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PLUTONIUM RECYCLE TEST REACTOR

FINAL SAFEGUARDS ANALYSIS

THE STAFF OF REACTOR ENGINEERING DEVELOPMENT OPERATION

OCTOBER 1, 1959

HANFORD LABORATORIES

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PLUTONIUM RECYCLE TEST REACTOR
FINAL SAFEGUARDS ANALYSIS

By

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ABSTRACT

The Plutonium Recycle Test Reactor (PRTR) is designed to obtain experimental data on plutonium fuel technology and plutonium fuel cycle physics, and to produce irradiated fuel for the development of plutonium fuel chemical processing. Its main purpose is to provide a pilot scale demonstration of the economics and practicability of various plutonium recycle fuel concepts. The reactor complex is described in detail. The operating procedures, which are designed to maintain a high degree of reactor safety, are described. Reactor safeguards aspects of possible equipment malfunctions and failures are analyzed.

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I. INTRODUCTION

The Plutonium Recycle Test Reactor was constructed under Atomic Energy Commission sponsorship as one of the facilities to be used to develop the technology required for the utilization of plutonium fuels in thermal heterogeneous power reactors. The reactor provides facilities for:

(1) Irradiation testing of plutonium fuel and feed fuel elements,

(2) Direct investigation of reactivity and exposure effects from isotope buildup on the uranium-plutonium fuel cycle,

(3) Production of pilot plant quantities of prototypic irradiated fuels for fuel reprocessing and refabrication studies,

(4) Investigation of control characteristics, reactor dynamics, and reactor operating problems for plutonium recycle operation, and

(5) Providing a convincing demonstration on the pilot plant scale of the economics and practicability of various fuel cycles.

A heavy water moderated and cooled vertical pressure tube type reactor was selected because it gave promise of best meeting the criteria prescribed for the reactor.

The following general criteria were established to delineate a safeguards philosophy upon which the design was based:

1. Containment - Radioactive materials released in a plausible incident must be contained within safe limits.

   (a) The containment shall accommodate the maximum credible accident. *

   (b) The containment shell shall be shielded or otherwise protected against penetration from within by missiles.

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* Maximum credible accident is defined as a forseeable accident based on a chain of events even though having a negligibly small probability of occurrence.

2. Control - The nuclear reactions shall be controlled at all times.

   (a) The rate of reactivity increase is to be limited by system capacity so that reaction time of the scram system will prevent unsafe nuclear excursions.

   (b) The emergency shutdown system shall be capable of overriding a "maximum credible power excursion".

   (c) Systems shall be designed as fail-safe (malfunction to shut-down reactor) to the maximum degree practicable.

3. Process Systems - The simultaneous loss or compound failure of any two system elements shall not put the reactor in jeopardy.

   (a) The loss of any two process components shall not endanger the off plant environs.

   (b) Simultaneous loss of two independent power sources shall not prevent safe shutdown of the reactor.

   (c) Simultaneous loss of two independent cooling (secondary) water systems shall not result in fuel element melting.

4. Fuel Elements - The design shall provide adequate assurance that fuel elements will not melt (see Control and Process Systems above).

   (a) The maximum heat flux shall be conservatively limited to avoid heat transfer burnout during both normal operation and emergency conditions.

   (b) Individual tube coolant flow, temperature, and radioactivity shall be monitored.

   (c) Design and orientation of fuel elements shall be such as to minimize flow irregularities and consequences of inadvertent boiling.

5. Piping and Vessel Standards - Shall conform to applicable codes for high pressure piping and vessels. An exception to this criterion is the process tube design.

6. Radioactive Waste Disposal - Facilities shall provide means of control to insure that the release of radioactive materials is accomplished within prescribed limits for radioactive waste disposal.

A preliminary safeguards analysis, based on the revised PRTR scope and detail design, was issued in June, 1958.[(1)] Since then the final design has been established. Further defined physics constants and continuing safeguards analyses involving more refined and extensive calculations based on the final design permit publication of the final hazards report.

Since publication of the preliminary safeguards analysis, a number of design changes have been made which affect reactor safety. Also, there have been some changes in the safeguards analyses. The most important changes are itemized below:

1. The safety circuit was redesigned to provide dual or triple channels for each trip point. Four trips were eliminated from the safety circuit without decreasing reactor safety; they were
   (a) Low pressure of steam generator feedwater,
   (b) Low level in secondary coolant storage tank,
   (c) Low flow of secondary coolant, and
   (d) High pressure in the containment building.

2. Shim control units were redesigned to minimize the possibility of a rod falling out of the core. The number of rods was increased from 36 to 54 half-rods. Total strength of the shim system was increased to 115 mk.

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(1) Wittenbrock, N. G. Plutonium Recycle Test Reactor, Preliminary Safeguards Analysis, HW-48800 REV. June 5, 1958.

3. Injection of back-up light water coolant was made automatic. A back-up coolant line to the steam generator shell was added.
4. The reflector was changed from light water to heavy water.
5. Additional pressure relief devices were installed on the primary coolant outlet piping.
6. Rupture disks were installed on the calandria gas piping to relieve pressure in the event of a process tube rupture.
7. A fourth bottom gas line was installed on the calandria to increase the initial flow rate of moderator during a scram.
8. The initial fuel element loading was changed from concentric cylinder UO_2 elements to 19-rod cluster elements.
9. A lower Doppler fuel temperature coefficient was used in kinetic and accident analyses.
10. Reactor safety was analyzed for operation only with D_2O coolant. If it is ever decided to operate with light water coolant a supplement to the safeguards analysis will be submitted for review.
11. The maximum credible accident is defined as the loss of primary coolant through a 12-inch diameter rupture rather than through a complete parting of the top 14-inch header. Actually, the consequences are the same except that the peak pressure in the containment vessel would occur about nine seconds earlier for a complete parting of the top 14-inch header.
12. Description of the Plutonium Fabrication Pilot Plant and the review of the site selection analyses were not repeated since they would be identical to the information presented in the preliminary report, HW-48800 REV.

Several test loops are being scoped for installation in the reactor to improve the flexibility of the reactor as an experimental facility. Hazard analyses of these loops will be published in succeeding reports.

Before this reactor is operated the design and plans for operation will also be reviewed by the General Electric Reactor Safeguards Council.

II. SUMMARY

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A. Reactor and Building

The PRTR is a vertical pressure tube type reactor, heavy water moderated and cooled, with a thermal power rating of 70 megawatts. A cutaway view of the reactor and building is shown in Figure 1. Fuel elements are charged into and discharged from the 85 Zircaloy-2 process tubes from the top face. Up to 50 per cent of the fuel elements will be spike enrichment plutonium-aluminum elements and the remainder will be natural UO_2 elements. Goal exposure for the UO_2 fuel elements will be approximately 5000 MWD/T.

Control of the reactor is achieved by regulating the level of the heavy water moderator, which is held in the reactor vessel by a helium gas balance system. Fifty-four shim rods are provided to compensate for local flux perturbations and to permit adjustment of the moderator level to the desired control range for equilibrium operation. Emergency shutdown (scram) is achieved by a gas-balanced moderator dump system which drains the moderator from the calandria at a rate in excess of 20,000 gpm.

Heat is removed from the recirculating heavy water primary coolant by vaporizing light water in a heat exchanger to generate 425 psia steam. The steam will be condensed in a barometric condenser before disposal to the Columbia River. An all welded steel cylindrical containment vessel, 80 feet in diameter, with a hemispherical dome and ellipsoidal bottom, houses the reactor. Over-all height of the containment vessel is 121 feet 6 inches, extending 75 feet above grade. The containment vessel is designed and constructed in accordance with Sections VIII and IX of the ASME Boiler and Pressure Vessel Code (1956 Edition) and Code Cases No. 1226 and No. 1228. The design pressure of the vessel is 15 psig. Normal access to the building will be by a personnel air lock, approximately 10 feet in diameter by 15 feet long.

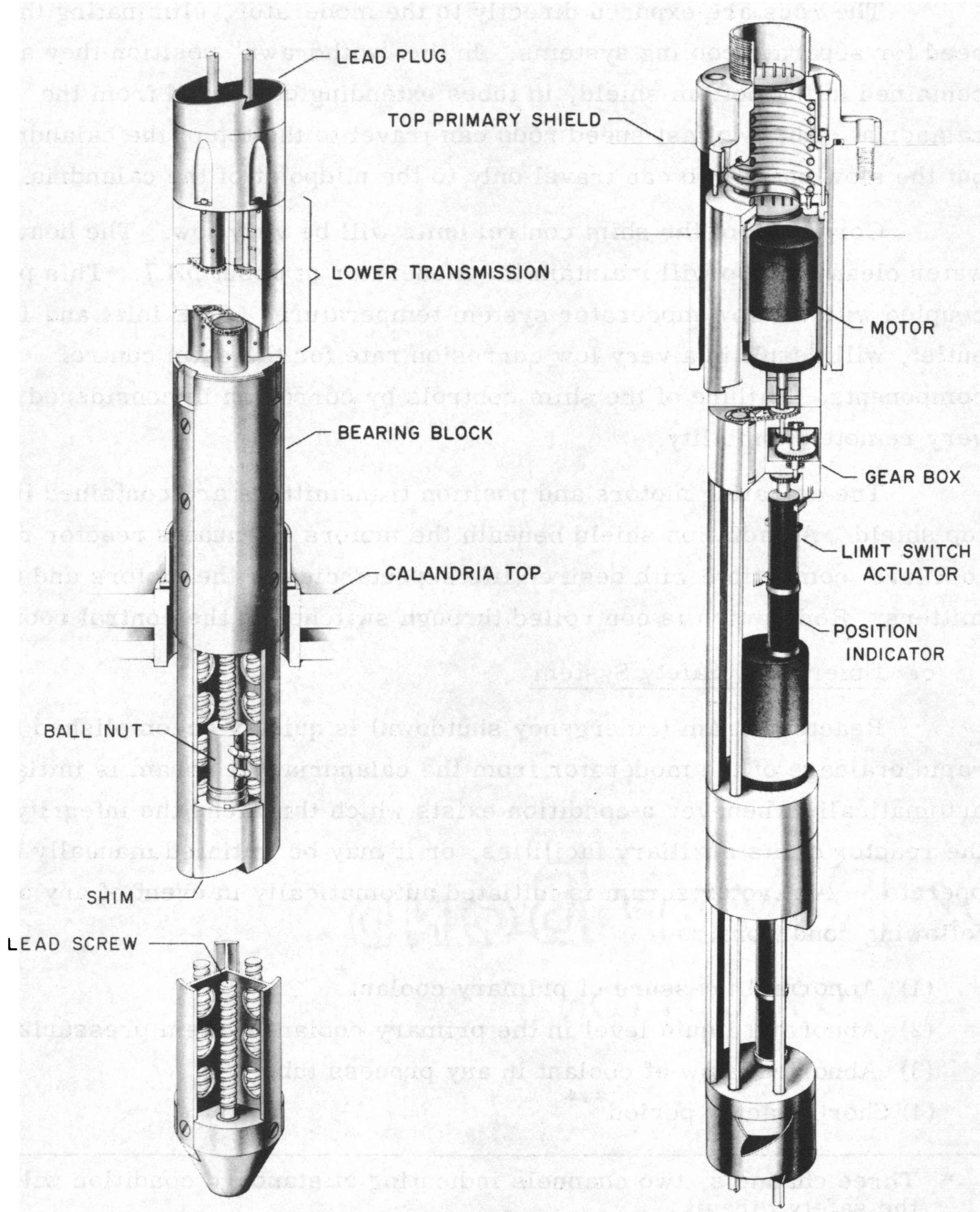

FIGURE 15

Shim Control Unit

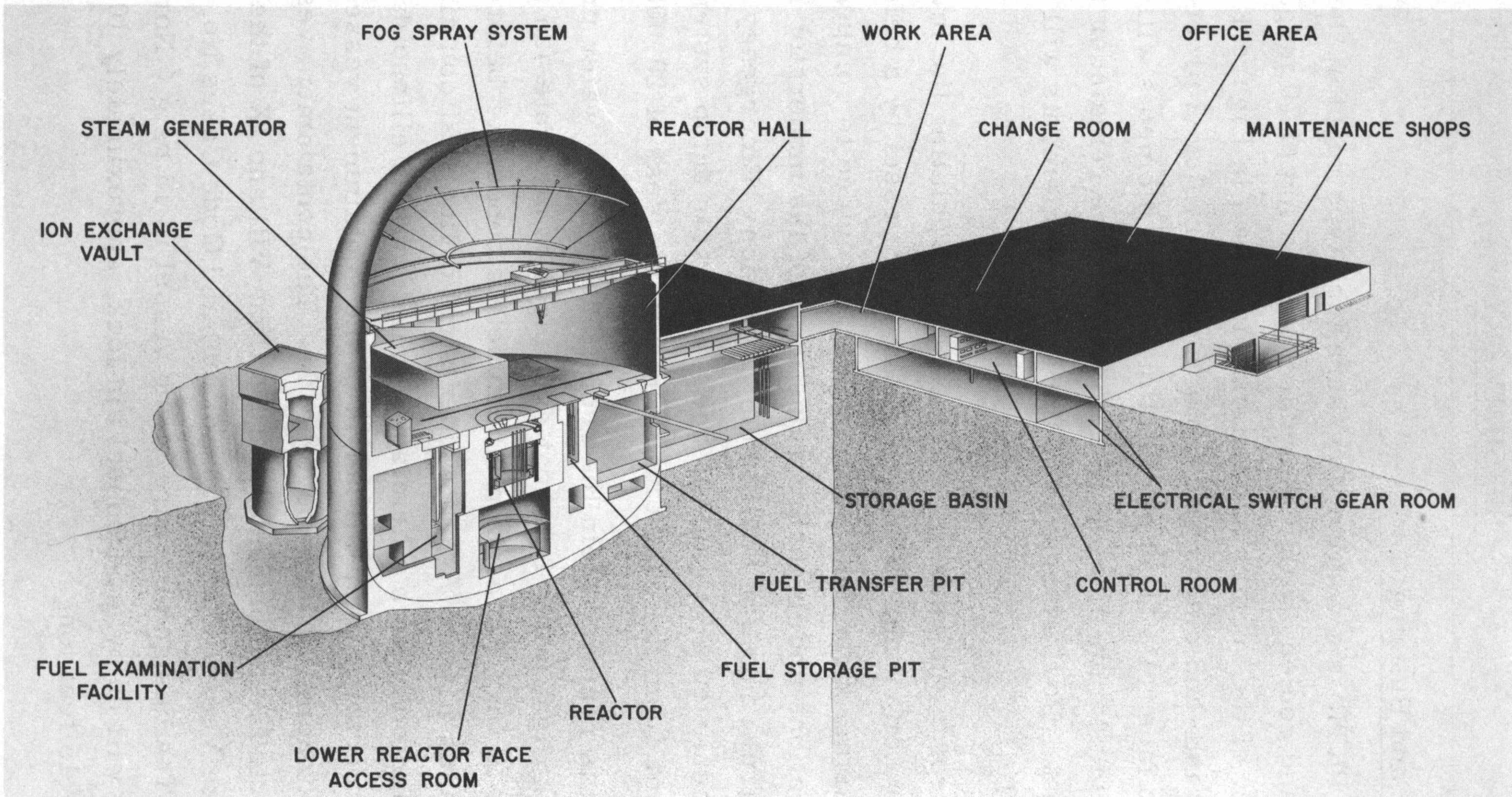

FIGURE 1

Plutonium Recycle Test Reactor Building

Both the fuel and the moderator of the PRTR exhibit negative temperature coefficients. The prompt or Doppler temperature coefficient of the fuel is the chief inherent safety mechanism of the reactor, and is the effect primarily responsible for the termination of major or fast excursions, aside from the safety system. In a slow reactor runaway, moderator effects may also be of significance in limiting the energy release.

B. Safeguards Analysis

Aside from inherent safety, the major safeguards against serious reactor incidents are: fail safe design of circuitry and components as far as practicable; interlocks to prevent unsafe combinations of manual or automatic functions; the use of key locks on controls; and most important, the selection and training of competent operating personnel. The organizational directives to PRTR Operating and Management personnel clearly define areas of responsibility and decision making powers. Procedures for review and approval of process limitations and proposed experiments by responsible parties are specified.

Analysis of the most severe incidents possible in the PRTR shows that at least three simultaneous failures must occur to cause an incident of significant consequences. Possible nuclear excursions were evaluated and it was found that the only credible excursion leading to fuel core melting was a start-up accident in which four simultaneous failures occurred; all shim controls erroneously removed, the three period instruments failing to respond to fast period, the automatic controller malfunctioning, and the high neutron flux instruments being set to trip in the operating power range. Even in this case, the time during which the Pu-Al would be molten is too short to expect Al-Zr diffusion to cause rupture of the Zircaloy jackets. No release of fission products to the atmosphere would occur.

Analysis of conceivable mechanical failures disclosed that the only possible sequence leading to fuel melting required external piping failure plus failure of the emergency coolant supply systems. Actually, this would require coincidental failure of three physically separated systems. In the worst case, the heavy water would be rapidly expelled from the reactor through a break in the 14-inch primary coolant line from the top ring header. Changes in system pressures and flows would shut down the reactor by actuation of the safety scram system. However, the fuel would melt in the dry tubes due to fission product decay heating. Chemical reaction of the melting fuel jackets, plutonium-aluminum alloy cores, and the process tubes would release 2,000,000 Btu. The pressure in the containment vessel would reach 9.9 psig, which is less than the design pressure of the vessel. Escape of fission products to the atmosphere would be limited to the amount carried by the specified maximum leakage of 1000 cu ft/24 hr, and radiation dosage to offsite inhabitants would be minimal.

III. THE REACTOR AND BUILDING

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A. Description of Reactor

1. Reactor Arrangement

The reactor is of the vertical pressure tube type, heavy water moderated and cooled. It has a thermal power rating of 70 megawatts generated in 85 tubes. A cutaway view of the reactor is shown in Figure 2. A description of the reactor and associated systems follows. For more detailed information a summary of engineering data is included as Appendix A.

a. Core and Reflector

The heavy water moderator, which is unpressurized, is contained in an aluminum tank called the calandria. All welded, all aluminum construction is used for the vertical cylinder 84 inches inside diameter and 115 inches inside height. Passing vertically through the calandria are 85 fuel channels, 18 shim control channels, and 13 flux monitor channels. The 85 process tubes are arranged on an 8-inch equilateral triangular lattice. Figures 3 and 4 show horizontal and vertical sections of the calandria.

To insulate the low temperature moderator from the high temperature primary coolant a double wall fuel channel arrangement is used. The outer shroud tube is aluminum and forms an integral part of the calandria. The shroud tubes are welded to the bottom tube sheet and are sealed to the top tube sheet by individual bellows to provide for differential thermal expansion. Eighty-four of the shroud tubes are 4. 250 inches outside diameter with 0. 065-inch wall thickness; the center shroud tube is 6 inches outside diameter with an 0. 085-inch wall thickness. The large central tube will accommodate special experiments.

