fission product activity, the dose has also been given for a wind speed of $\bar{u} = 15$ meters/sec (33.5 miles/hr). The beta doses given in Table X have been reduced by a factor of 10 to take account of the shielding effect of clothing.

For wind speeds of 30 miles per hour, the atmospheric diffusion conditions are probably much better than has been assumed (n is more like 0.25 than 0.5). Also, a rapid escape of the fission product activity such as has been assumed here should result in considerable turbulence and in a cloud which rises some 300 to 1300 meters, in which case the external radiation doses should be less because of the lower concentration of fission product activity at the position of the observer. On the other hand, if the release of activity to the atmosphere is slow (say 15 minutes or more), then the activity will decay considerably during containment in the reactor building and the doses should be smaller. Moreover, in a disaster in which the building is not completely destroyed and in which a radioactive plume is formed, there would probably be considerably less than 10^{20} fissions and correspondingly less fission product activity to give rise to an external radiation hazard.

As was mentioned before, the estimated gamma dose was obtained by using a nomograph prepared by J. Z. Holland (cf. Meteorology and Atomic Energy, p. 104). This nomograph takes into account the size and shape of the cloud puff, the attenuation of the gamma rays by the air, the decay of the fission product activity, and the multiple scattering buildup factor. The external gamma ray doses at several distances downwind from radioactive clouds at several heights are given in Table XI for $n = 0.5$, $C = 0.45$, and $\bar{u} = 1$ meter/sec (2.24 miles/hr) and 15 meters/sec (33.5 miles/hr). The dose at one mile from the disaster from a cloud for which $h = 0$, $\bar{u} = 15$ meters/sec and $C = 0.45$ is 4.0 roentgens.

An observer can also get an external radiation dose from exposure to activity deposited on the ground. As was mentioned above, the largest ground concentrations of activity will occur as a result of precipitation (at considerable distances from the disaster). Let

$$Q = 8 \times 10^{19} \times t^{-1.21} \text{ Mev/sec} = \text{total beta (or gamma) source strength of the cloud or plume at time } t \text{ (in seconds).}$$

$$(W/Q) = \text{ground concentration of activity in Mev/sec/meter}^2 \text{ for a source strength of } Q \text{ Mev/sec. Maximum values for } W/Q \text{ have already been calculated in the discussion of ingestion hazards.}$$

Table X
EXTERNAL BETA RADIATION DOSE FROM
A RADIOACTIVE PLUME

Distance Downwind	Mean Wind Speed in m/sec	External β Dose in Rep
3280 ft	15	50
3280 ft	2	33
1 mile	15	14
1 mile	2	9
5 miles	15	0.2
5 miles	2	0.1

Table XI
EXTERNAL GAMMA RADIATION DOSE FROM A RADIOACTIVE CLOUD (IN ROENTGENS)

Downwind Distance	External γ Dose in Roentgens					
	Cloud Height h = 0 meters		Cloud Height h = 300 meters		Cloud Height h = 500 meters	
	$\bar{u}$ = 15 m/sec	$\bar{u}$ = 1 m/sec	$\bar{u}$ = 15 m/sec	$\bar{u}$ = 1 m/sec	$\bar{u}$ = 15 m/sec	$\bar{u}$ = 1 m/sec
800 meters ($\frac{1}{2}$ mile)	13	8.0	0.48	0.30	0.06	0.04
1600 meters (1 mile)	4.0	2.1	0.21	0.12	0.06	0.04
2400 meters (1.5 miles)	1.4	0.80	0.16	0.08	0.01	0.01
3200 meters (2 miles)	0.51	0.30	0.16	0.08	0.01	0.01
4000 meters (2.5 miles)	0.40	0.24	0.16	0.08	0.01	0.01

Table XII
EXTERNAL RADIATION DOSE DUE TO WASHOUT

Distance Downwind	External Radiation Dose (in Rep)				% of Total Dose Obtained in the First Three Hours for 15 m/sec Windspeed	Radius of Contaminated Area (in meters)
	(Washout at "Optimum" Scavenging Rate)		(Total Instantaneous Washout)			
	γ -dose	β -dose	γ -dose	β -dose		
800 meters (½ mile)	96	960	1730	17300	63%	102
1600 meters (1 mile)	25	250	530	5300	57%	172
3200 meters (2 miles)	6.3	63	161	1610	51%	289
6400 meters (4 miles)	1.6	16	49	490	43%	484

E_γ , E_β are conversion factors which give the beta and gamma dosage rates per unit ground concentration of beta and gamma activity; the values for E_γ and E_β are those used in Wash III:

$E_\gamma = 1.3 \times 10^{-4}$ roentgens/day for a ground concentration of 1 Mev of gamma activity/sec/cm² or 1.51×10^{-13} roentgens/sec for a ground concentration of 1 Mev of gamma activity/sec/meter².