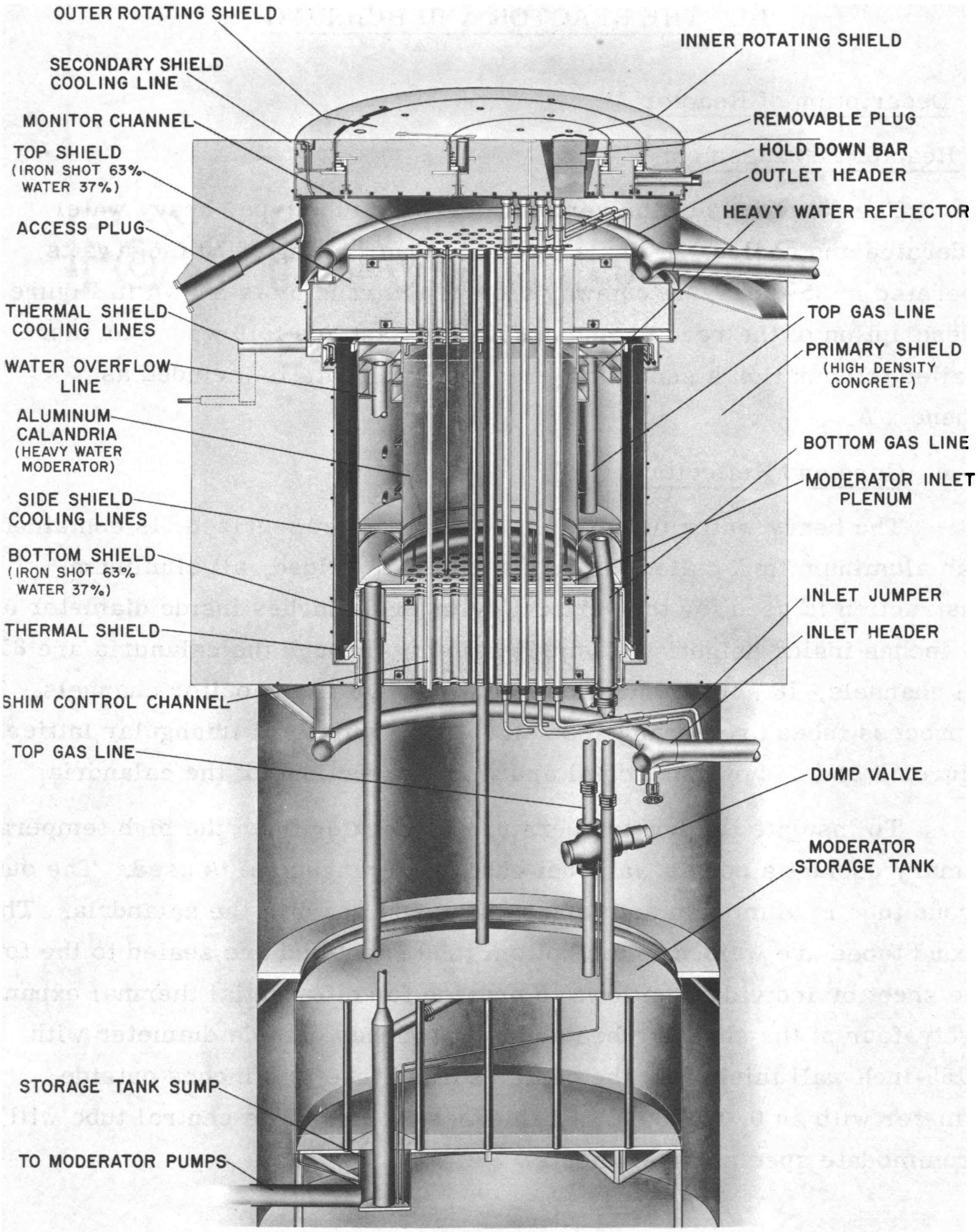

FIGURE 2

Reactor Core and Shielding

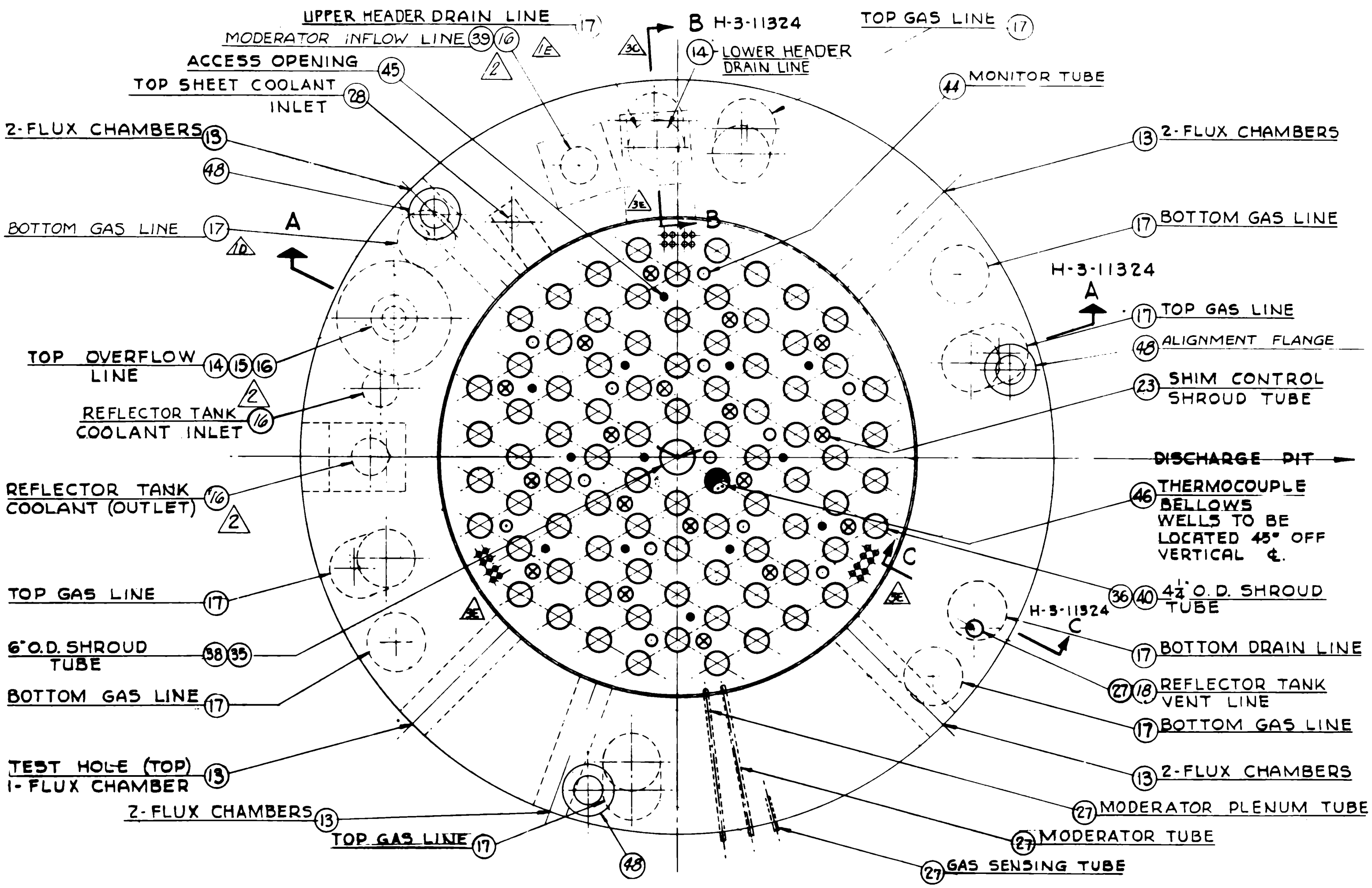

FIGURE 3

Top Plan of Calandria
(H-3-11320)

The 85 vertical process tube assemblies provide pressurized water cooled channels through the reactor in which uranium and plutonium fuel elements can be irradiated. Heavy water coolant flows upward through these tubes; the maximum single-tube flow rate is 123 gpm. The process tubes are supported by shoulders at the upper end resting upon flanges supported by the top shield. Each process tube is connected to ring headers by individual jumpers at the top and bottom faces of the reactor, as shown in Figure 2. This design allows maximum flexibility in the use of the reactor since tubes can be individually monitored or piped to separate cooling systems.

Process tubes are fabricated of Zircaloy-2 and are 3.250 ± 0.010 inches inside diameter with a 0.154 ± 0.008-inch wall thickness in the reactor core. The lower ends of the tubes are tapered to a smaller diameter with greater wall thickness for ease of assembly of the lower face piping. Figure 5 shows the arrangement of the process tubes. The allowable design stress of 14,400 psi is 80 per cent of the stress required to produce a secondary creep rate of 10^{-7} in/(in)(hr) at 500 F in vacuum-annealed Zircaloy-2.[2] Although this allowable design stress corresponds to an operating pressure of 1250 psig, the pressure at the outlet of the process tubes will be limited to 1050 psig until more information of the in-reactor behavior of Zircaloy-2 pressure tubes is available.

Attached to the lower end of the calandria is an annular dump chamber which is connected to the calandria by a water trapped weir. The dump chamber serves two purposes:

(1) It acts as a collecting header for a part of the moderator effluent which is always passing over the weir and which is returned to the moderator storage tank by an 8-inch drain line; and

(2) It acts as a temporary storage volume for a significant fraction of the moderator during a reactor scram.

(2) Fox, J. C. and D. E. Johnson. Design Basis for PRTR Process Tubes, HW-50337. May 24, 1957.

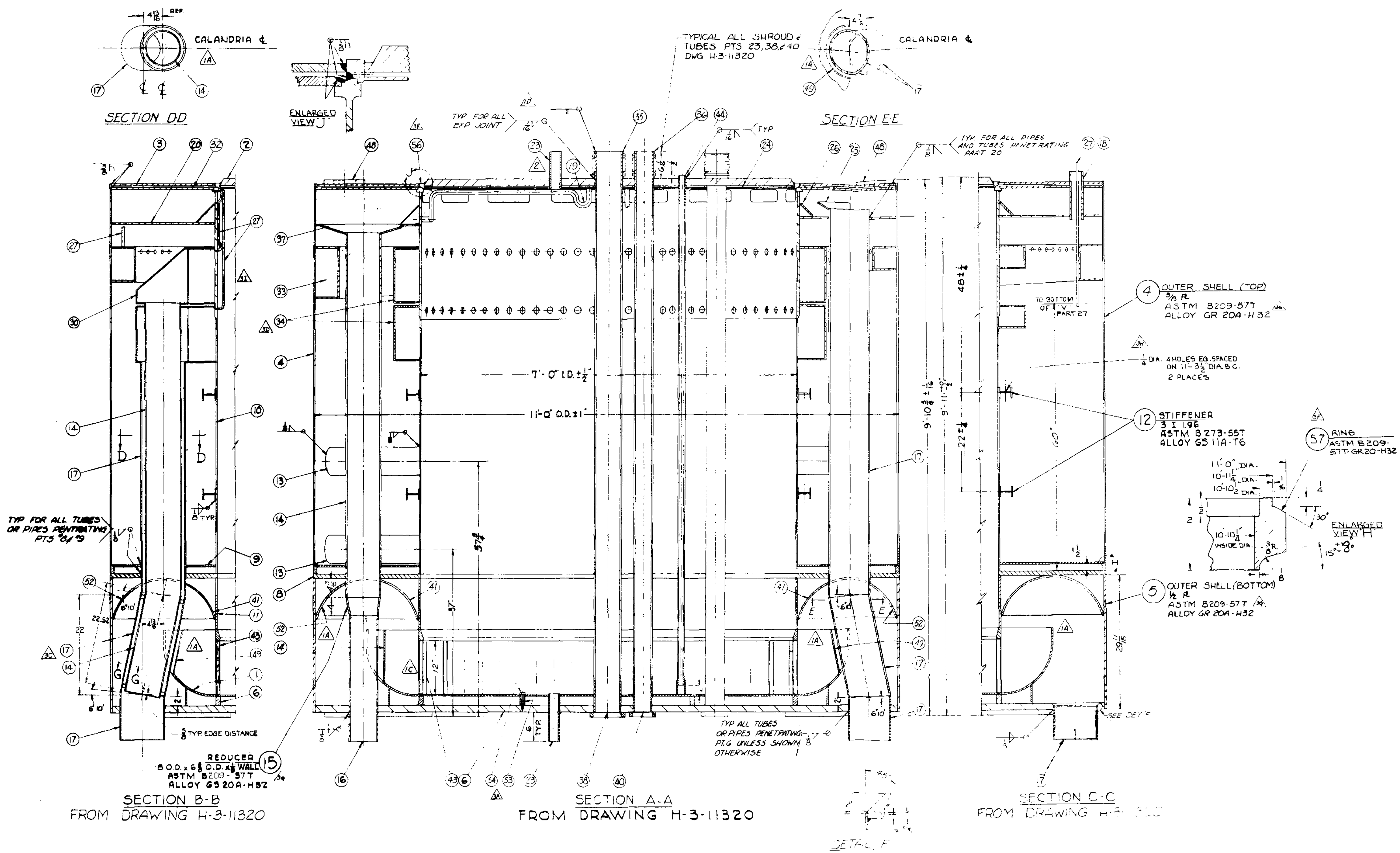

FIGURE 4

Vertical Section of Calandria
(H-3-11324)

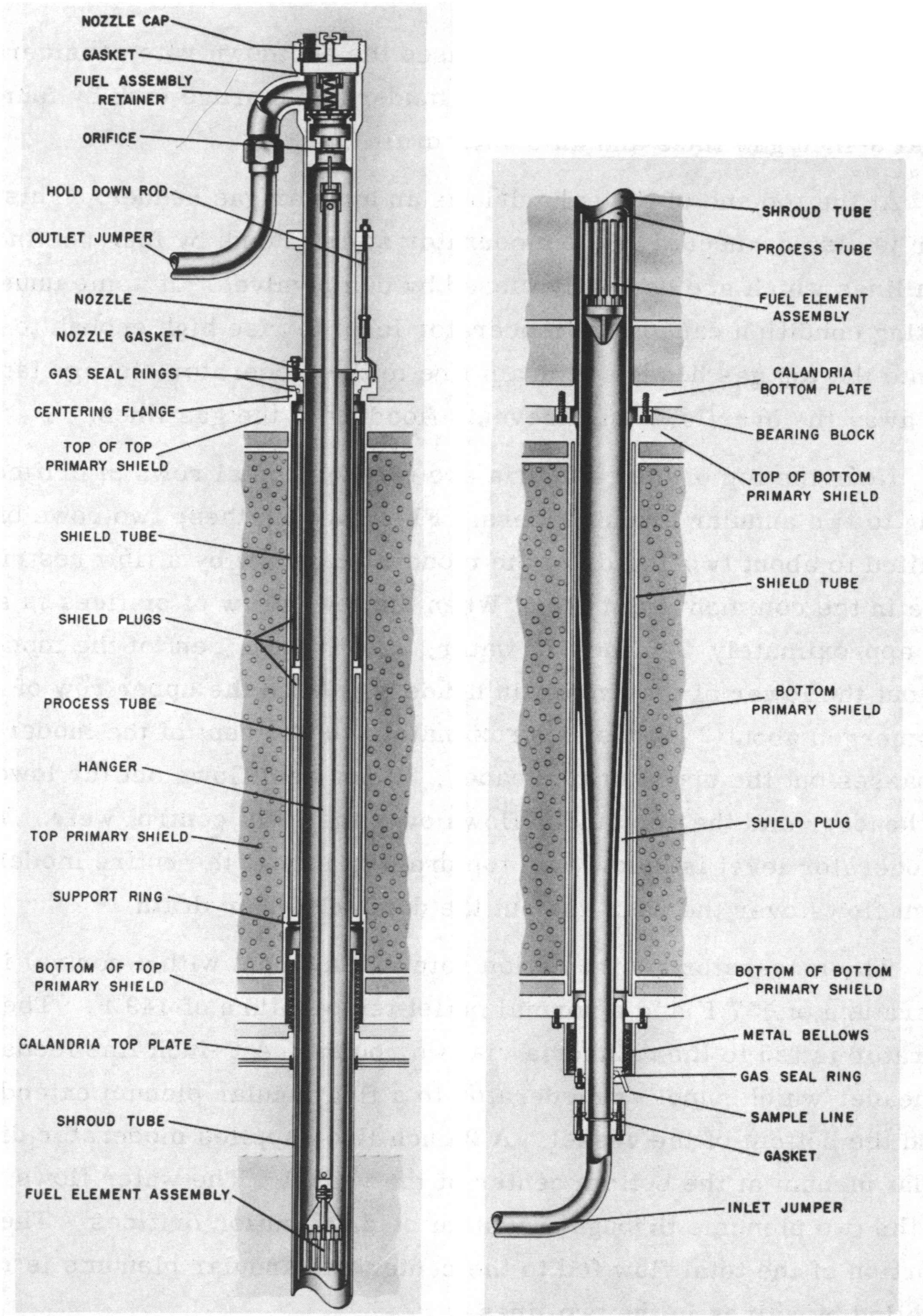

FIGURE 5

Process Tube Assembly

Using this nearby storage space increases the shutdown rate considerably. The dump chamber is connected to the moderator storage tank by four nominal 8-inch gas lines and an 8-inch drain line.

At the top end of the calandria is an annular gas header. This header is also connected to the moderator storage tank by four nominal 8-inch lines which are normally closed by dump valves. If some unusual operating condition causes the moderator level to rise high enough to overflow into the top gas header, a drain line to the moderator storage tank will carry away the overflow; this prevents flooding of the gas lines.

Near the top of the calandria are two horizontal rows of orifices leading to two annular drain headers. Flow through these two rows of orifices is limited to about two-thirds of the moderator inflow by a flow restricting orifice in the common drain line. When the lower row of orifices is submerged under approximately 2 inches of water, about 65 per cent of the moderator flows out the lower of the two drain headers. When the upper row of orifices is submerged about 2 inches, approximately 40 per cent of the moderator flow passes out the upper drain header, 30 per cent flows out the lower drain header, and the remainder flows over the level control weir. When the moderator level is below both top drain headers, the entire moderator effluent flows over the weir and out the dump chamber drain.

The moderator recirculation rate is 1086 gpm with a normal inlet temperature of 137 F and a normal outlet temperature of 149 F. The moderator is fed to the calandria via two routes. A 6-inch line feeds a ring header which supplies moderator to a flat annular plenum extending around the bottom of the vessel. A 2-inch line supplies moderator directly to a flat plenum at the bottom center of the vessel. The water flows upward from the two plenums through a system of distribution orifices. The proportion of the total flow fed to the center and annular plenums is remotely controlled by valves in the two lines.

The water reflector is contained in an annular cylinder 11 feet in outside diameter and 7 feet inside diameter and 6 feet 10 inches high. The vessel surrounds the calandria at the side and fits vertically between the dump chamber at the bottom and the gas header at the top. The vessel is made integral with the calandria with three common walls. The general arrangement is shown in Figure 4.

The reflector is cooled by the recirculation of approximately 176 gpm of heavy water at inlet and outlet temperatures of 137 F and 160 F. The water is fed into the system from the bottom by a 6-inch pipe which terminates at the bottom of the vessel; it feeds into an annular flat plenum which extends around the vessel to provide equal distribution of the water.

The water is drained from the vessel by an annular ring header located about 6 inches below the top of the vessel. The ring header drains into a 6-inch outlet line. Since the reflector vessel acts as the reflector cooling loop surge tank, the vessel water level is controlled to a point a few inches above the drain header.

Spaced around the reflector vessel at two different levels are nine flux chamberholes and one test hole. These facilities consist of 6-inch outside diameter aluminum tubes which penetrate the outer wall of the vessel but not the inner wall.

To facilitate reactor core disassembly, all line connections to the calandria and reflector vessel through the shields are made vertically.

b. Fuel Elements

It is planned that the initial loading will be of the spike enrichment type using two kinds of fuel elements, Mark I with plutonium-aluminum alloy or sintered uranium dioxide cores, and several Mark II elements with sintered uranium dioxide cores. The uranium dioxide is PWR grade extruded or pressed and sintered to a minimum of 94 per cent theoretical density. Plutonium alloy cores will contain plutonium in corrosion resistant aluminum-nickel-silicon alloy. Other plutonium fuel elements, such as uranium dioxide - plutonium dioxide or plutonium salt-ceramic carried cores, will be candidates for investigation during the Plutonium Recycle Program development.

Initially all jackets for fuel elements are Zircaloy-2, although at a later date other sheathing materials may be used.

Fuel element assemblies consist of a cluster of 19 rods, Mark I element, as shown in Figure 6; a configuration of two concentric cylinders and a central rod, Mark II element, as shown in Figure 7, or other as yet undetermined shapes.

In the Mark I 19-rod cluster assembly each fuel element rod has the following dimensions:

Core (bare)	0.504 inches OD
Jacket Thickness	0.030 inches
Core Length	7 feet 4 inches

Twelve of the nineteen rods have a spiral wrapping of 0.072-inch Zircaloy-2 wire to space the elements for adequate coolant flow over all surfaces. The bundle is also wrapped in several places to maintain alignment of the elements. Each element is rigidly fastened to the bracket at each end. Fuel element assemblies are suspended by hangers supported near the top of the process tube assembly.

Mark II concentric fuel element assemblies contain three fuel components arranged as a rod surrounded by two concentric tubes. These fuel elements have the following dimensions:

	Mark II C
Rod (bare)	0.548 inches OD
Inner Tube (bare)	1.082 inches ID 1.782 inches OD
Outer Tube (bare)	2.328 inches ID 2.948 inches OD
Jacket Thickness	0.060 inches
Core Length	7 feet 4 inches

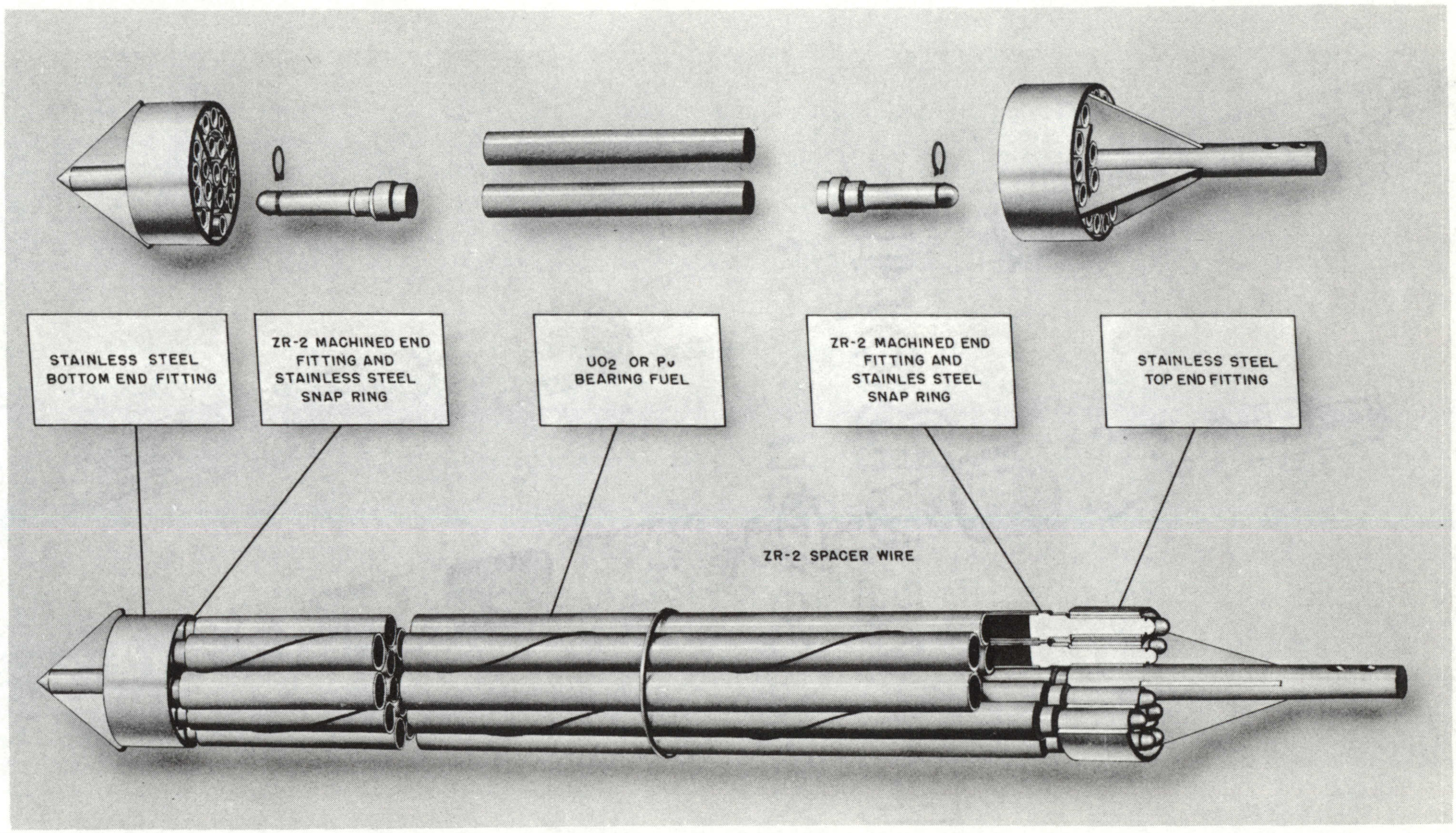

FIGURE 6

Mark I, 19-Rod Cluster Fuel Element Assembly

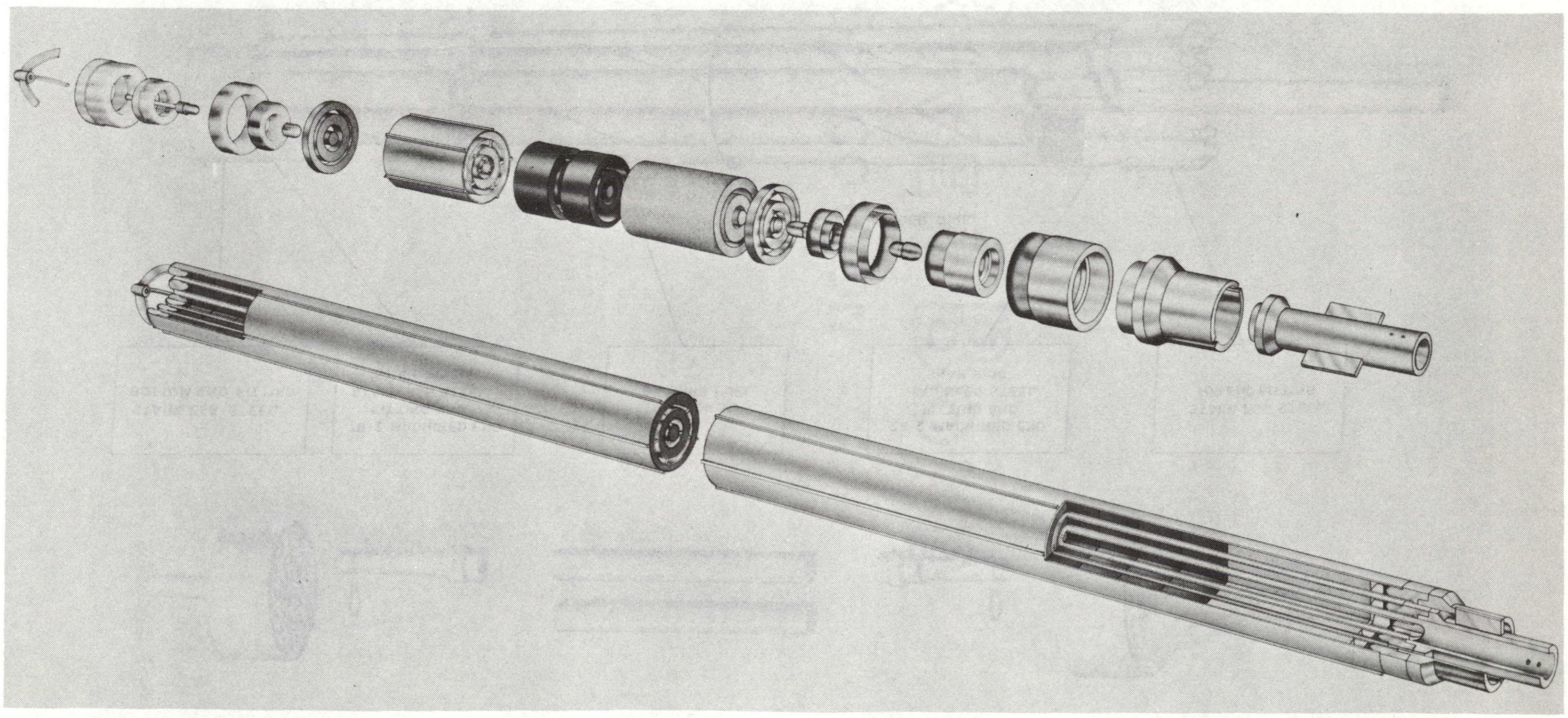

FIGURE 7

Mark II, Concentric Cylinder Fuel Element Assembly

These components are spaced by ribs or projections on the jackets to assure adequate coolant flow over all surfaces. Components are rigidly fastened to a bracket at each end. Fuel element assemblies are suspended by hangers supported near the top of the process tube assembly.

c. Shielding

The biological shield of the PRTR consists of a cylindrical wall of high-density concrete, 71 inches thick and about 21 feet deep. The inside surface is faced with a steel liner, 1/2 inch thick, which also functions as a gas seal for the reactor atmosphere. The outside surface is faced with a 1/4 inch thick steel shell. That portion of the shield forming the inside wall of the process equipment cell is filled with magnetite-limonite concrete having a wet density of about 210 lb/cu ft, while the other half of the shield is filled with iron-limonite concrete having a wet density of 265 lb/cu ft. The inside face of the biological shield is kept below 120 F by circulating water through 1/2-inch pipes attached to the concrete side of the steel liner.