$E_\beta = 1.3 \times 10^{-2}$ roentgens/day for a ground concentration of 1 Mev of beta activity/sec/cm² or 1.51×10^{-11} roentgen/sec for a ground concentration of 1 Mev of beta activity/sec/meter².

The total external beta and gamma doses received by an observer who remains in the center of the washout area until the activity has decayed to a negligible amount are given by the following expressions:

$$\text{gamma dose} = \int_{t_0}^{\infty} Q (W/Q) E_\gamma dt = 5.75 \times 10^7 (W/Q) t_0^{-0.21}$$

$$\text{beta dose} = \int_{t_0}^{\infty} Q (W/Q) E_\beta dt = 5.75 \times 10^9 (W/Q) t_0^{-0.21} ,$$

where t_0 is the time of travel from the disaster to the observer.

The total external beta and gamma doses have been calculated for ground concentrations of activity due to washout at the "optimum" scavenging rate and also for total instantaneous washout; these doses are given in Table XII. The values of (W/Q) used are those for which $n = 0.5$ and $C = 0.45$. A wind speed of 15 meters/sec was used. Reducing the wind speed from 15 to 1 meter/sec reduces the doses to about 57% of the doses given in Table XII. The beta doses have been arbitrarily reduced by a factor of ten to account for the shielding effect of the clothing which the observer wears. Column 6 of Table XII gives the fraction of the total dose (for exposure for an infinite time) which is received in the first three hours after the disaster when the wind speed is 15 meters/sec. Column 7 gives the radius at which the ground concentration of activity drops to 1/10 of the maximum concentration which was used in calculating the doses in columns 2, 3, 4, and 5.

The external radiation hazard due to total instantaneous washout is probably not as serious as the estimated doses would indicate. The area in which the hazards exists is rather small as can be seen from the last column of Table XII. Actually, for wind speeds of 10 to 15 meters/sec, the

ground concentrations of activity due to total instantaneous washout at one to two miles from the disaster will be smaller and the area in which the hazard exists will be larger, than the estimated values. For example, suppose that it takes two minutes for a heavy rain storm to bring to the ground all of the activity in cloud puff. If the wind speed is 10 meters/sec, the cloud will travel 1200 meters during the time that the activity is being deposited. At one mile from the disaster, the contaminated area will be roughly $2 \times 172 \times 900 = 3.10 \times 10^5$ meters² instead of $\pi(172)^2 = 9.29 \times 10^4$ meters². Then the seriously contaminated area will be about 3.3 times as large, but the ground concentration of activity (and also the external radiation dose) will be about 30% of that calculated by assuming that the activity is washed to the ground instantaneously. Then too, the assumption that 50% of the fission product activity escapes from the reactor room and travels one to two miles downwind without fallout on the way is rather pessimistic.

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APPENDIX F

MULTIPLICATION - FLOW INTERLOCK

To comply with the recommendation of the AEC's Hazards Evaluation Staff that the changeover from high to low fill rate near criticality be guaranteed by the incorporation of an interlock or alarm in the system, a neutron multiplication-flow interlock has been added to the ZPR-VII system.

The multiplication - water flow interlock is designed to prevent the reactor from becoming critical on high water flow rate. Normally, the operator will initiate high flow in filling the reactor tank and will switch to low flow when approaching the desired water level. This level will be one estimated as requiring subsequent withdrawal of control rods to achieve criticality. Should the reactor inadvertently be in a condition where criticality could occur during the addition of water, a divergent condition could result rapidly if the water flow rate is high. The multiplication - water flow interlock will prevent this by automatically stopping the high flow rate (if the flow switch is on "high") when the neutron level reaches a multiplication level just below critical. This is initiated by a normally closed switch actuated by the slide-wire mechanism on the log level recorder, Channel V (see Figure 18). Further addition of water to the system will require that the operator switch the pump to the low fill rate. The circuit will continue to prevent the addition of water at the high flow rate unless the neutron flux level drops below the set value. This switch can be adjusted to operate anywhere over the log level range. The level at which the switch from high flow rate occurs will be established on the basis of operating experience. It will be set to occur at a flux level corresponding to a subcritical assembly. Because the neutron source decays with time, this level will be periodically adjusted in the downward direction to insure that the switch-over always occurs before criticality has been reached.

If the flow is already in the "low" condition, no action takes place, and the flow continues. Here the rate of addition of reactivity is sufficiently slow that adequate time exists for action by the operator or the scram circuits.

The interlock, in either condition, does not prevent the operator from turning the water flow off.

FIGURE 18
NEUTRON LEVEL – FLOW RATE INTERLOCK