Supported inside the steel liner is the thermal shield consisting of a series of twelve iron slabs, 6 inches thick. The purpose of this auxiliary shield is to remove about 90 per cent of the energy escaping from the reflector before it reaches the concrete biological shield. Individual thermal blocks are about 3 feet wide and 11 feet high, and weight about 4 tons each. They are cooled by a series of vertical cooling tubes, leaded into grooves located on the inside face. The maximum temperature in the iron thermal shield is approximately 150 F. The thermal shield is supported near the bottom by the side biological shield.

The calandria and core components of the reactor rest on the bottom primary shield, which is supported by the steel reinforced base ring of the concrete biological shield. The bottom primary shield, as well as the top primary shield above the calandria, consists of a cylindrical steel tank, 40 inches thick, pierced by process, control, access, and monitoring channels.

The space inside the top and bottom primary shields is filled with a mixture of iron pellets and water, 63 per cent iron and 37 per cent water, by volume. The minimum density of this mixture is approximately 285 lb/cu ft. In addition to its moderating ability, the water functions as a shield coolant which is circulated upward through the shield at about 50 gpm. This water has an inlet temperature of about 112 F and an outlet temperature of about 150 F when the reactor is operating at the design power level.

Directly above the calandria and reactor core is the top primary shield, similar in construction to the bottom shield. The top primary shield supports the process piping assembly and the reactor fuel elements.

Above the top primary shield is the top secondary shield which consists of a stationary ring in which are mounted two rotating disks and an access plug. Both rotating disks are supported on ball bearings. The center disk is located eccentrically in the outer disk so that the eccentrically located access plug can be positioned over any process tube. Details of the secondary shield are shown in Figure 8. Components of the secondary shield are 27 inches thick and consist of steel forms filled with a mixture of steel punchings and magnetite grout having an average density of 320 lb/cu ft. The lower surfaces of the shield components are cooled by circulating water through cooling tubes imbedded in the concrete.

Since a sizable leak in the primary coolant piping below the top secondary shield could build up a pressure great enough to lift the shield, it is necessary to hold down the shield. The stationary ring is held down by six equally spaced radial hold down pins that protrude into sockets in the reactor hall floor slab. Interlocking lugs hold down the two rotating discs and access plug when the shield is rotated to the operating position. The discs must be oriented in the operating position to permit connection of the shield cooling water lines. The hold-down devices are designed to withstand the force exerted by an 18 psi pressure difference.

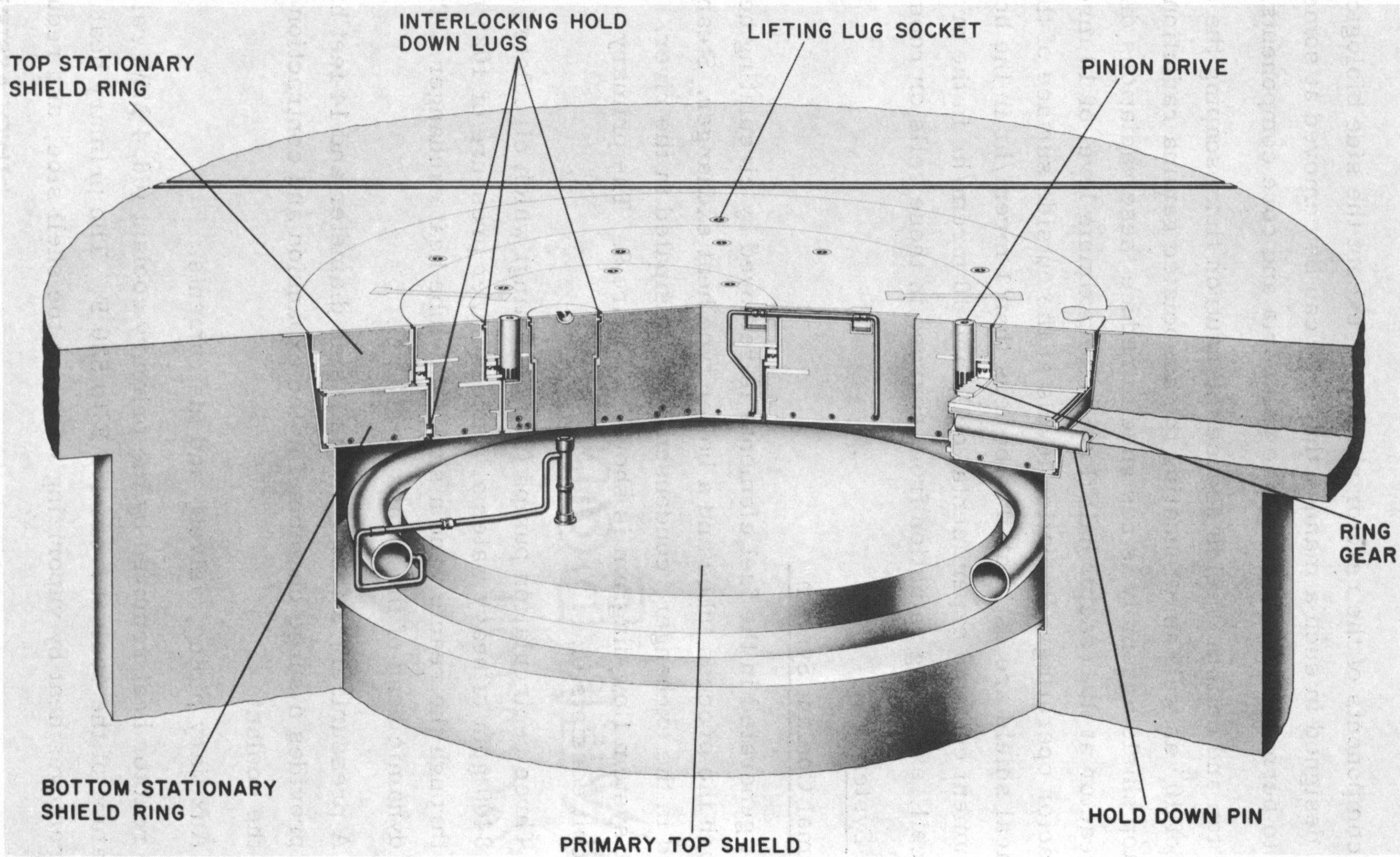

FIGURE 8

Top Secondary Shield

All components of the reactor shielding, except the side biological shield, are designed in such a manner that they can be removed at some future date to permit replacement of the calandria and core components.

The top and bottom shields reduce the neutron flux escaping the core by a factor of 10^6 as well as attenuating the associated gamma radiation. The composite top shield assembly is designed to reduce these radiations directly above the reactor at the reactor hall floor to a maximum level of 10 mrem/hr with the reactor operating. Radiation levels at the outside surface of the side biological shield are estimated to be less than 1 mrem/hr in the hot shop, instrument cell and experimental cell, and 10 mrem/hr in the process equipment cell, excluding radiation from sources in these cells or rooms.

2. Coolant System

a. Normal Coolant System

Heat generated in the fuel elements is removed by circulating heavy water through the process tubes and a boiler type heat exchanger. Steam is generated in the exchanger, condensed, and dissipated in the river. The coolant system flow diagram is shown in Figure 9. The primary coolant circuit consists of:

(1) Three recirculation pumps (two operating) which circulate 8400 gpm of heavy water at a pressurizer pressure of 1025 psig through the reactor and a shell-and-tube heat exchanger. The dynamic head of the pumps is 110 psi.

(2) A pressurizer, 3 feet 6 inches inside diameter and 14 feet high, provides a surge chamber for the expansion and contraction of the coolant.

(3) Auxiliary piping, valves, and instruments.

The reactor heat removed by the primary coolant (66.4 MW) raises the temperature of the coolant from 478 F to 530 F. The primary heat exchanger removes heat by vaporizing water on the shell side, and reduces

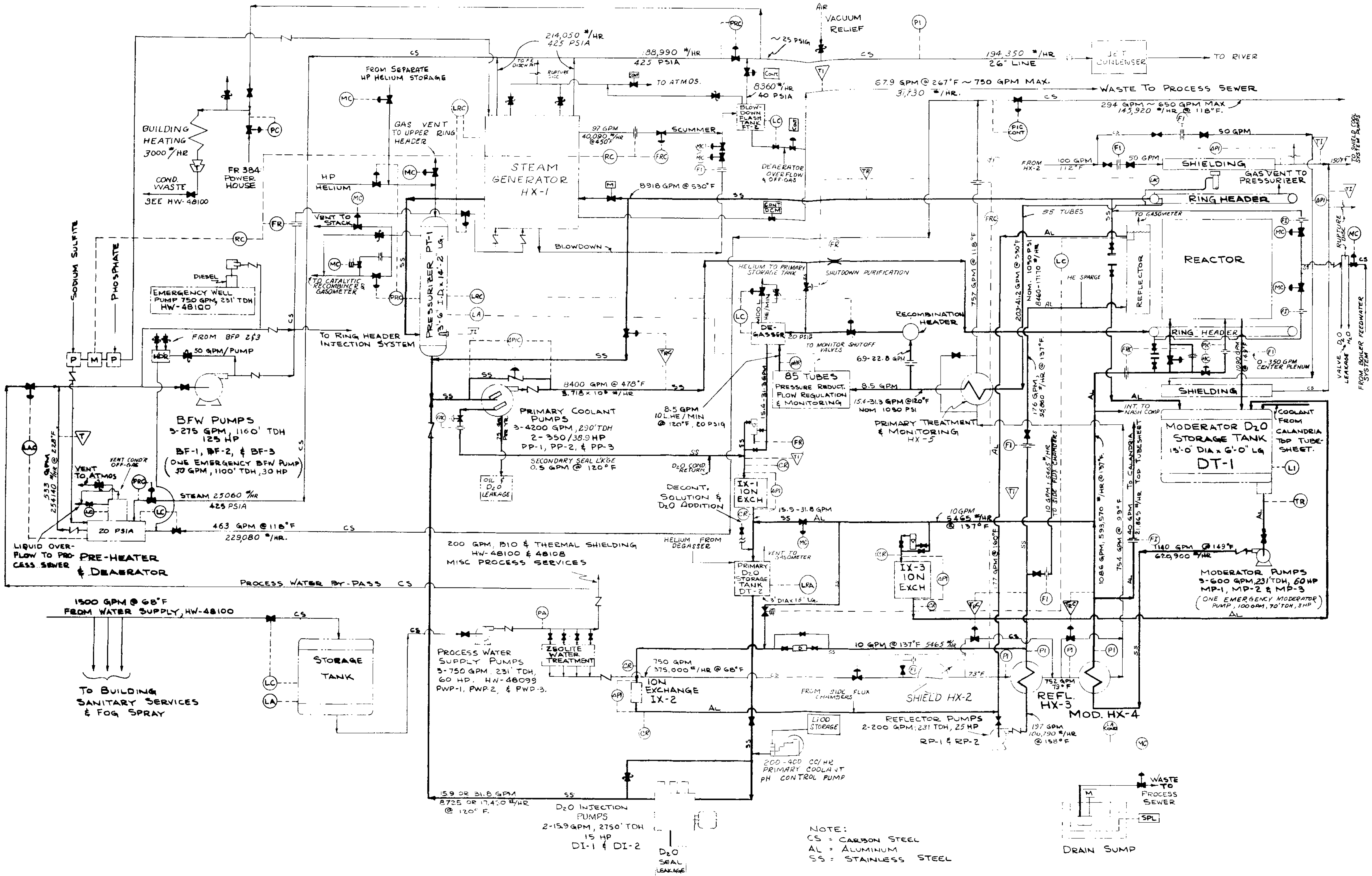

FIGURE 9

Coolant System Flow Diagram
(SK-1-6375)

the temperature of the primary coolant on the tube side from 530 F to 478 F or less. A temperature-controlled bypass around the heat exchanger maintains the reactor inlet temperature of the primary coolant at 478 F.

On the steam side heat is removed by vaporizing 200,000 lb/hr of water into 425 psia saturated steam. The steam is throttled to 40 psia and piped to a barometric condenser. Blowdown of 50 to 100 gpm serves to purge silica and accumulated solids.

Cooling water for the barometric condenser is pumped from the Columbia River at a rate of 6,000 gpm. The effluent from the condenser is returned to the river through an outfall structure.

b. Emergency Coolant System

The emergency light water injection system will provide coolant to the reactor on loss of primary coolant. Two separate systems are provided for light water injection. One system is supplied by the boiler feed pumps and can deliver 700 gpm of light water at 400 psig. The other system is supplied by either the process water pumps or the diesel driven well pump and can supply 750 gpm of light water at 100 psig. Both systems can inject water into the top and bottom ring headers. The points of injection into the ring headers for the two systems are about diametrically opposite. Light water injection is initiated automatically by a low-low liquid level trip on the pressurizer.

A manually started emergency pump, 50 gpm at 475 psig, can be used to supply water to the steam generator through a separate line connected to the top of the steam generator in the event of a boiler feed line rupture. This water is distributed by the demister to cascade over the tube bundle and cool the primary coolant by evaporation after reactor shutdown.

3. Control and Safety Systems

The control and safety systems of the PRTR consist of (1) the primary control system, which provides for automatic operation of the reactor; (2) a manually adjustable shim control system; and (3) the emergency safety system.

a. Primary Control System

The primary control system of the reactor is based on the principle of variation of reactivity of the reactor by variation of the level of moderator in the calandria. Typical relationships of reactivity to moderator level are shown in Figure 10; curves are shown in this figure for several vertical positions of fuel elements in the core (resulting in changes in bottom reflector thickness).

The moderator level in the calandria is maintained and varied by a helium gas pressure balance system. See Figure 11 for the primary control flow sheet. The moderator in the reactor is continuously recirculated. Heavy water is withdrawn from the storage tank below the reactor, cooled in a heat exchanger, and fed into the calandria through two distribution plenums. The heavy water returns to the storage tank, when the moderator level is within the "normal" operating range, both through a series of discharge ports in the upper part of the calandria and by flowing over the annular weir at the base of the calandria. With the moderator level below the "normal" operating range, the total outflow of moderator is over the weir.

Moderator level is controlled by applying a differential helium gas pressure across the surface of the moderator in the calandria and the surface of the moderator in the annular weir at the base of the calandria. When a pressure differential is established, flow over the weir is suppressed and the moderator seeks a level at which the liquid head exactly balances the applied pressure differential. If the pressure differential is then adjusted to a new value, the moderator level changes correspondingly until equilibrium is again established. In similar manner, the calandria may be drained rapidly by quickly equalizing gas pressures; this feature forms the basis for the safety system.

The pressure differential across the calandria is established and maintained by a helium compressor and associated control valves connected between the outlet and inlet of the compressor. The positive displacement

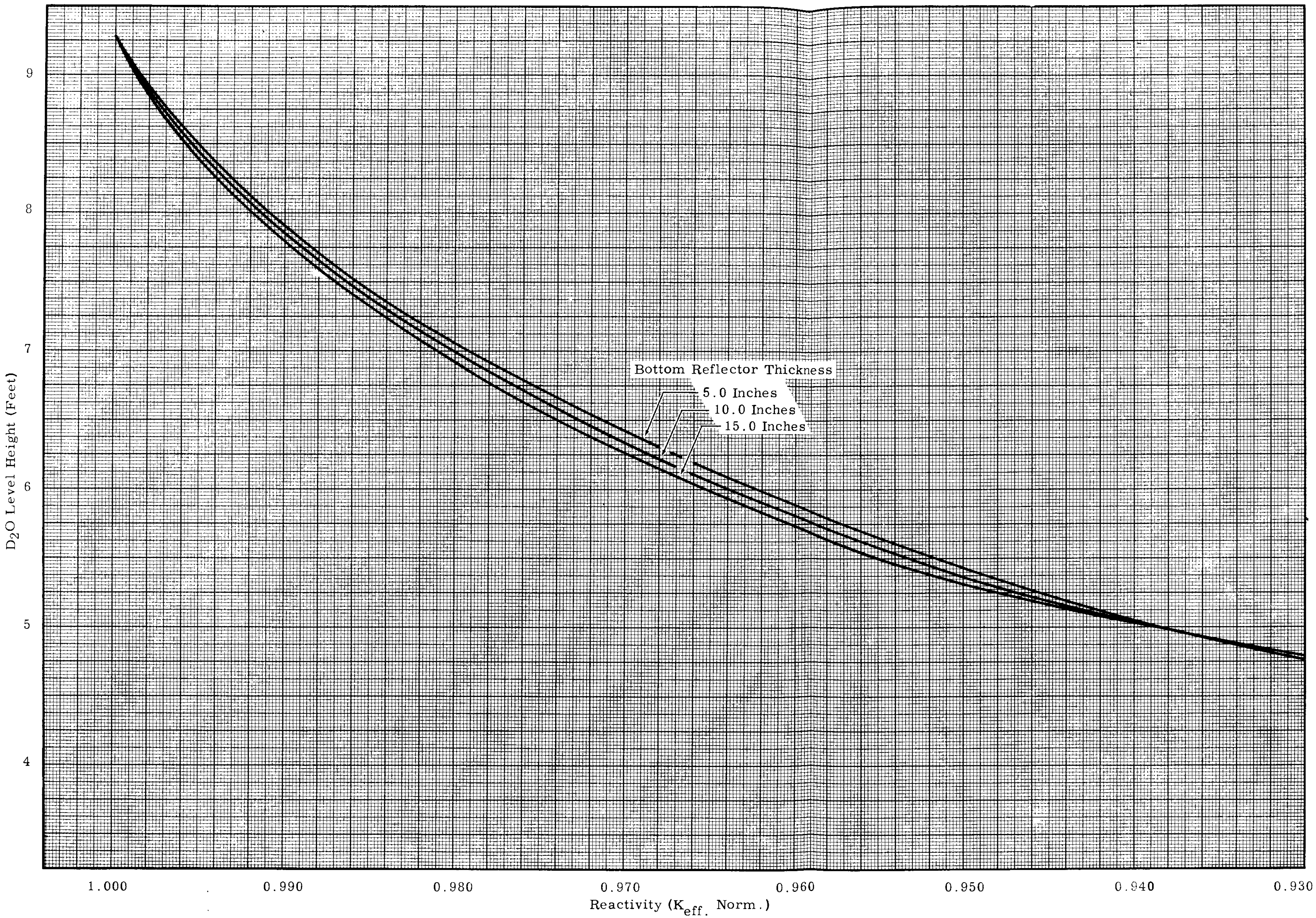

FIGURE 10

Reactivity vs. Moderator Level

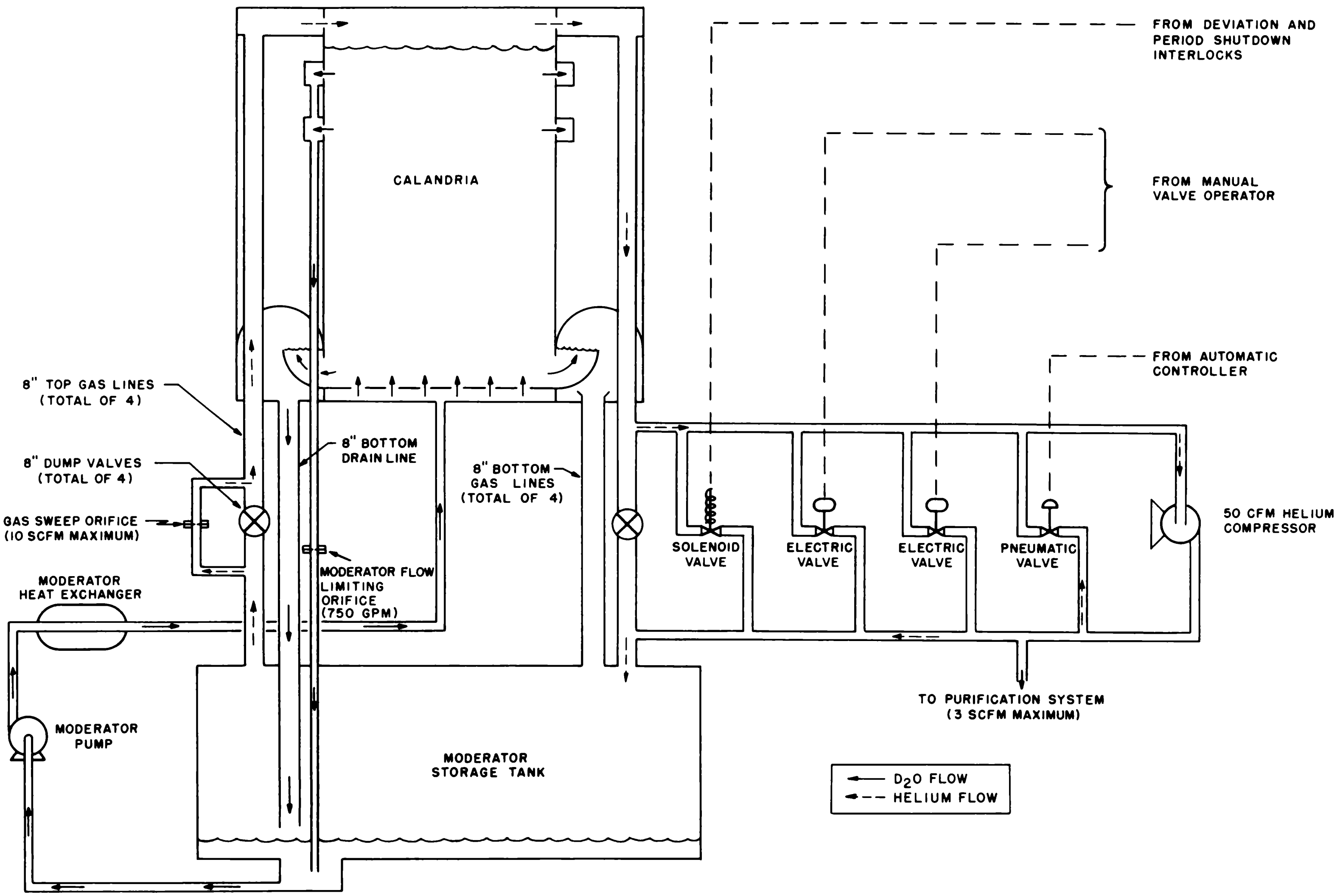

FIGURE 11

Primary Control System Flow Diagram

rotary type compressor has a capacity of 50 scfm. A small flow, normally 10 scfm bleeds into the top gas plenum to supply a gas sweep through the upper part of the calandria. Another small flow, normally 3 scfm, is purged to the helium purification system. The equilibrium pressure in the storage tank is attained when the remainder of the 50 scfm supplied by the compressor is returned by the control valves to the compressor inlet. Thus the differential pressure applied across the calandria is determined by the setting of the control valves, and may be varied by changing this setting. The control valves are sized to maintain the moderator level within ± 0.05 inch of the control point over a range of moderator levels (measured from the bottom of the calandria) from 36 inches (91 cm) to 111 inches (282 cm). This range will provide control action extending well below the minimum level, ~ 53 inches (~ 135 cm), at which the reactor could credibly reach criticality.

The maximum rates of increase of reactivity through increases in the moderator level are limited by the capacity of the helium compressor. For negative level changes, short of scram, the limiting factor is control valve capacity.

Since the rates of level change are limited by gas system capacities, they are dependent on the pressure difference and thus on the moderator level. In Figure 12 are shown the relationships between limiting rates of level change and moderator level; the corresponding rates of reactivity change are shown in Figure 13.

The features of reactivity control through moderator level adjustment make the PRTR particularly amenable to automatic control. The controller, shown in block diagram on Figure 14 adjusts the moderator level control valve settings to maintain desired reactor neutron flux levels and/or periods.

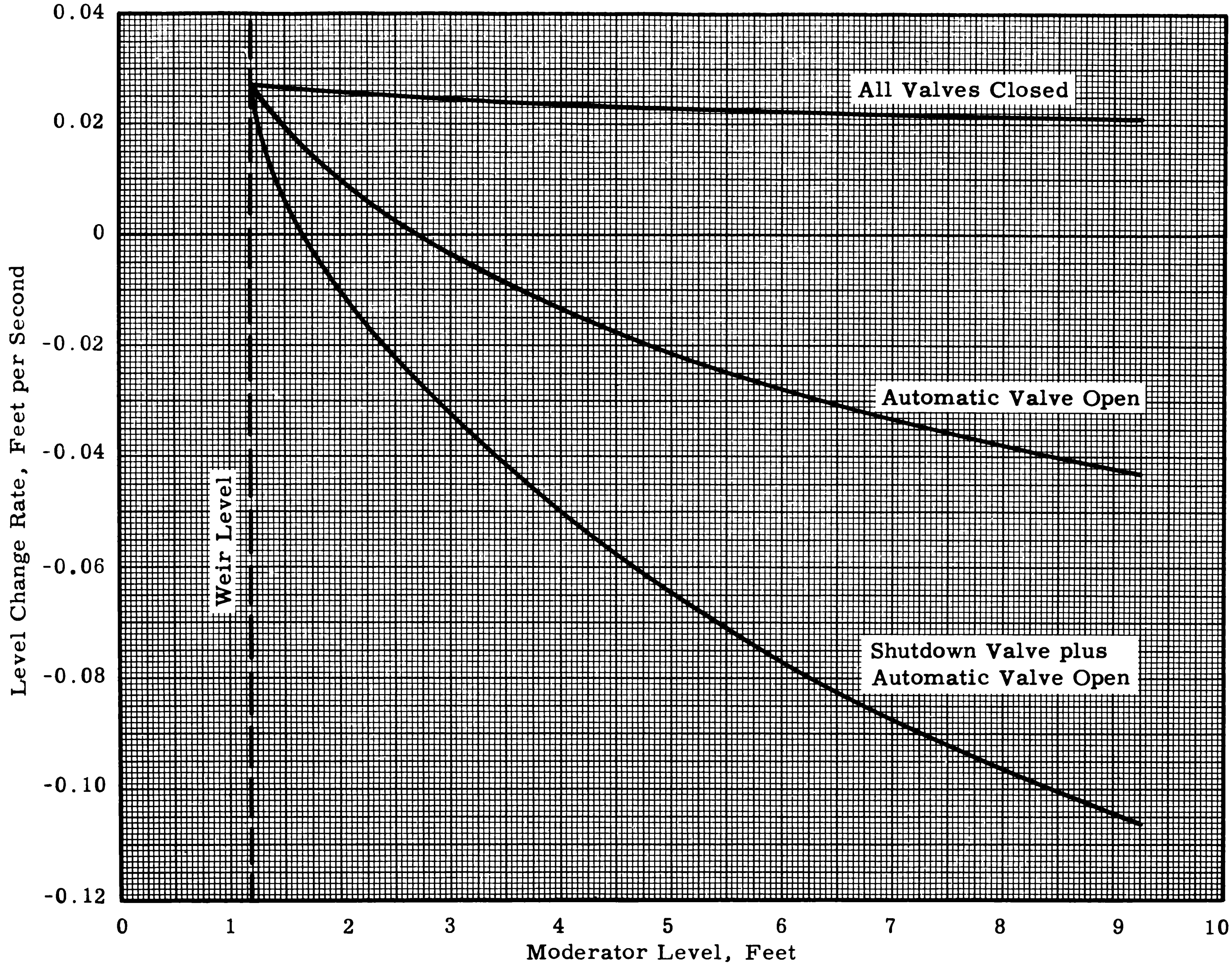

FIGURE 12

Effect of Moderator Level on Limiting Rate of Moderator Level Change

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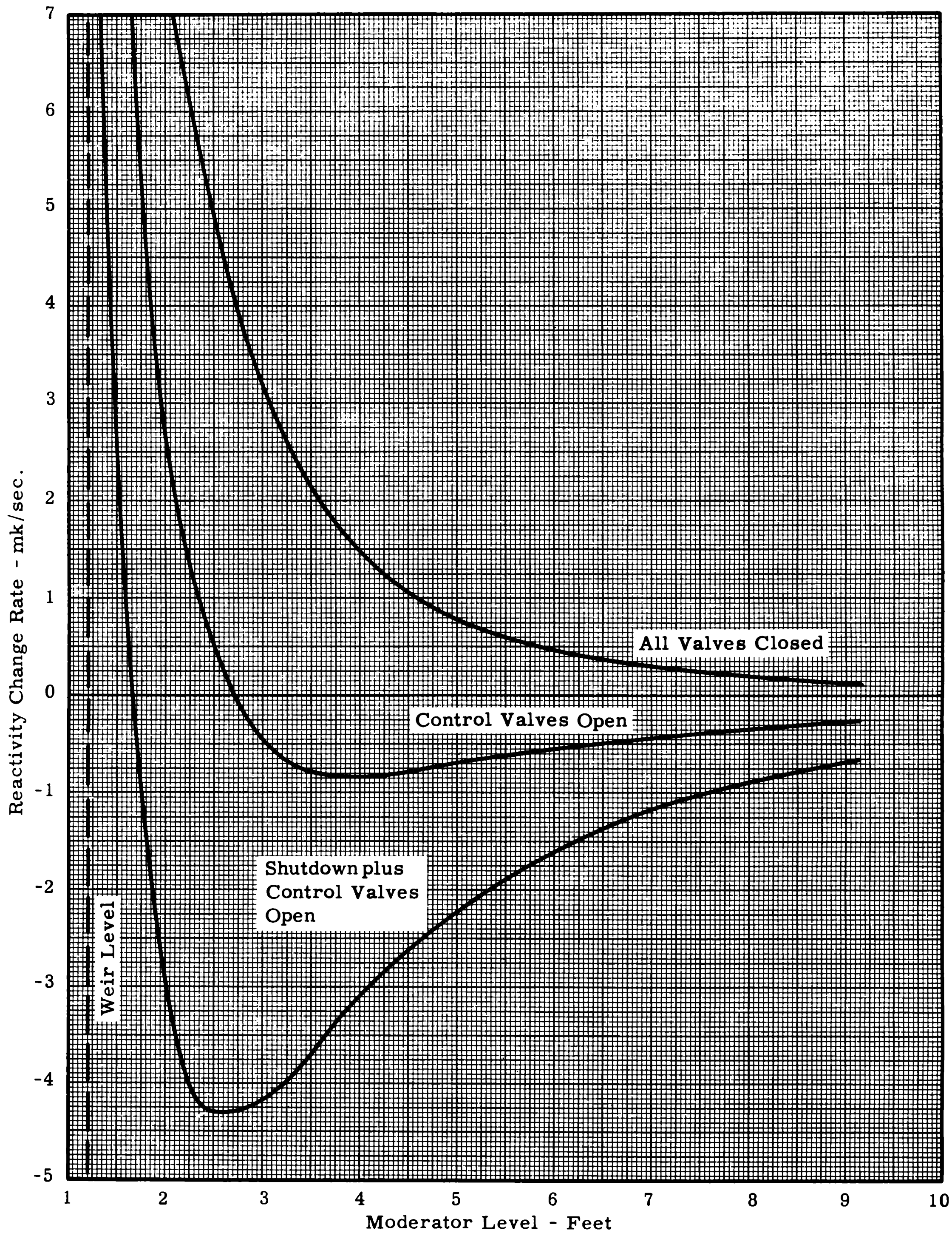

FIGURE 13

Effect of Moderator Level on Rate of Reactivity Change

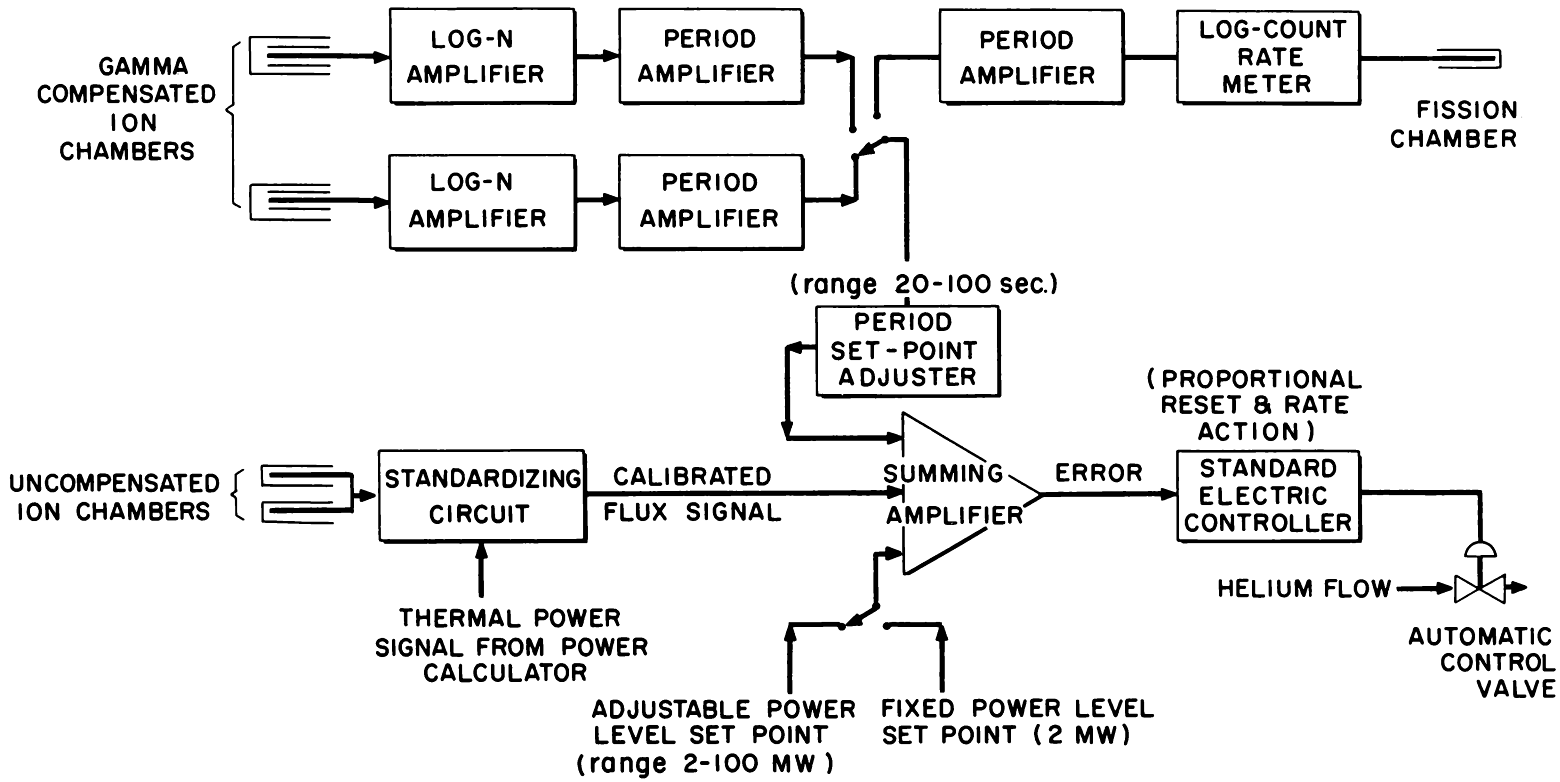

FIGURE 14

Functional Diagram of Automatic Controller

Three input signals which are supplied to the controller from the neutron flux monitoring system are:

(1) a signal proportional to reactor thermal power (neutron flux) level from the high level channels;

(2) a period signal from the low-level logarithmic channel period amplifiers; and

(3) a period signal from the startup (count rate) channel period amplifier.

The signals are so selected by the controller as to provide period control at reactor powers below the high level range, and power level control, with short-period override, within the high level range.

A control programmer is capable of directing control of the reactor to provide, as a minimum, the following actions:

(1) Raise the reactor power, on startup, from initial shutdown levels to a nominal value of power level ($\sim$ 2 MW), on period control, with the period not to be shorter than a preset "demand" value.

(2) Hold the reactor at constant power level at the nominal value until further command is received from a human operator.

(3) Upon command, increase reactor power at a preset rate to normal operating level of 70 MW, or to such other "demand" level as may be set.

(4) Maintain reactor power smoothly and evenly at the desired level.

(5) Shut down the reactor by opening the automatic control valve and the shutdown valve.

A comprehensive study of the dynamic operating characteristics of the reactor on either manual or automatic control was made. This study took account of the following variables which affect control.

(1) Plutonium concentration of the fuel ranging from none to 50 per cent of fissions occurring in plutonium.

(2) Fuel element heat transfer time constants ranging from about 3 to 15 seconds.

(3) A fuel temperature coefficient of reactivity of $2.3 \times 10^{-5}\ \Delta k/C$.

(4) Variations in neutron lifetime from 0.65×10^{-3} to 0.83×10^{-3} seconds.

(5) Variations in critical moderator level ranging from 4 feet to full tank.

(6) Variations in proportional band, reset rate, and derivative action of servo control channel.

(7) Variations in the natural frequencies of the gas balance system, reactor neutron kinetics, and controller.

The stability characteristics were shown to be good. Tests performed on the full-scale mockup of the moderator gas balance system demonstrated that moderator level oscillations (hence reactivity) are of negligible amplitude and of too high a frequency to affect the reactor power level. The minimum range of adjustments of reset rate, proportional band, and derivative action is provided in the servo-channel of the controller to accommodate future fuel loadings and potential changes from design parameters. With proper adjustment of the controller servo-channel no unstable reactor and controller interaction can occur.

It is possible at some reactor operating conditions that the controller settings could be adjusted to a combination of values which would lead to unstable reactor and controller interaction. However, these would pose no danger to the reactor since the amplitude of such oscillations would be limited by the physical limitations of the control system which is able to insert and remove reactivity only at the relatively low rates previously discussed. In addition, such oscillations would cause both period and high power level safety system trips as well as a power deviation trip in the controller itself.

In addition to the above actions the controller operates a valve, designated the shutdown valve, to assist control valve action. This valve is a quick-opening type, with capacity equal to the automatic control valve capacity. The controller opens this valve whenever the reactor power level exceeds the demand level, or is significantly less than the demand level, or the period is shorter than demand value, by preset amounts. Opening of this valve accentuates control valve action, producing a rapid decrease in reactivity as shown in Figure 13.

b. Shim Controls

The shim control system with a total strength of 115 mk, is provided for gross adjustments of reactivity. Primary purposes of the shim controls are: maintenance of the moderator level within the normal operating range; compensation for fuel burnout; provision for xenon override; and flattening of neutron flux, or if desired for experimental purposes, depression of the flux in portions of the reactor. Operation of the shim units is entirely manual, by means of switches located in the control room. The shim controls are not intended as a safety device.

Eighteen shim control units, as shown in Figure 15, are provided in the reactor; locations of these units are shown in Figure 3, page 11. In each shim unit are three Inconel "half rods", 36 inches long. Each rod is raised and lowered on its own lead screw. The shim rods and lead screws are held in position by an extruded aluminum web which extends through the calandria. The lower end of each lead screw runs in a ceramic bushing and the upper end in two open type ball bearings. Thrust is carried by the upper ball bearing. The three lead screws in each unit are driven by two motor drive assemblies. Two lead screws are driven by one motor and travel at two different speeds. The third lead screw is driven by the other motor. Shim rod travel for the two coupled rods is 36 ipm and 24 ipm. The effect of these coupled rods is that of a "telescoping" rod. The third rod will travel at 36 ipm. Position indications for the two fast speed rods are transmitted to the control room by electrical position transmitters.

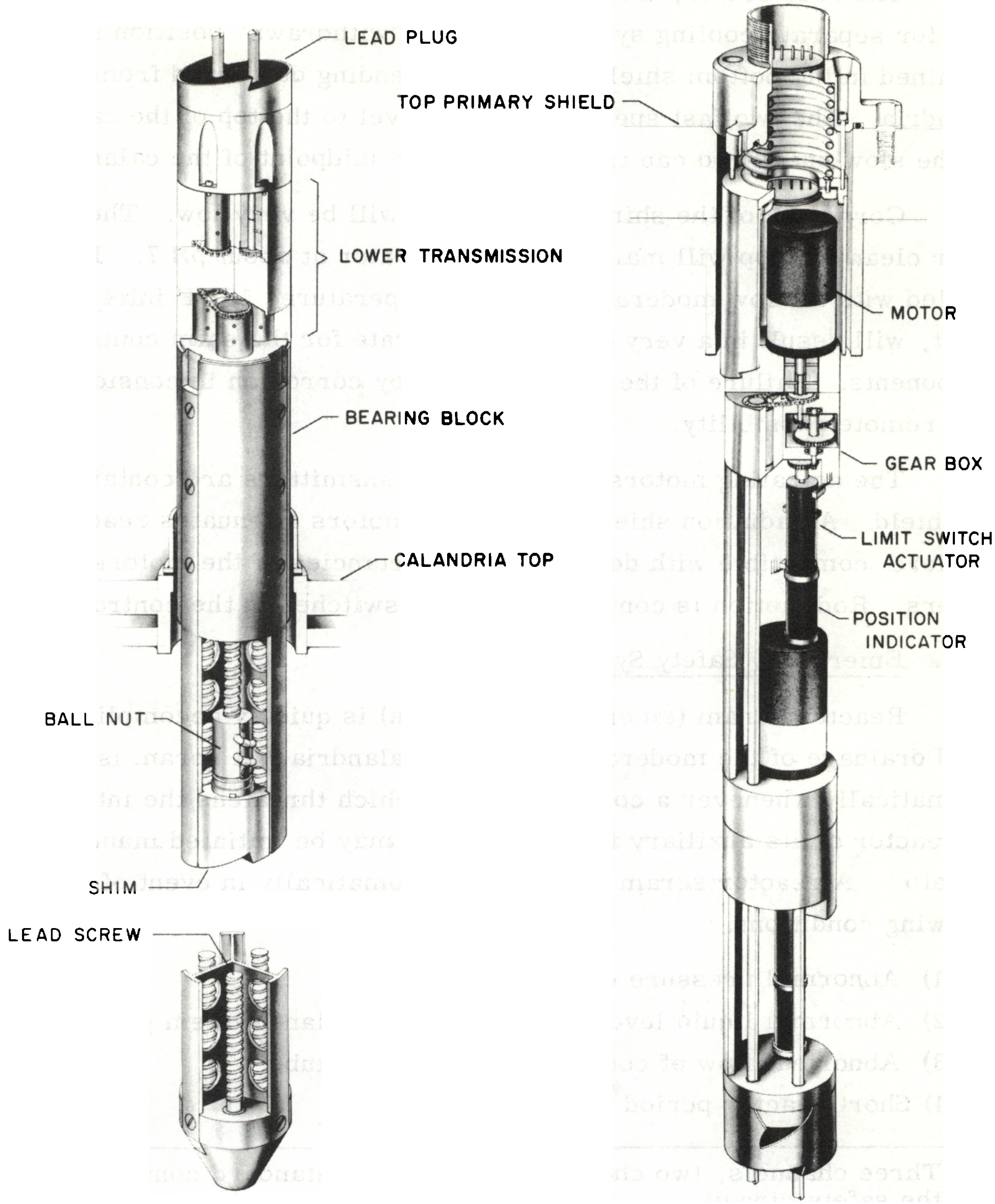

FIGURE 15

Shim Control Unit

The rods are exposed directly to the moderator, eliminating the need for separate cooling systems. In the "withdrawn" position they are contained in the bottom shield, in tubes extending downward from the calandria. The two fast speed rods can travel to the top of the calandria, but the slow speed rod can travel only to the midpoint of the calandria.

Corrosion of the shim control units will be very low. The heavy water clean up loop will maintain the moderator at about pH 7. This pH coupled with the low moderator system temperature, 137 F inlet and 149 F outlet, will result in a very low corrosion rate for the shim control components. Failure of the shim controls by corrosion is considered a very remote possibility.

The actuating motors and position transmitters are contained in the top shield. A radiation shield beneath the motors attenuates reactor radiation to a level compatible with desired life expectancies of the motors and transmitters. Rod motion is controlled through switches in the control room.

c. Emergency Safety System

Reactor scram (emergency shutdown) is quickly accomplished by a rapid drainage of the moderator from the calandria. A scram is initiated automatically whenever a condition exists which threatens the integrity of the reactor or its auxiliary facilities, or it may be initiated manually by the operator. A reactor scram is initiated automatically in event of any of the following conditions:

(1) Abnormal pressure of primary coolant*
(2) Abnormal liquid level in the primary coolant system pressurizer*
(3) Abnormal flow of coolant in any process tube**
(4) Short reactor period***

* Three channels, two channels indicating offstandard condition will trip the safety circuit.

** Two channels, either channel indicating offstandard condition will trip the safety circuit.

*** Two channels, either channel indicating offstandard condition will trip the safety circuit. In addition during reactor startup a sensitive startup channel will be connected in the safety circuit.

(5) Abnormally high power level*

(6) Low liquid level in steam generator*

(7) High pressure in steam generator*

(8) Low pressure of instrument air*

(9) High activity in building ventilation air***

(10) High activity in secondary coolant effluent*

(11) Electrical power failure**

(12) Earthquake of sufficient force*

Four 8-inch gas pressure equalizing lines connect the top gas plenum of the calandria with the moderator storage tank gas volume. These lines are normally closed by quick opening dump valves. The dump valves are held closed by solenoids during reactor operation. Upon receipt of a scram signal, current to the solenoids is interrupted. Powerful springs then open the valves rapidly. At the same time the compressor in the primary control system is stopped and the control valves open fully. Gas pressures within the system equalizes rapidly and the moderator drains from the calandria by gravity.

In tests, the dump valves opened fully in ~ 0.05 second and had low resistance to gas flow when open. When the valves open, gas pressures in the calandria are equalized quickly. The moderator within the calandria immediately begins to drop into the dump chamber under gravity flow. A small portion of the moderator flow is returned to the storage tank by the 8-inch drain line, but the larger part is retained in the dump chamber. The four bottom gas lines, with elevated entrances, permit the gas displaced from the dump chamber to flow to the storage tank.

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* Three channels, two channels indicating offstandard condition will trip the safety circuit.

** Two channels, either channel indicating offstandard condition will trip the safety circuit.

*** Two channels, both channels must indicate offstandard condition to trip the safety circuit.

Within 0.85 second after receipt of a scram signal, the moderator level has fallen 2 feet, decreasing reactivity by a minimum of 18 mk (with a moderator level initially at its maximum in the calandria). At this time the dump chamber has filled with moderator, and further flow is limited by the capacity of the five 8-inch lines leading to the storage tank. Under these conditions the moderator level continues to drop at about 1.0 ft/sec, the rate decreasing as the calandria empties. Within 6 seconds the moderator surface is at the level of the outflow weir. The corresponding decrease in reactivity is approximately 1400 mk.

Figures 16 and 17 show the behavior of moderator level and of reactivity during a scram. The curves are shown for the conditions with the moderator initially at its maximum level. With the moderator initially at a lower level the rate of level change would be less; however, because of the greater reactivity change for a given moderator level change at lower initial operating levels the rate of reactivity decrease would be as fast or faster than those shown.

The safety system is so designed that the failure of two dump valves to open will not materially decrease dump rates. Failure of a third valve would decrease dump rates somewhat, but still would provide rapid shutdown for all conceivable incidents.

d. Instrumentation

Seven channels of reactor flux instrumentation are used. These fall into the following four categories:

(1) Startup Channel

The startup channel consists of a fission chamber with a remotely controlled positioning device, linear amplifier, log count rate meter, recorder and period amplifier. The period amplifier is connected in the reactor shutdown safety circuit and will "scram" the reactor on too short a period. A period signal is also supplied to the automatic controller.

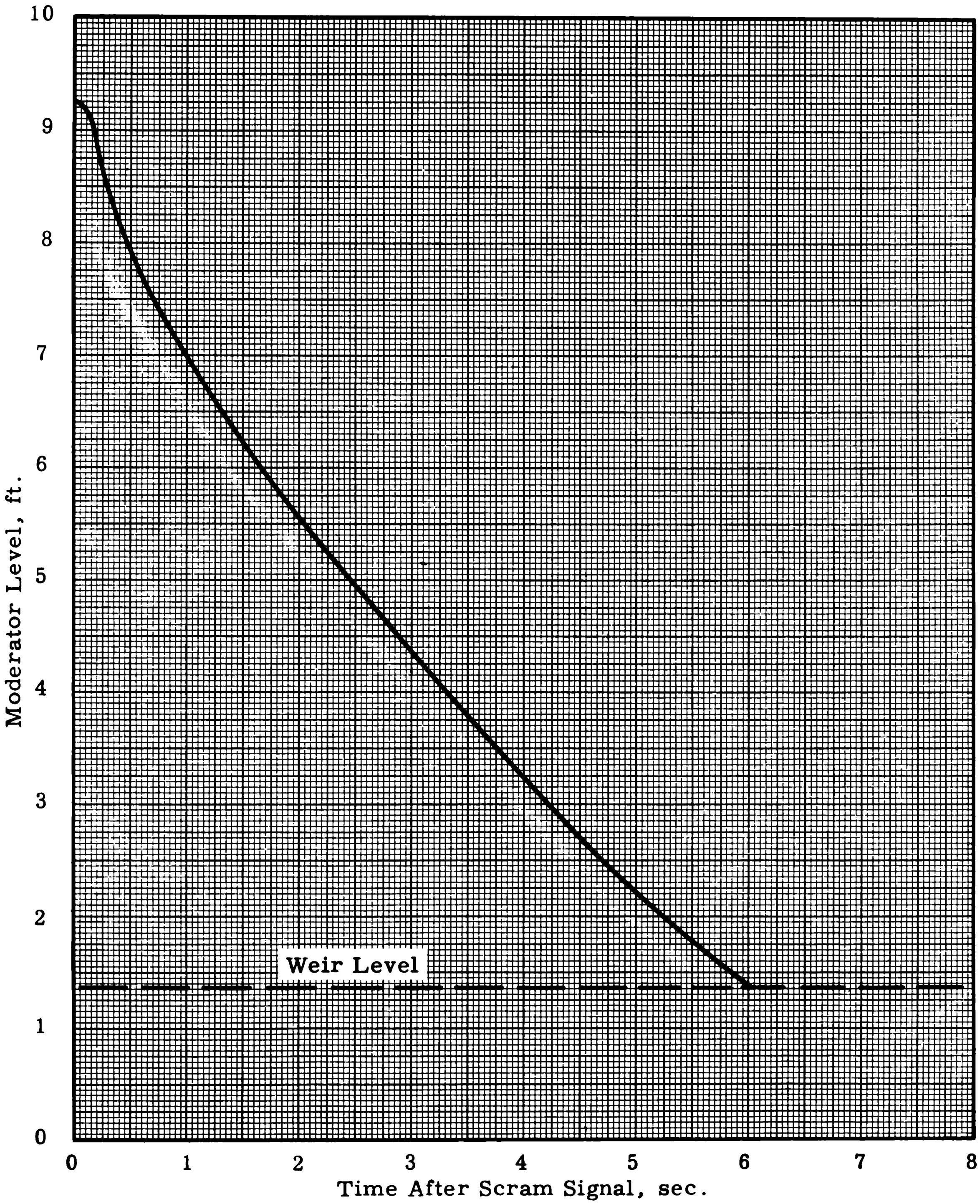

FIGURE 16

Moderator Level Decrease During Scram

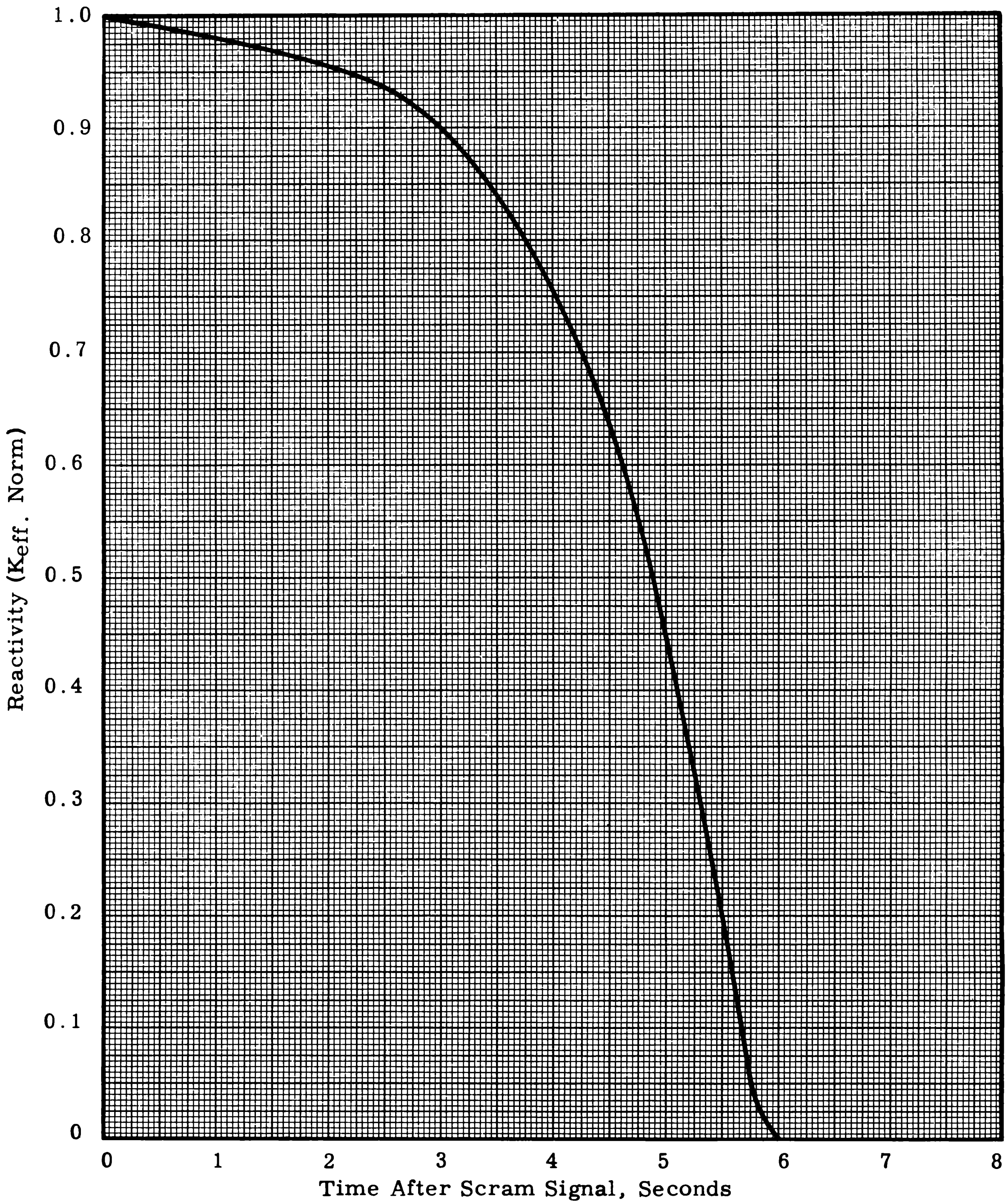

FIGURE 17

Reactivity Decrease During Scram

(2) Logarithmic Channels

The two logarithmic channels consist of two compensated ion chambers, log amplifiers, period amplifiers and recorders. The period amplifiers are connected in the reactor safety circuit and also furnish signals to the automatic controller.

(3) High Level Channels

The three high level channels consist of three uncompensated ion chambers, linear amplifiers and recorders. The three channels are connected in the reactor safety circuit. An interconnection circuit with the logarithmic channels protects against instrument failure by causing a "scram" if less than two of the three high level channels indicate "on-scale" signals and the log channels are "upscale" past a preset trip point.

(4) Reactor Control Channel

The reactor control channel consists of two uncompensated ion chambers, which provide a neutron flux level signal to the automatic controller. Two galvanometers are also provided in this circuit to indicate flux levels; these are utilized when the reactor is controlled manually. These chambers are used for reactor operation only and are not included in the safety circuit. Safety backup is provided by the three high level channels.

Additional instrumentation for operation of the reactor includes:

(5) Flow Monitor

The flow monitor measures the inlet flow through each reactor process tube and actuates the reactor shutdown circuit if out-of-limit high or low flows occur.

The system consists of:

(a) 85 venturis, one on each process tube;
(b) flow meters, located in an instrument cell and equipped with high and low flow trips and readout circuitry;

(c) a recorder and type-out system located in the control room.

(6) Temperature Monitor

The temperature monitor measures the temperature of the outlet flow of each tube and actuates alarms if an over-temperature exists. A surface type resistance temperature detector is mounted on the outlet of each process tube; bridge and read-out circuitry is located in the control room. This system utilizes the same type-out equipment provided for the flow monitor system.

(7) Water Activity Monitor

The water activity monitor consists of two main systems:

(a) The primary activity monitor, which monitors the gamma activity of small sample flows from each process tube and a bulk sample flow by means of scintillation detectors indicating any tube in which a rupture has occurred; and

(b) The secondary coolant activity monitor, which monitors the secondary coolant for gamma radiation to detect traces of primary coolant that may result from leakage in the heat exchangers.

(8) Power Calculator System

The power calculator system monitors the power level of the reactor as determined by measurement of the heat removed by the primary and moderator reactor coolants. The system consists of two channels. One channel measures the power generated in the primary coolant loop; the second channel measures the power generated in the moderator cooling loop. Each channel consists of a temperature difference bridge and flow measuring instrumentation. The output is a signal representative of power level.

(9) Reactor Thermocouple System

Mineral insulated, swaged type thermocouples are provided for measuring temperatures of reactor components as follows:

(a) Moderator temperature within the calandria
(b) Thermal shield temperatures
(c) Top and bottom shield temperatures
(d) Main side shield temperatures.

Temperatures are recorded and indicated in the control room.

(10) Steam Generation System

Standard power house type instrumentation is used to control the steam generation system. All signals are transmitted to the control room. The reactor safety circuit is actuated by a high pressure or low water level in the steam generator.

(11) Building Radiation Monitor System

Beta and gamma sensitive ion chambers are placed in various locations inside the reactor building. The radiation level is indicated and recorded by instrumentation on a centrally located panel. Air samples from enclosures containing primary coolant and moderator piping are monitored with beta sensitive detectors for traces of tritium.

(12) Reactor Safety Circuit

The reactor safety circuit causes a reactor shutdown by opening valves which lower the moderator level. This shutdown will occur when certain conditions exist that would result in marginal reactor safety.

(13) Other Systems

Additional instrumentation is provided to monitor flows, temperatures, pressures, etc., of the gas system, coolant purification systems and building services, such as ventilation. Instruments are of standard type with all indicating and recording in the control room or in the instrument cell.

4. Helium System

The reactor helium system performs three major functions. These are:

(a) To pressurize the primary coolant. The pressurization system is composed of a high pressure storage tank, an intermediate pressure storage tank, a high pressure booster compressor, an intermediate pressure compressor, and appropriate valves, coolers, water traps, etc.

(b) To provide a helium blanket for the moderator and working fluid for the gas balance system. This system was described in more detail in the previous section on Control and Safety Systems.

(c) To provide an inert and unactivated gas blanket around the reactor core and to provide a working fluid for the water leak detection system.

5. Fuel Handling System

Charging and discharging of the fuel elements is done on an individual basis through a hole in the reactor hall floor, while the reactor is shut down. This hole is normally closed with a shielding plug. It is placed eccentrically in a disk which is in turn located eccentrically in a larger disk. By rotating the two disks, the port in the floor can be positioned over any process tube, and discharging of fuel elements can be accomplished with the removal of only a minimum of shielding. The shielding floor remains intact during discharge operations.

The charge-discharge operations is based on the use of a large self-propelled fueling vehicle. This vehicle carries an unshielded tube for charging operations with unirradiated fuel elements and a heavily shielded cask for both discharging and recharging of irradiated fuel elements. The arrangement of the fueling vehicle is shown in Figure 18. This device is about 25 feet high and weighs 50 tons. Much of the weight is concentrated

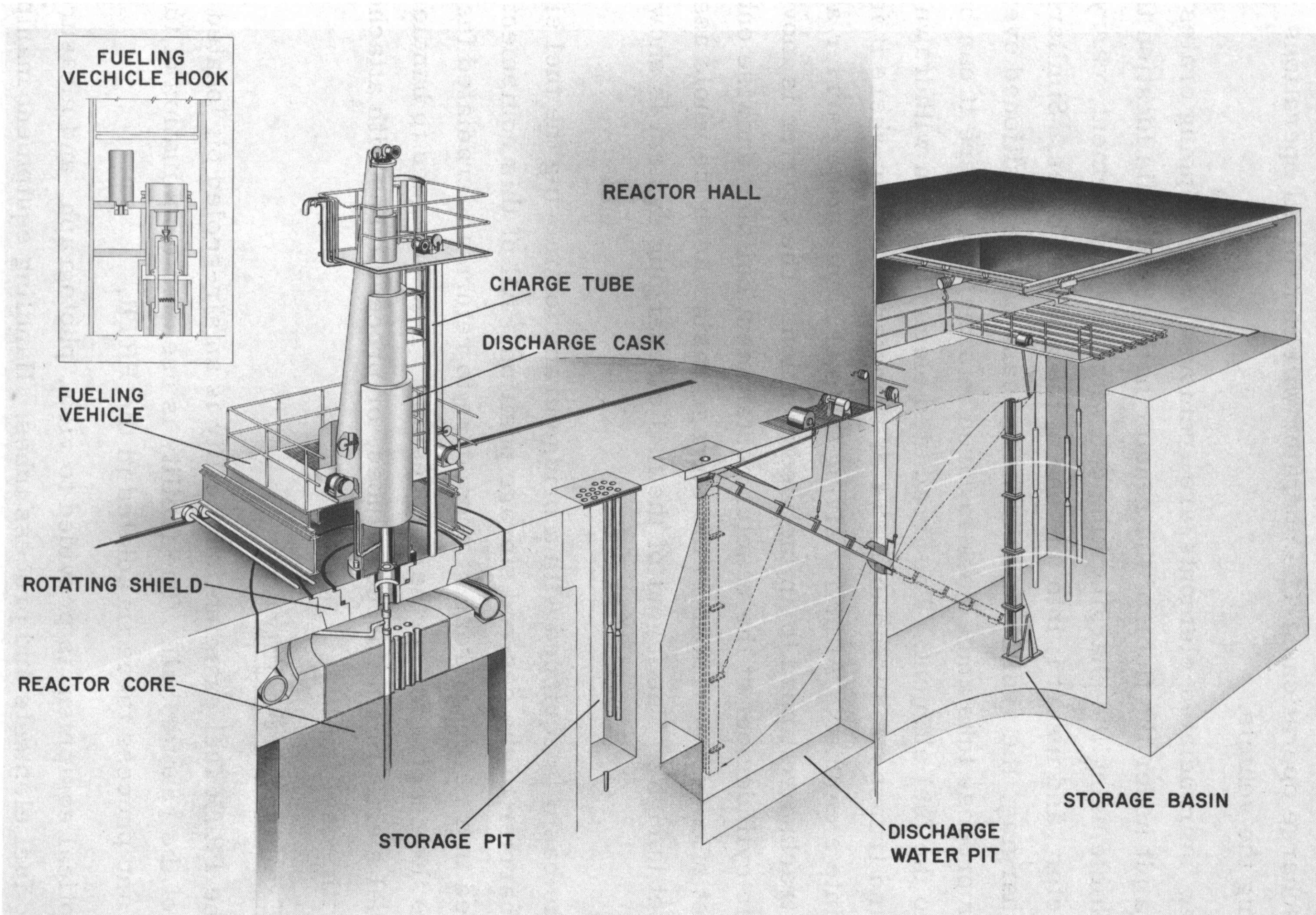

FIGURE 18
Fuel Handling System

in the 12-1/2 inches of lead required as shielding to protect personnel during discharge operations. The operator will control all operations while riding the vehicle.

New unirradiated elements are removed from shipping crates and stored in a pit under the floor of the reactor hall. From this location the fueling vehicle using the charging tube can withdraw any element, carry it to the reactor, and insert it into any one of the process tubes. Similarly, when discharging, the vehicle and discharge cask can be positioned over the proper process tube and a hook lowered into the tube where it can be attached to the fuel element. The irradiated element is then withdrawn into the cask and transported to an underwater transfer conveyor where it is released into a carrier. After receiving the fuel element the carrier and conveyor tracks are tipped to an inclined position. The carrier is moved by a hydraulic cylinder out of the reactor containment building endwise onto a similar set of conveyor tracks in the storage basin where the whole assembly is again set upright. The layout of the fuel discharging system is shown in Figure 18.

The basin is equipped with a bridge crane for moving the fuel elements from the carrier to the desired storage position. All of this equipment can be operated in a reverse order of procedure to return an irradiated fuel element to the reactor for further processing. It is designed to handle the various fuel elements, the process tubes, or process tubes with attached nozzles with equal facility.

6. Fuel Examination Facility

The PRTR fuel examination facility is an air-cooled pit located in the floor of the reactor hall. The facility is used for examination of fuel elements and process tubes irradiated in the PRTR.

Optical equipment is provided to view, photograph, and measure irradiated fuel elements and process tubes. Handling equipment manipulates the element past the viewers. Elements are cooled to prevent overheating. No sectioning equipment is provided.

The infinite multiplication constant for the reactor, and hence the number of spike (Pu-Al) fuel elements, is determined by the excess reactivity required. The values given below are discussed later.

	Excess Reactivity Required, mk
Operating:	
Equilibrium Xe and Sm	45
Fuel Temperature	32
Moderator Temperature	5
Shutdown: Total Xenon and Samarium at 2 hours	90

Thus the total excess reactivity required is 127 mk to which is added an arbitrary 10 mk to allow for fuel depletion between charge-discharge operations. The required infinite multiplication constant for the reactor is thus given by:

$$K_\infty = \frac{1 + k_{ex}}{\text{(Non leakage probability)}} = \frac{1.137}{0.900} = 1.263$$

The relative number of spike elements may be estimated by assuming a neutron flux distribution and volume and flux weighting the respective k_∞ to match the required reactor k_∞. However, greater accuracy may be obtained by performing multi-group calculations on the reactor core.[(3)] Results of three-group, three region critical loading calculations of this type are shown in Figure 19.

2. Reactivity Control

a. Level Control

The level control strength varies strongly with the moderator level height. The vertical buckling is given by

$$B^2 = \left[\frac{\pi}{H}\right]^2$$

(3) Peterson, R. E. and J. J. Regimbal. Reactivity Worth and Power of Spike Loading Configurations in PRTR, HW-58016. October 31, 1958.

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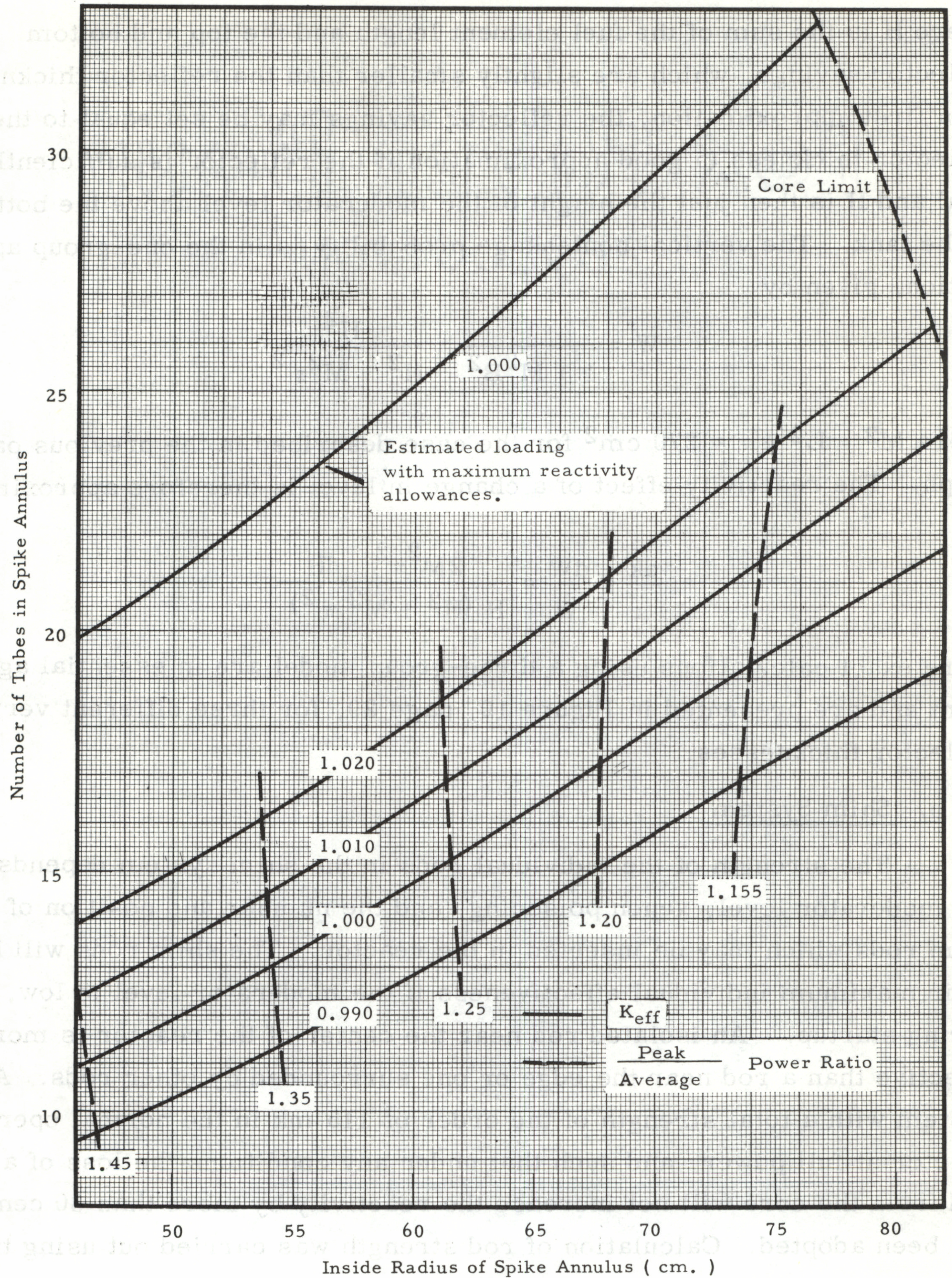

FIGURE 19

Spike Enrichment Loading Requirements

where H is the sum of the fuel element length and the top and bottom reflector savings, which are slightly smaller than the reflector thickness. As a first approximation, the reflector savings may be set equal to the reflector thickness (a good approximation if the reflector is sufficiently thin) and H is then just the height of the moderator level above the bottom of the tank. The vertical nonleakage probability is in the one-group approximation, given by

$$W = \frac{1}{1 + M^2B^2} = \frac{H^2}{H^2 + M^2\pi^2}$$

where $M^2 = L^2 + \tau \approx 270 \text{ cm}^2$ for the case described in the previous paragraph. The reactivity effect of a change in level is therefore approximately

$$\frac{dk}{k} = \frac{dW}{W} = \frac{2M^2\pi^2 \quad dH}{H\,(H^2 + M^2\pi^2)}$$

More exact calculations using a three-group model are in essential agreement and are presented in Figure 10, page 29, for three different vertical reflector thicknesses.

b. Shim System

The strength of the individual rods in the shim system depends upon the moderator level, xenon poisoning, and the number and position of other shim rods which may be inserted in the reactor. The shim rods will have their maximum individual effectiveness if the moderator level is low, as during startup. An isolated rod near the center of the reactor is more effective than a rod near the edge or one surrounded by other rods. A system with a total strength of the order of 115 mk in the normal operating reactor at full power, and such that under any conditions the loss of a rod from the core will not increase the reactivity by more than 80 cents has been adopted. Calculation of rod strength was carried out using two group theory with an independent check by means of the P-3 approximation. The basic design of the shim system affords considerable flexibility to meet the requirements of particular reactor conditions.

c. Xenon Override

The increase in xenon poisoning after shutdown and the subsequent decrease in poisoning after the reactor is again started up at various times is shown in Figure 20 for full power operation. An allowance of 90 mk for total xenon transient poison should permit startup within approximately two hours after shutdown. Unlimited startup could be achieved by allowing of the order of 140 mk for this purpose, but this would render self-sustaining recycle operation essentially impossible and would require undesirably strong individual shim rods. If the reactor is not started up within the 2-hour xenon override period it cannot start up until approximately 26 hours after shutdown.

3. Reactor Kinetics Behavior

a. Neutron Lifetime and Delayed Neutron Fraction

Values in this paragraph are discussed more fully in report HW-48907.[4] A neutron lifetime of 0.82 milliseconds is calculated by conventional methods. Twelve groups of delayed neutrons are employed, corresponding to production in U^{235}, U^{238}, and Pu^{239} fission plus deuterium photoneutrons. The total delayed neutron yield varies from 0.83 per cent of the total if no plutonium is present to 0.541 per cent if 50 per cent of the fissions occur in plutonium. A good value for use in most hazards analyses is 0.72 per cent corresponding to 35 per cent of the fissions occurring in plutonium This means, in other words, that 7.2 mk corresponds to one dollar of excess reactivity, and produces a prompt period. The corresponding conversion to inhours at criticality is 35.6 ih = 1 mk, which is

(4) Houser, D. T. and M. V. Davis . Kinetic Behavior of a Uranium-Plutonium D_2O System, HW-48907. March 11, 1957.

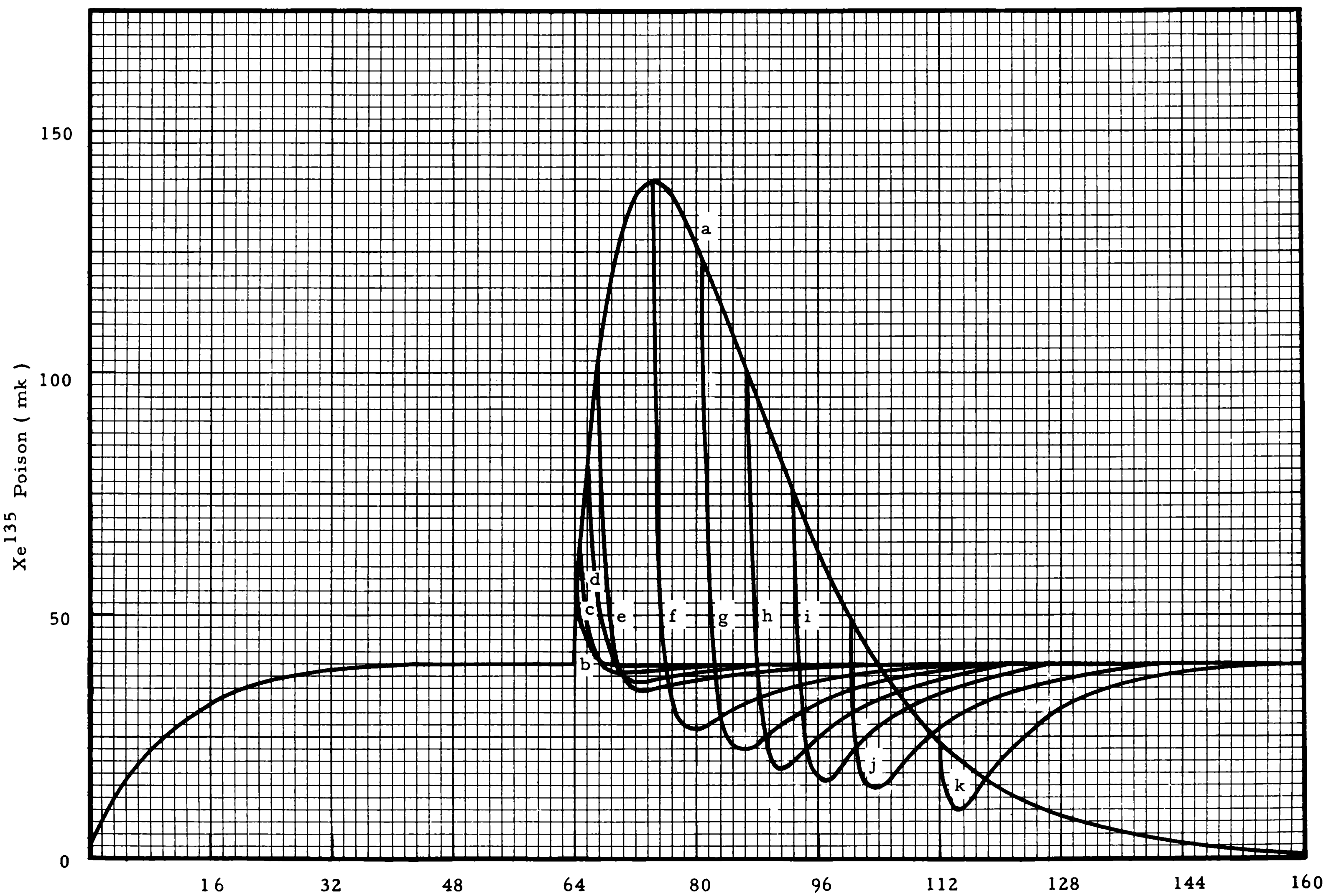

FIGURE 20

Xenon Poison Transients

close to values characteristic of several presently operating thermal reactors. Response to rapid changes in reactivity due to errors or malfunction are discussed in Section VI.

b. Moderator Coefficients

The effects of moderator temperature, voids, and H_2O content are discussed in HW-46679 REV,[5] which presents results of a series of calculations using the P-3 method. The conclusions of this study may be summarized by quoting the appropriate coefficients:

$$\text{Moderator temperature, } \frac{1}{k}\frac{dk}{dT} = -1.06 \times 10^{-4}/C$$

This is entirely a density effect; the contribution of fuel cross sections is not included. The calculated density effect itself is

$$\frac{dk}{k} = -0.323\frac{d\rho}{\rho}$$

where ρ is the density, so that, for example, replacement of three per cent of the moderator by voids reduces reactivity by approximately 10 mk. Loss of coolant produces a more complex effect than does simple reduction of moderator density, since it results in

(1) increases of thermal utilization and fast effect, and

(2) in a reduction of the surface resonance absorption cross section of the fuel and the epithermal non-leakage probability.

It is possible that in some fuel element configurations these reactivity increases would override the effect of reduced moderation and increased leakage, resulting in a small net increase of reactivity on loss of heavy water coolant. For example, although the Mk I UO_2 fuel shows an increase of 12 mk in k_∞ (most of which is in ϵ) on coolant loss, the increased fast leakage results in a maximum net gain of only 4 mk in k_{eff}. Calculation and measurement have shown that the presence of Pu-Al fuel in the reactor further reduces the net gain in k_{eff}. This is due to the much smaller change

(5) Davis, M. V. Reactivity Changes Resulting from Variations of Moderator Quality in the PRTR, HW-46679 REV. undated.

in k_∞ with coolant loss for this fuel. Therefore, the net increase in k_{eff} for a spike loading will be less than 4 mk.

The effect of addition of light water to heavy water moderator or coolant is pronounced. The first one per cent increment of light water costs about 10 mk of reactivity; the second one per cent increment an additional 19 mk. Five per cent light water represents a total of 55 mk reactivity loss. Large leakages of light water into the moderator or coolant would thus shut down the reactor.

c. Fuel Temperature Coefficient

The temperature coefficient due to Doppler broadening of the uranium resonance lines in UO_2 has been measured in the Physical Constants Testing Reactor and found to be

$$\frac{1}{k}\frac{dk}{dT} = \frac{1}{p}\frac{dp}{dT} = (-2.3 \pm 0.2) \times 10^{-5}/C$$

for MK I UO_2 fuel in the temperature range of from 20 C to 365 C.[6] The resulting resonance integral coefficient is

$$\frac{1}{\Sigma_{res}} \quad \frac{d\Sigma_{res}}{dT} = (2.0 \pm 0.2) \times 10^{-4}/C$$

Although the Doppler coefficient of the Pu-Al fuel has not been measured, it is known to be small. Therefore, a partial UO_2 charge, such as results from the employment of Pu-Al fuel material in part of the reactor, produces a smaller effect. There is also evidence that the temperature coefficient of the resonance integral in UO_2 at higher temperatures (200 to 1000 C) is only about half of the value measured in the PCTR over the temperature range indicated above.[7]

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(6) Heineman, R. E. et al. "Experience in the Use of the Physical Constants Testing Reactor", Geneva Conference Paper 1929. June, 1958.

(7) Blomberg, P. and E. Hellstrand, et al. "Measurements and Calculations of the Effective Resonance Integral in Uranium Metal and Oxide", Geneva Conference Paper 150. September, 1958.

For these reasons, the reactor fuel coefficient used in the safety analyses has been reduced from the measured value and is described in Section VI.

C. The Building

The PRTR building arrangement consists of two primary parts (a) the process area comprising the containment vessel and the storage basin, and (b) the service and utilities building. The reactor is located axially in the lower portion of the containment vessel and is surrounded by a thick cylindrical shielding wall which is common to each of the three process cells. A perspective view of the PRTR building is shown in Figure 1 , page 6.

1. The Process Area

a. Description

The containment vessel, which houses all of the process area except the storage basin, is an all welded steel cylinder(ASTM-A-212, Grade B) , 80 feet in diameter, with a hemispherical dome and an ellipsoidal bottom head. Over-all height of the containment vessel is 121 feet 6 inches, extending 75 feet above grade. Gross volume of the containment building is about 500,000 cubic feet with a net free volume of about 400,000 cubic feet. Arrangement of the building is shown in Figures 21, 22, 23, and 24.

The exterior surface of the containment shell above grade is covered with three inches of insulation over which a waterproof membrane is applied. Below grade the exterior surface of the vertical cylinder is protected with 1/4 inch of waterproof membrane and the bottom with 5/32 inch of waterproof membrane. A cathodic protection system is provided for the vessel. The inside surfaces of the steel containment shell which are in contact with the concrete are coated with "Koppers Bitumastic No. 50" and one inch of fiberglas insulation to allow for independent movement of the shell and the concrete.

The reactor hall floor is slightly above grade and is ordinary concrete five feet thick for radiation shielding. The reactor hall houses the decontamination equipment, fuel examination facility, fuel handling cask, fuel storage transfer and handling facilities, and a circular 30-ton overhead crane with a 1 1/2-ton auxiliary hoist.

One personnel air lock, approximatley 10 feet in diameter by 15 feet long, is provided for normal access into the reactor hall from the service area. This air lock is equipped with doors, 4 feet wide by 8 feet high, which are provided with mechanical interlocks and indicating lights to assure that both doors cannot be open at the same time during reactor operation. An emergency air lock, approximately 5 feet in diameter by 7 feet long, is equipped with 3 foot diameter doors for egress under emergency conditions or in the event of failure of the main air lock doors. The doors of this airlock are provided with interlocks to insure that both doors are not opened at the same time. The emergency air lock allows exit from the containment building directly to the outside. An equipment door, 5 feet 6 inches wide by 10 feet high, opens directly to the outside from the reactor hall. Procedures specify that the equipment door may be opened only when the reactor is shut down.

The space below grade is divided into three cells by 5-foot thick ordinary concrete radial shielding walls. The cells are

(1) process equipment cell,

(2) experimental equipment cell, and

(3) instrument and hot shop cell (three levels).

The reactor and all components associated with radioactive materials, except the ion exchangers, are located below the reactor hall floor for containment and radiation shielding. A reinforced concrete vault is located directly outside the containment vessel to house the ion exchangers and to provide a means for disposal of the spent ion exchange units.

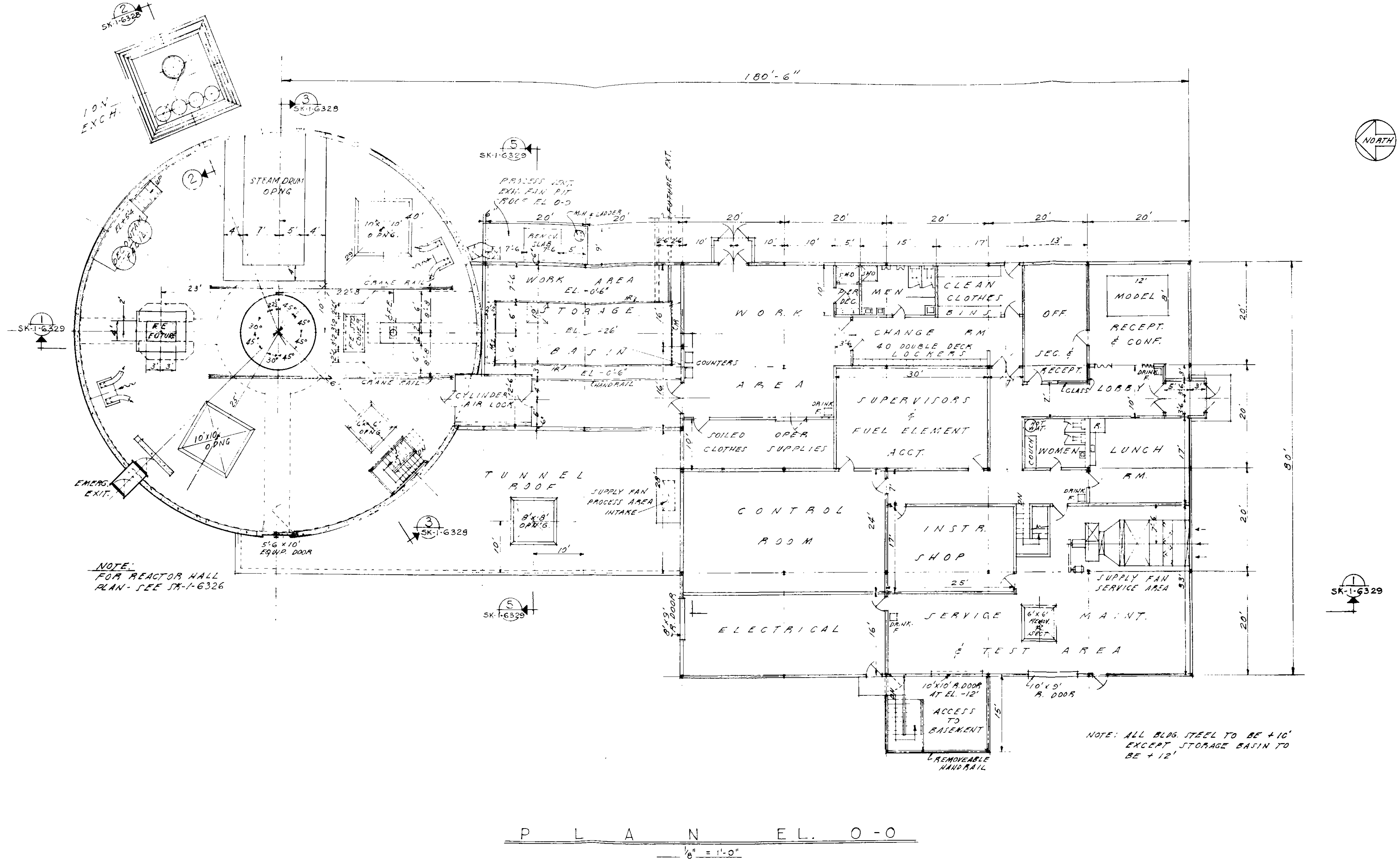
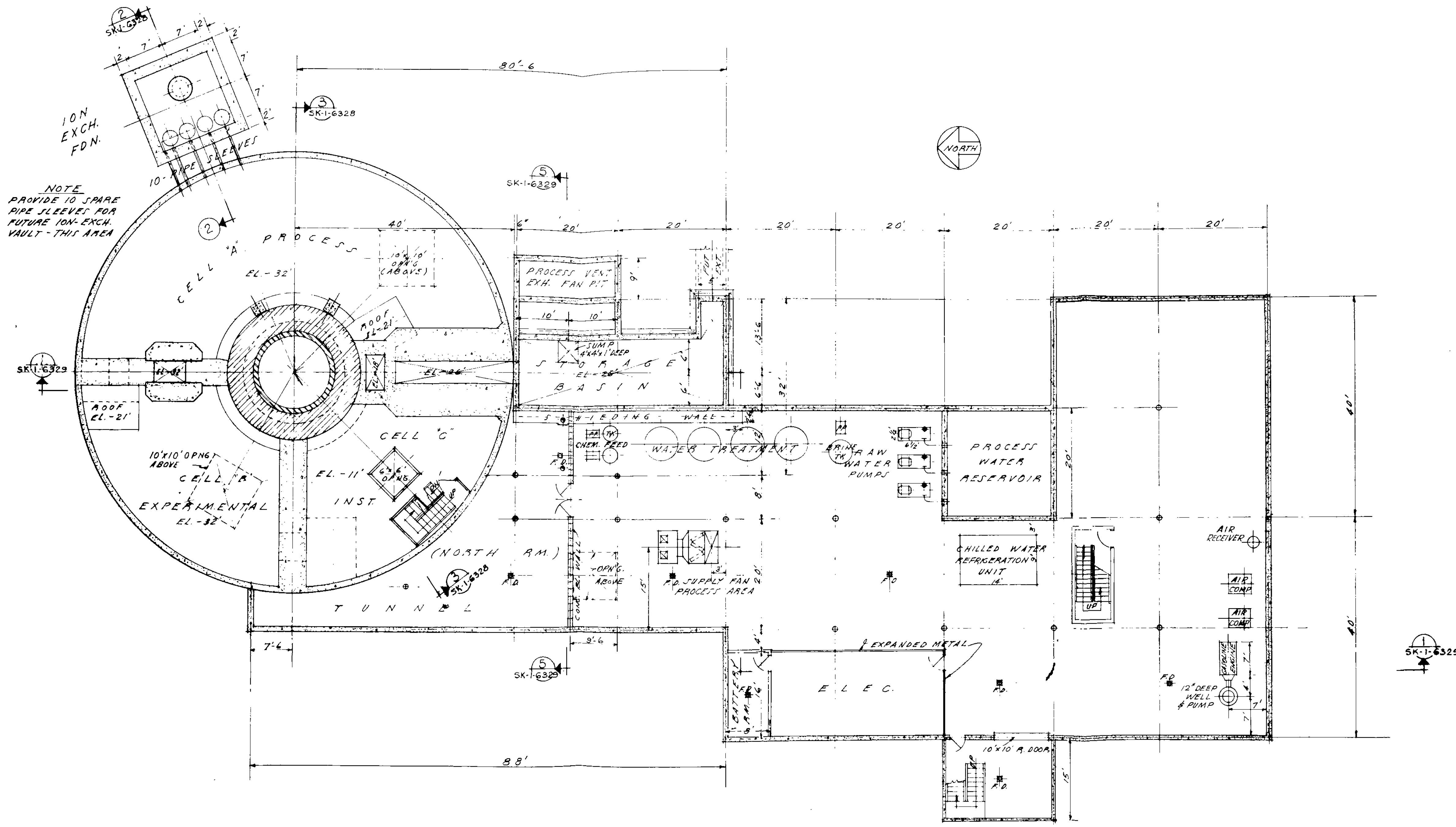


FIGURE 21

Building Plan at Grade Level
(SK-1-6325)



P L A N EL. -12'-0"

1/8" = 1'-0"

FIGURE 22

Building Plan at Minus Twelve Feet
(SK-1-6327)

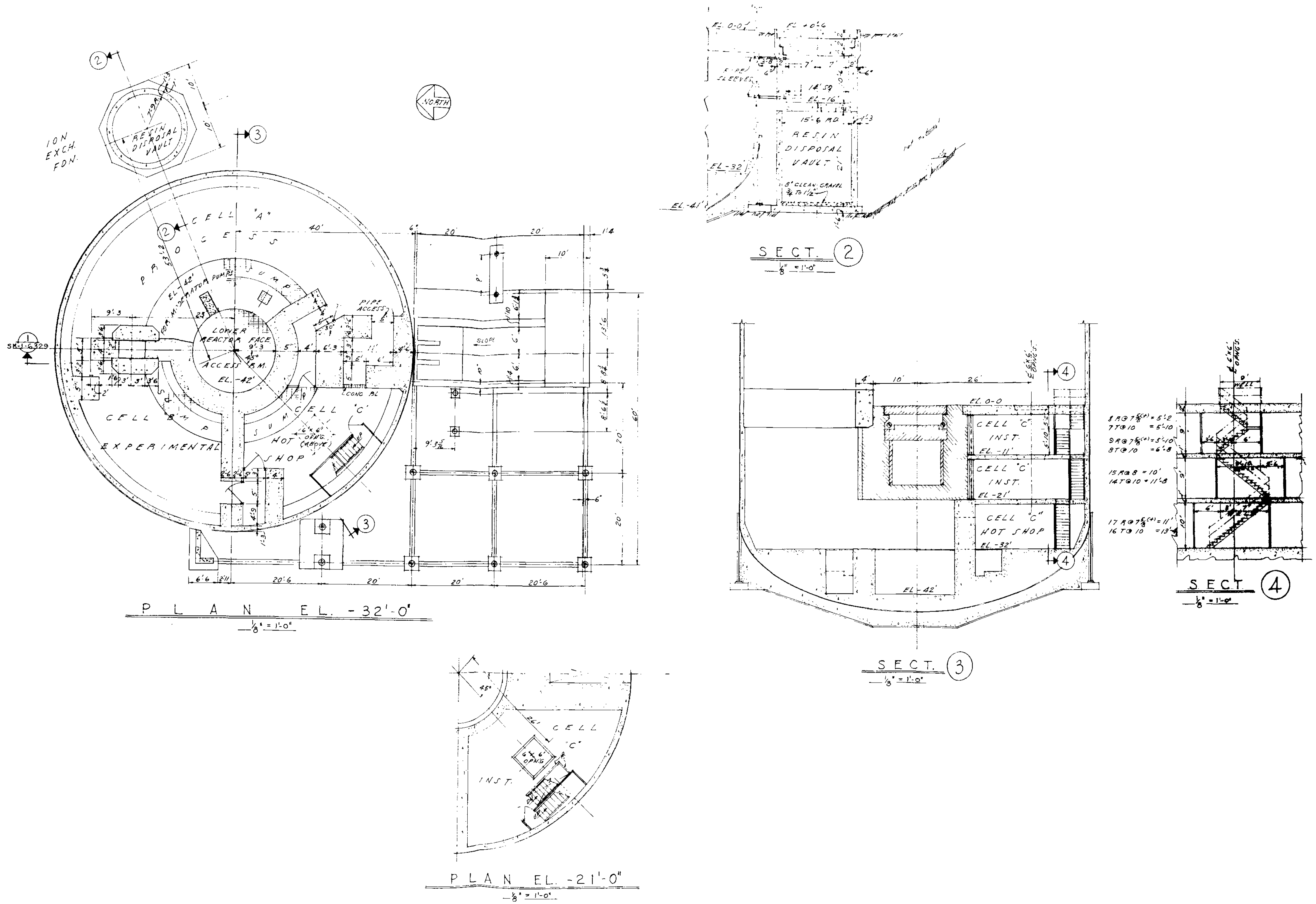


FIGURE 23

Building Plans at Minus Twenty-one and Minus Thirty-two feet
(SK-1-6328)

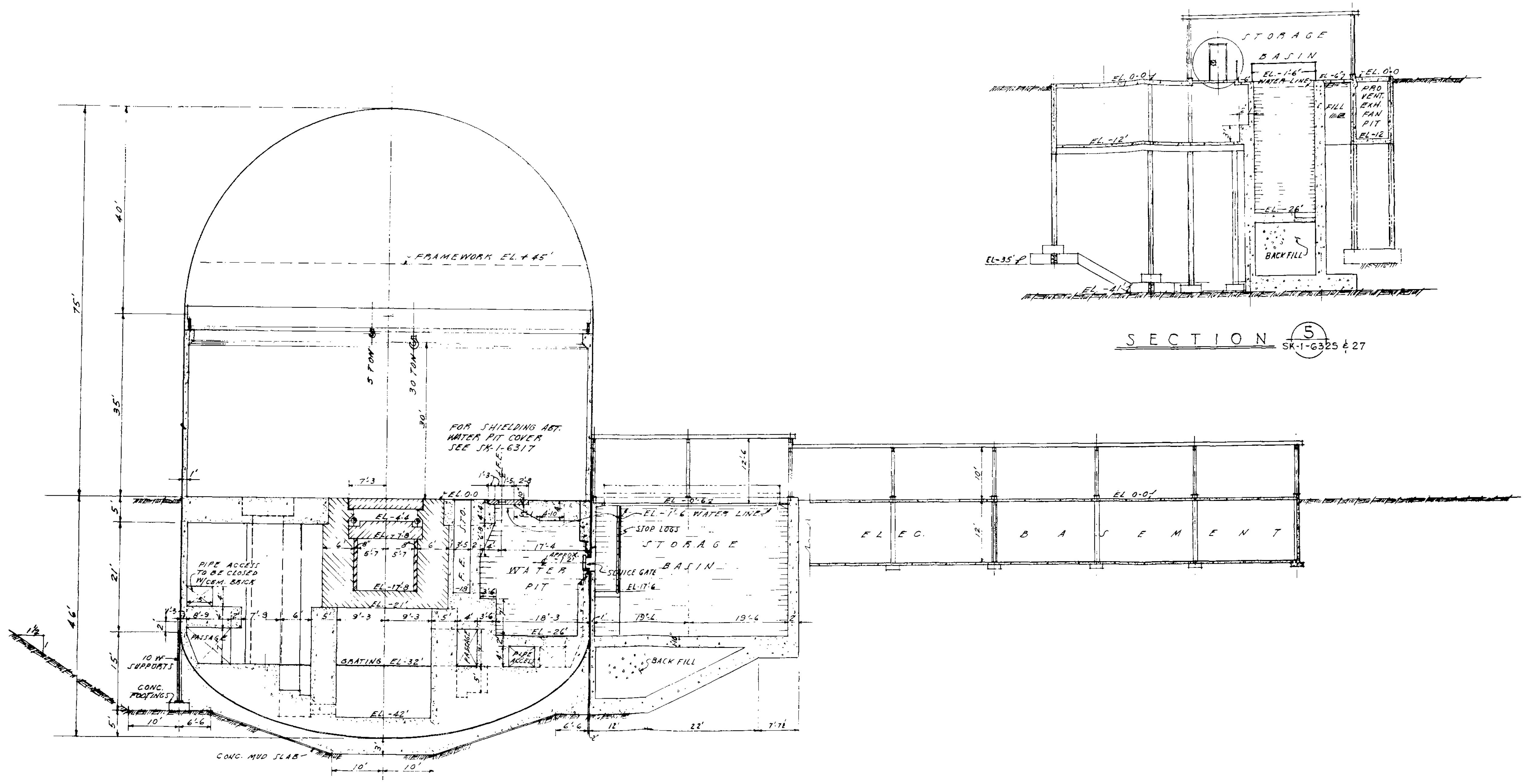


FIGURE 24

Longitudinal Section of Building (SK-1-6329)

Access to the lower cells is via stairs through the two instrument levels into the hot shop. From the hot shop, access to the process cell and the experimental cell is through labrinths in the radial shielding walls. A labyrinth through the common shielding wall between the process cell and the experimental cell is also provided.

The lower reactor face access room is located directly below the reactor with the floor at -42 feet elevation. This room houses the inlet coolant piping, moderator dump valves, and the moderator stroage tank.

Underwater fuel element storage is provided by a storage basin, 12 feet wide by 40 feet long by 26 feet deep, located adjacent to the containment vessel. A water passage into the containment vessel is provided for transfer of irradiated fuel elements from the reactor hall to the storage basin. A sluice gate in the containment vessel wall closes the fuel element underwater port in the discharge pit during reactor operation. Attached to the storage basin is a fuel element load-out facility.

b. Containment Design Features

Design and construction of the containment vessel is based upon the provisions of sections VIII and IX of the ASME Boiler and Pressure Vessel Code (1956 Edition) and Code Cases No. 1226 and No. 1228. The design pressures and temperatures are as follows:

Internal Pressure	15 psig
Equilibrium Temperature	205 F
Maximum External Pressure	0.58 psig

Pressure relief valves would defeat the purpose of the containment vessel since it is, itself, a protective device designed for complete containment of all contents. Thus, no pressure releif valves are installed on the containment vessel. A vacuum relief valve designed to open at -0.13 psig and to withstand 25 psig normal working pressure is installed on the containment vessel.

ASME Code vessels intended to contain air, steam, water, or any combination thereof are provided with a corrosion allowance of 1/6 the plate thickness or 1/16-inch whichever is smaller. However, this is not generally applicable to the design of containment vessels, since the inside is accessible for inspection and maintenance or is embedded in concrete. Since the duration of the maximum pressure, when other than atmospheric air might be present, is short, no corrosion allowance is made.

The containment vessel building was pressure tested at 18.75 psig, 125 per cent of the normal design pressure A 72 hour leak rate at 15 psig indicated a leak rate of less than the specified maximum rate of 1000 scf per 24 hours.

Internal missile protection of the containment vessel above the reactor hall floor consists of a one foot thick concrete cylindrical wall approximately 33 feet high which also supports the rails for the overhead crane. The top reactor shields and access hole plugs are anchored to the reactor hall floor.

A manually operated water fog spray system in the reactor hall will provide 500 gpm of cooling water for reducing the pressure in the containment vessel immediately following an incident Water for the spray system is supplied from the 300 Area sanitary water system. The water fog spray valve can be operated from either the control room or the 300 Area power house.

c. Pressure Equalization

The reactor containment vessel is designed to withstand the uniform internal pressure resulting from the maximum credible accident. However, in the case of a sudden rupture of a major line in the high pressure, high temperature, primary coolant system, provision must be made to distribute rapidly the released vapor throughout the reactor containment vessel. Without this provision, there possibly would be a buildup of high pressures in some compartments of the reactor building. These local high pressures might lead to failure of the containment vessel by creating high energy missiles which could penetrate the containment vessel.

To prevent such localized pressure buildup the following features were incorporated in the reactor and building design.

(1) To prevent excessive pressure buildup in the lower face access room below the reactor as a result of an inlet piping rupture, there is no wall between this room and the process cell.

(2) To prevent excessive pressure buildup in the piping space between the top primary shield and the top secondary shield as a result of a failure in the outlet piping, approximately 42 square feet of venting area is provided through the side shield into the process cell. This vent opening is sealed off at the inner face of the side shielding wall with an 8 psi rupture diaphragm. Screw operated lead shielding doors are provided to give shielding between the piping space and the process cell during reactor charge-discharge operations. A fuel element being discharged must pass through the top piping space. If the vent between the cells were open the radiation level in the process cell while a fuel element is being withdrawn from the reactor would be so high that personnel could not continue to work in the cell.

The top primary shield and the top secondary shield are designed to withstand an 18 psig internal pressure in the piping space.

(3) To provide for the passage of air and water vapor from the process cell to the reactor hall, approximately 100 square feet of venting area between these two areas is provided in the primary heat exchanger enclosure.

(4) To prevent the cover blocks over the access holes to the process cell and the experimental cell from being lifted by pressure buildup in

the cells below, these cover blocks are provided with hold down devices to withstand differential pressures up to 15 psi.

(5) To provide for equalization of pressures between the process cell, experimental cell, hot shop, and instrument cells, all doors are either louvered or wire mesh.

2. Service and Utilities Building

The service building, 100 feet long by 80 feet wide, is connected to the process area via the storage basin. All personnel entering the process area must pass through the change room facilities which consist of a work area, personnel decontamination room, locker room, shower room, and toilet facilities. Also provided in this area are; operating supplies storage, soiled clothes storage, and clean clothes storage and dispensing rooms. Immediately beyond the change room facilities is the office area, lunch room, men's and women's toilets, and the main entrance.

A corridor leads from a central point between the change room and office area to the control room, ventilation equipment room, maintenance shop, electrical switchgear room and instrument shop. From the corridor a stairway leads to the partial basement at -12 feet elevation. The basement houses the following equipment:

(1) Air compressors,
(2) Water booster pumps,
(3) Process water reservoir,
(4) Process area ventilation supply unit,
(5) Water chillers for refrigeration cooling of the reactor building,
(6) Water softeners,
(7) 480 volt switchgear,
(8) Battery room,
(9) 300 KW emergency diesel generator, and
(10) Diesel driven emergency deep well pump.

One exterior stairway and one equipment access well to the basement are provided. From the basement a 40-foot wide utility tunnel extends to the containment vessel. The north end of the tunnel houses the process water treatment equipment.

3. Electrical Power

The primary source of electrical power for operating the PRTR facility is the Bonneville Power Administration System.

For emergency service at times of BPA System outages, a 300-KW diesel generator in the basement of the service building supplies electrical power for essential services. During a BPA outage part of the lighting load is carried by D. C. -powered flood and spotlights.

Distribution in the facility is at 480 volts for all loads except the primary coolant pumps (350 hp units) which operate at 2400 volts. Total connected load is approximately 2300 kw, with normal demand estimated at 1500 kw. All instrument supply circuits, 120 volts AC, are on separate transformers supplied by separate feeders from the reactor operation power bus.

A one line diagram of the electrical power system is shown in Figure 25.

4. Ventilation System

The purposes of the ventilation system are:

(1) Assist in the control of contamination by providing directional control of airflow within the building.

(2) Maintain temperature, humidity, and air cleanliness conditions compatible with process and comfort requirements.

(3) Provide safe discharge of contaminated air.

Two independent ventilation supply systems are provided to prevent the spread of contaminated air through the ductwork to the service area from the process area. One system provides ventilation for the service area while the other provides ventilation for the process area. Reactor generated steam is supplied to the heating coils during normal reactor operation. During reactor shutdown steam is supplied by the 300 Area Boilerhouse.

The service area fresh air supply system contains filters, air washer, heating coils, and fan. The air washer is used during the summer for cooling and during the heating season to provide humidity control for greater personnel comfort. The service area above grade is divided into several zones so that more uniform temperature control may be obtained during the heating season. The supply air is tempered by the central unit heating coils while local zone reheat coils provide the final required air temperature. The basement is supplied with tempered air only. Air is exhausted from the service area through both gravity flow and powered exhausters.

The process area fresh air supply unit contains filters, cooling coils, heating coils and fan. The process area is supplied with 6500 cfm of outside air during normal operation. The reactor hall receives 5000 cfm and the storage basin receives 1500 cfm. During reactor shutdown air flow to the reactor hall can be increased to 10,000 cfm. Recirculating ventilation units equipped with refrigeration cooling coils and electric heating elements are installed in the process area. Tempered air is provided by the supply system.

Air flow in the containment building is from areas least likely to be contaminated. Fresh air is supplied to the reactor hall and flows down into the process cells. An exhaust fan draws air from the process cells and exhausts it through a 150 ft stack to atmosphere. Single, air-operated butterfly valves are installed in the main supply and exhaust ducts. The purpose of these two valves is to close the supply and exhaust ducts in the

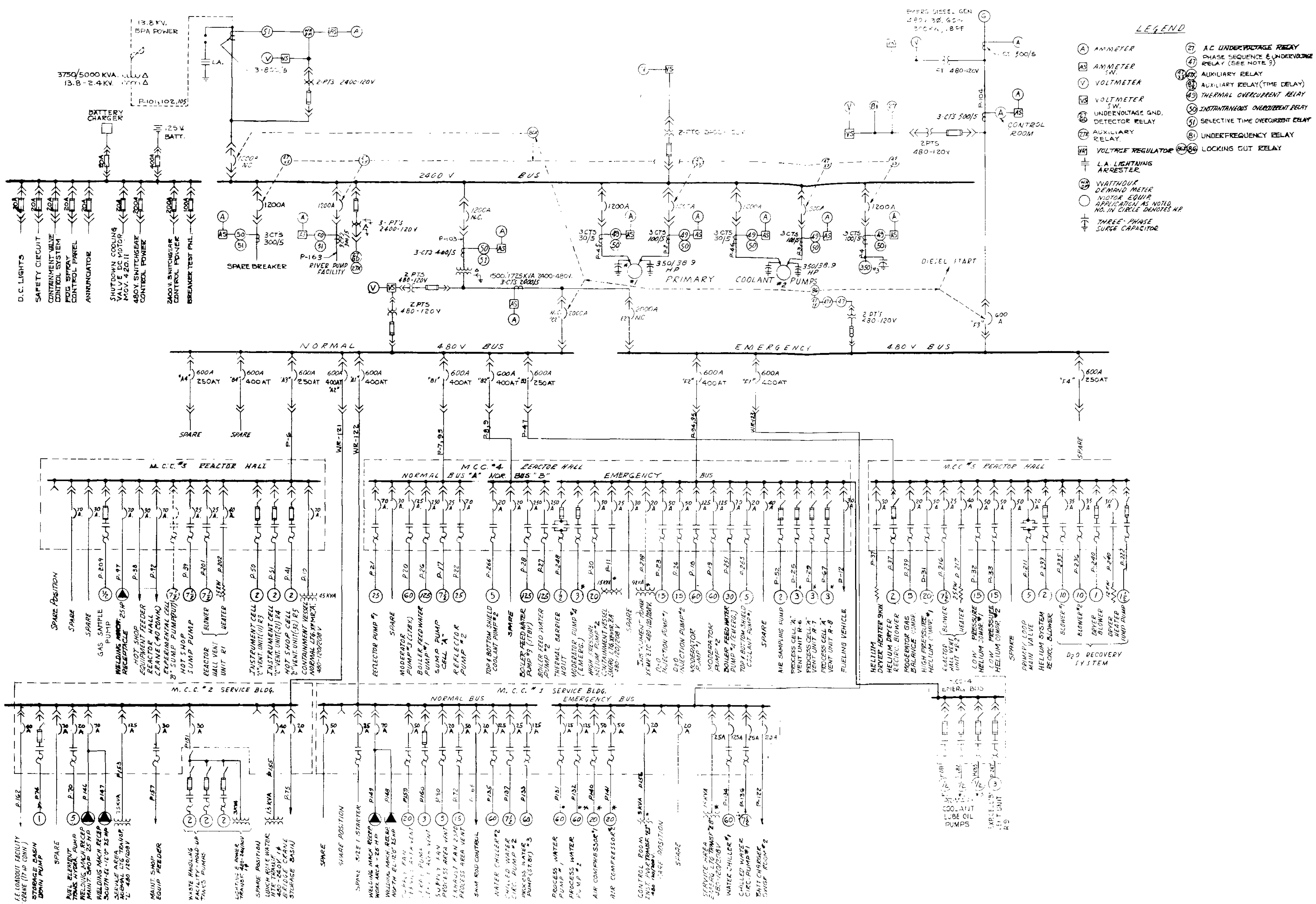


FIGURE 25

One Line Electrical Diagram
(H-3-11130)

event of an accident. The valves are manually or automatically operated to close in 0.5 seconds or less upon reciving the signal. Opening of the valves is controlled from the control room by a mechanism that does not compromise the ability of the valves to close. Closure of the valves is actuated by:

(1) Manually operated switch in the reactor control room.

(2) Radiation detectors in the exhaust duct.

(3) Radiation detectors in secondary coolant line.

(4) Failure of tripping mechanism air supply.

The process ventilation system supply and exhaust fans will stop automatically upon closure of the valves caused by (1), (2), or (3) above.

IV. SITE

IV. SITE

A. Location

The reactor is located within the Hanford Works restricted area adjacent to the 300 Area. As shown on the Plot Plan, Figure 26, the site adjoins the southeast corner of the 300 Area and is approximately 1500 feet from the west bank of the Columbia River. This site offers relatively level ground at an elevation well above the estimated 100 year maximum flood stage. The Hanford site location is shown on Figure 27.

B. Geology

Two formations underlie the site at depths shallow enough to be of concern in the disposal to ground of contaminated cooling water. They are the Ringold formation and the unconsolidated sands and gravels of the later fluviatile series of sediments. This latter formation is generally much more permeable than the Ringold formation.

This site is about 400 feet above sea level. The water table level is normally about 345 feet, or 55 feet from the ground surface. Minimum depth to ground water, during high water stages of the Columbia River, is about 35 feet. Unconsolidated sands and gravels of the fluviatile series of sediments underlie the site to a depth of about 20 to 25 feet. Beneath these fluviatile sediments are the semi-consolidated gravels of the Ringold formation to a depth of 70 to 100 feet, in turn underlain by the clays, silts and fine sands of the lower Ringold formation which extends to the basalt bedrock at a depth of about 250 feet.[8]

Estimates of transmissibilities were calculated from the observed cyclic change in water levels in wells as a result of river level changes. The velocity of water flow toward the river is estimated to be 75 ft/day. The vertical rate of movement is about 15 ft/day through the soil with an infiltration rate of about 10 to 15 gal/sq ft/day.

(8) Brown, R. W., Personal Communication, 11-5-56.

C. Hydrology

The Columbia River and its tributaries drain this region. Only the Columbia River itself directly affects the reactor site. During the past fifteen years the seasonal flow variation has ranged from a minimum of 34,000 cu ft/sec to a maximum of 692,000 cu ft/sec. The reactor hall elevation of 400 feet above sea level is 35 feet above the estimated 100 year maximum flood stage of 365 feet (flow of 740,000 cu ft/sec). Similarly, the reactor hall floor is 49 feet above the estimated average flood stage of 351 feet (flow 400,000 cu ft/sec). This site, therefore, offers safety from natural Columbia River floods.

All liquid wastes discharged to the ground in the vicinity of the site will seep into the Columbia River in a short time. Since the first intake for a municipal sanitary water system is only fourteen miles downstream from the reactor, cribbing of wastes will be limited to wastes containing less than the maximum permissible concentration of radioactive contaminants for drinking water. The Yakima River, which is free of contamination from Hanford Works, is the source of sanitary water for Richland and Camp Hanford. Kennewick and Pasco are the nearest cities which depend upon the Columbia River for sanitary water supply.

D. Meteorology

Meteorological and climatological data concerning the PRTR site and environs were obtained from the HAPO Atmospheric Physics Operation. Major equipment for observing atmospheric conditions consists of a 420-foot tower fitted with temperature and wind measuring devices and a network of ground wind measuring stations. The tower is located on a plateau about 22 miles northwest of the PRTR site. A summary of the meteorological data is given in Appendix B.

FIGURE 26

Plot Plan
(SK-1-6324)

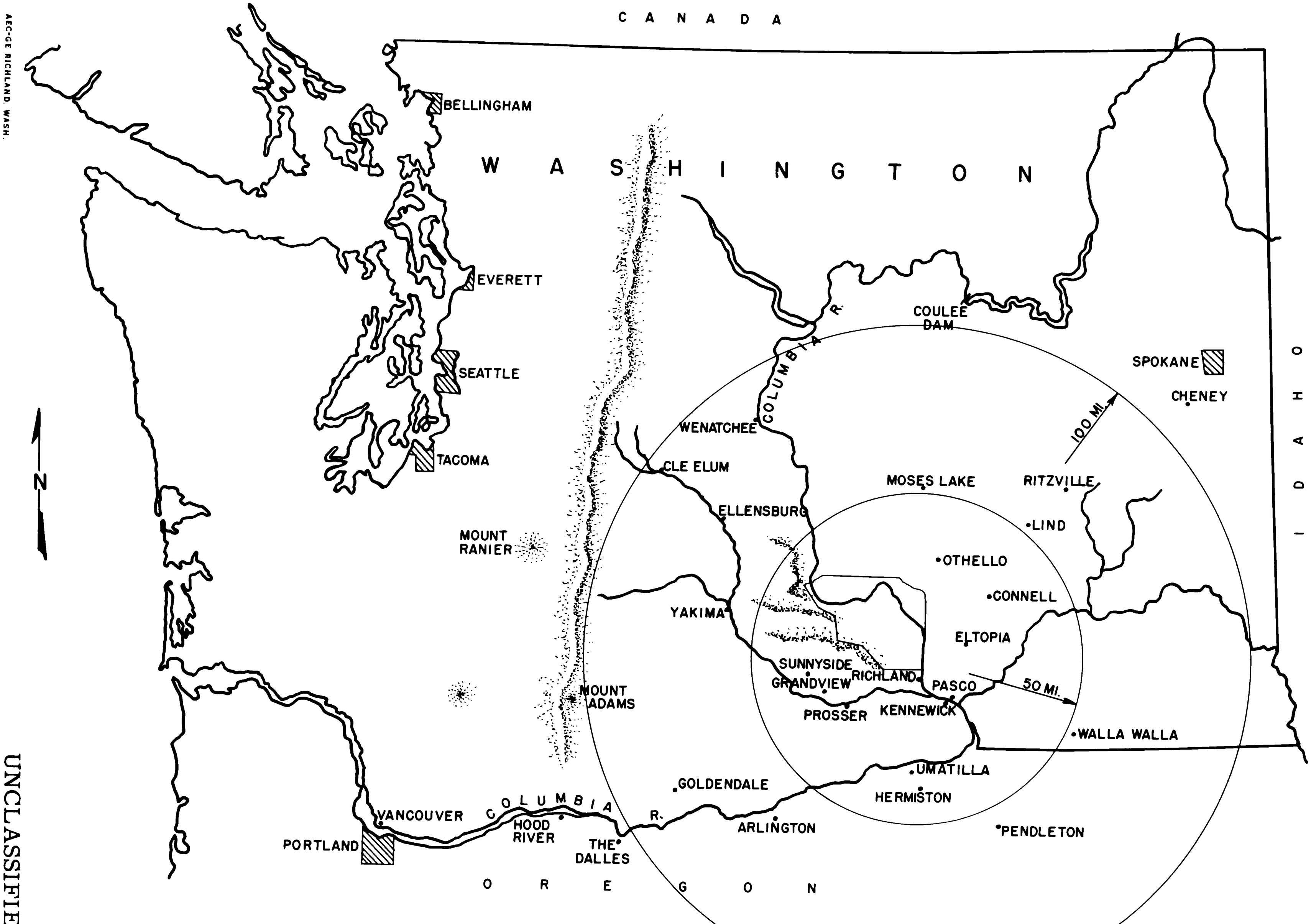


FIGURE 27

Hanford Site Location

AEC-GE RICHLAND, WASH.

The transport of air-borne materials to points up to 200 miles distant from Hanford has been considered in a special analysis of hypothetical clouds leaving the Hanford Area.[9, 10, 11, 12, 13] In this study the material was assumed to travel in accordance with the wind pattern at 5000 feet above sea level. The frequency of wind direction at 5000-foot level over Hanford is shown in Table I.

These data indicate only the initial direction of motion of an air-borne cloud leaving Hanford. However, the trajectory analyses show a considerable persistence in direction and, as indicated in Table I, the northeast quadrant is the most likely area to be affected. The results of these analyses, shown graphically in Figure 28, indicate that no point within 200 miles of Hanford is entirely immune from such releases. The speed of motion of air-borne clouds, well removed from the surface layer, may be judged from Figure 29 which shows the probability of a cloud being past any radial distance, X, within N hours after emission at Hanford.

E. Seismology

The Hanford Area is in a region which is prone to earthquakes on the threshold of moderate structural damage. Studies by the University of Washington seismologists and the latest Seismic Probability Map published

(9) Hilst, G. R. The Determination of Probable Trajectory for Airborne Waste Emitted in the Hanford Works Area, Report No. 1, January, 1951, HW-20502. February 26, 1951.

(10) Hilst, G. R. Probable Trajectories for Hypothetical Airborne Wastes Emitted in the Hanford Works Area During March, April, and May, 1951, HW-21414. June 19, 1951.

(11) Nickola, P. W. Probable Trajectories for Hypothetical Airborne Wastes Emitted in the Hanford Works Area During June, July, and August, 1951, HW-22470. October 12, 1951.

(12) Nickola, P. W. Probable Trajectories for Hypothetical Airborne Wastes Emitted in the Hanford Works Area During September, October, and November, 1951, HW-23601. February 20, 1952.

(13) Nickola, P. W. Probable Trajectories for Hypothetical Airborne Wastes Emitted in the Hanford Works Area During December, 1951, and January, February, and March, 1952, Report No. 5, HW-27172. January 30, 1953.

TABLE I

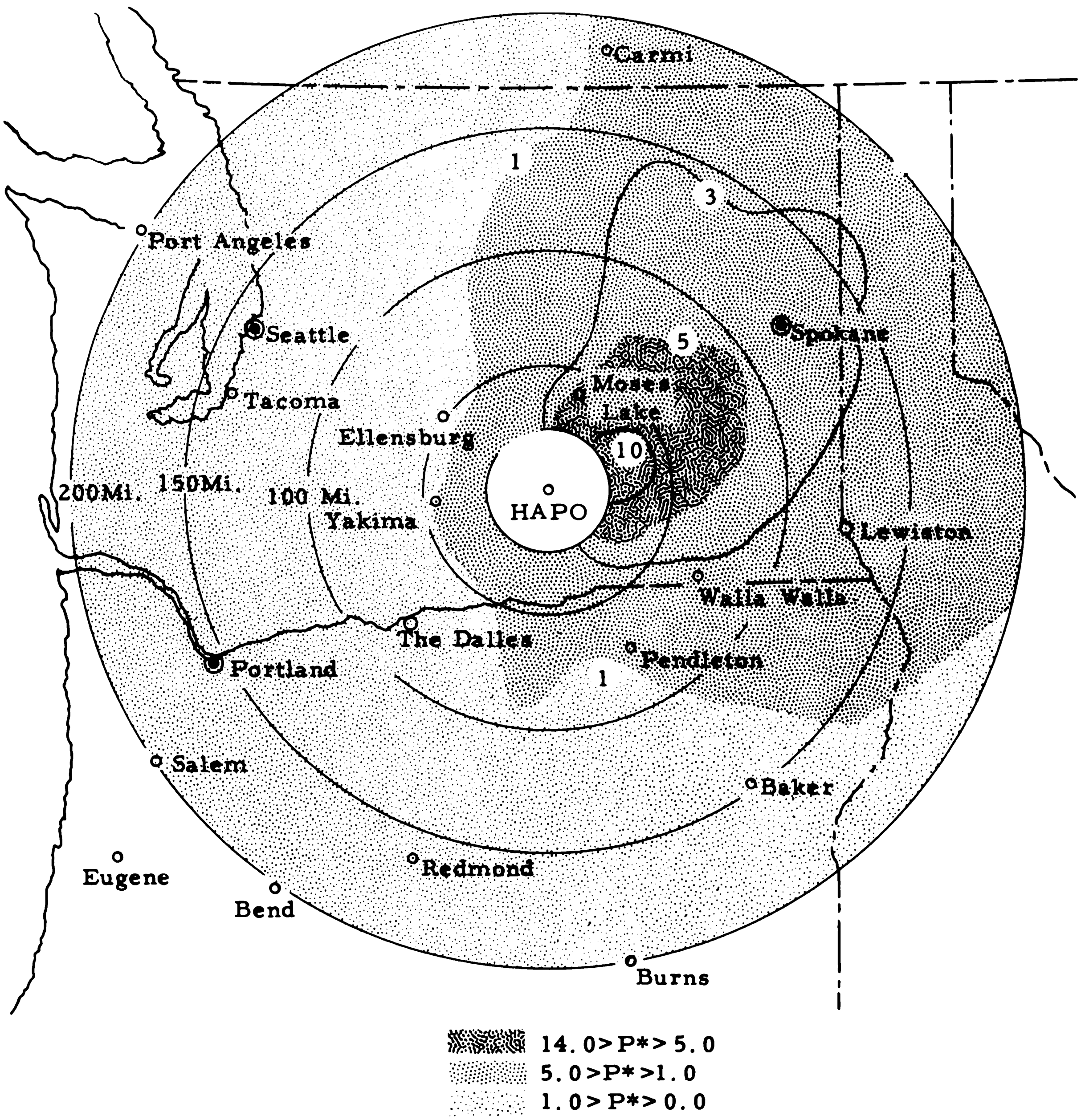
PERCENTAGE FREQUENCY OF WIND DIRECTION AT 5000-FOOT LEVEL OVER HANFORD

(Compiled from five years of record, 1946 - 1950)

Wind Direction Class Interval	Season				
	Spring	Summer	Fall	Winter	Year
345-15°	5	5	7	5	6
15-45	5	7	9	4	6
45-75	6	5	6	4	5
75-105	4	4	5	3	4
105-135	3	3	3	2	3
135-165	4	3	2	3	3
165-195	5	7	5	5	6
195-225	10	13	11	11	11
225-255	21	21	19	24	21
255-285	22	20	20	23	21
285-315	10	8	8	12	10
315-345	5	4	5	4	4

in the 1952 Uniform Building Code places all of Washington east of the Cascades in zone two. Zone two is a zone of potentially moderate earthquake damage and implies earthquake intensities of seven or eight on the Modified Mercalli Scale.

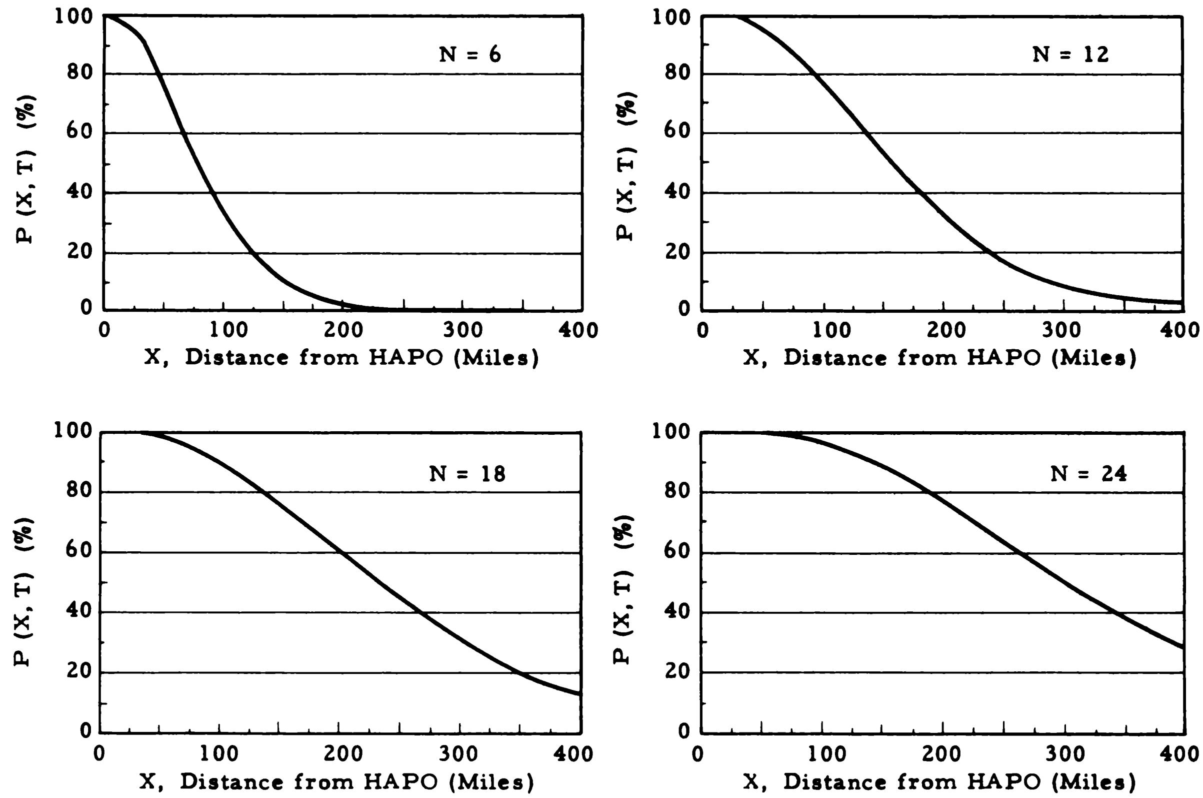
The West Yellowstone earthquake of August 17, 1959, was felt in the Hanford Area with an intensity of 3.5 to 4 on the Modified Mercalli Scale. Of the other earthquakes detected at Hanford, those of Ketchikan, Alaska, August 21, 1949, and Seattle, April 13, 1949, were approximately 6 to 7 in intensity in the Hanford region. A quake at Corfu, Washington, 15 miles north of Hanford townsite, occurred November 1, 1918, with an intensity of about 6.

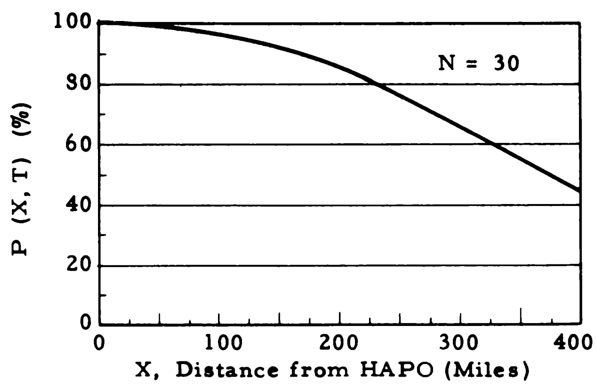


P* is the probability that a cloud one mile in diameter will pass over a point within 30 hours after emission from HAPO. The value of P* should be multiplied by 10^{-3} for true probability.

FIGURE 28

Areal Contours for P*





NOTE:
P (X, T) is the probability that a single cloud will be found at a radial distance greater than X, N hours after emission from the HAPO area. (For one year period, April 1951 - March 1952.)

FIGURE 29

Radial Movement of Clouds

F. 300 Area Facilities

In the adjacent 300 Area are located the Hanford Laboratories and the Fuels Preparation Department. The nearest building is the Plutonium Fabrication Pilot Plant 100 yards north of the PRTR. The Radiochemistry Building is about 150 yards northwest of the PRTR. About 600 yards northwest of the PRTR is the nearest manufacturing facility, the Metal Fabrication Plant, where fuel elements for Hanford production reactors are manufactured.

G. Make-Up of Surrounding Area

A graphic presentation of the results of the 1950 census for the state of Washington is given in Figure 30. A more detailed map for the immediate vicinity of the plant is given in Figure 31. The small towns around the plant are all indicated on this map while at some distance only the larger cities are indicated. It should be noted that these numbers were obtained from the 1950 census. The towns in the region of Moses Lake, Warden, and Othello north of the Saddle Mountains are growing rapidly as the irrigation project from Grand Coulee progresses.

The population of the surrounding area is further illustrated by Table II which gives some of the characteristics of the counties in the immediate vicinity.

The area in the central part of the state is largely devoted to agriculture so that major industry is at a minimum. Figure 32 illustrates the location of major industries and dams throughout the Inland Empire.[14] Small industries employing only a few people are not included.

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(14) Directory of Manufacturers; the State of Washington, Washington State College, 1952.

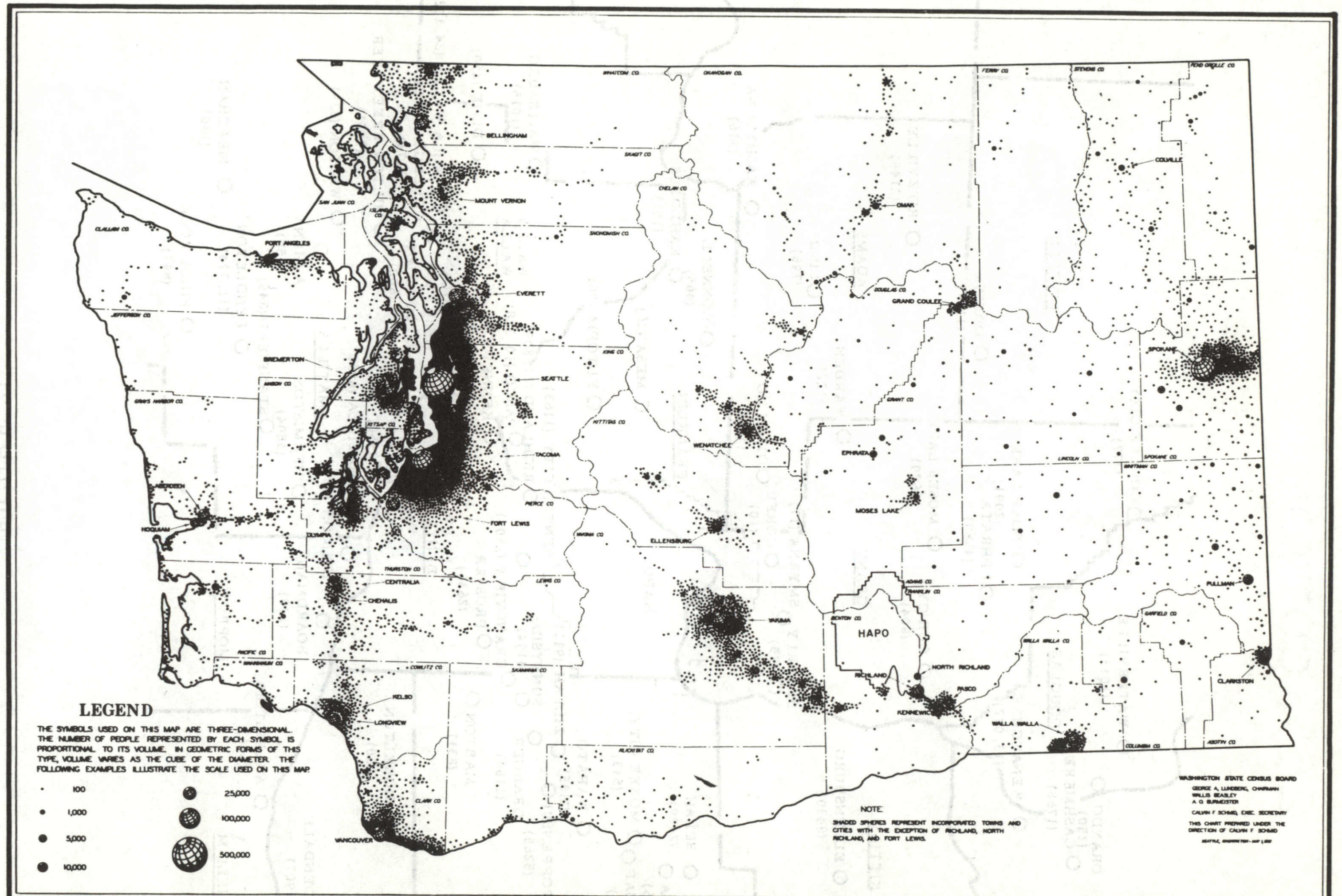

FIGURE 30

Washington Population Distribution: 1950

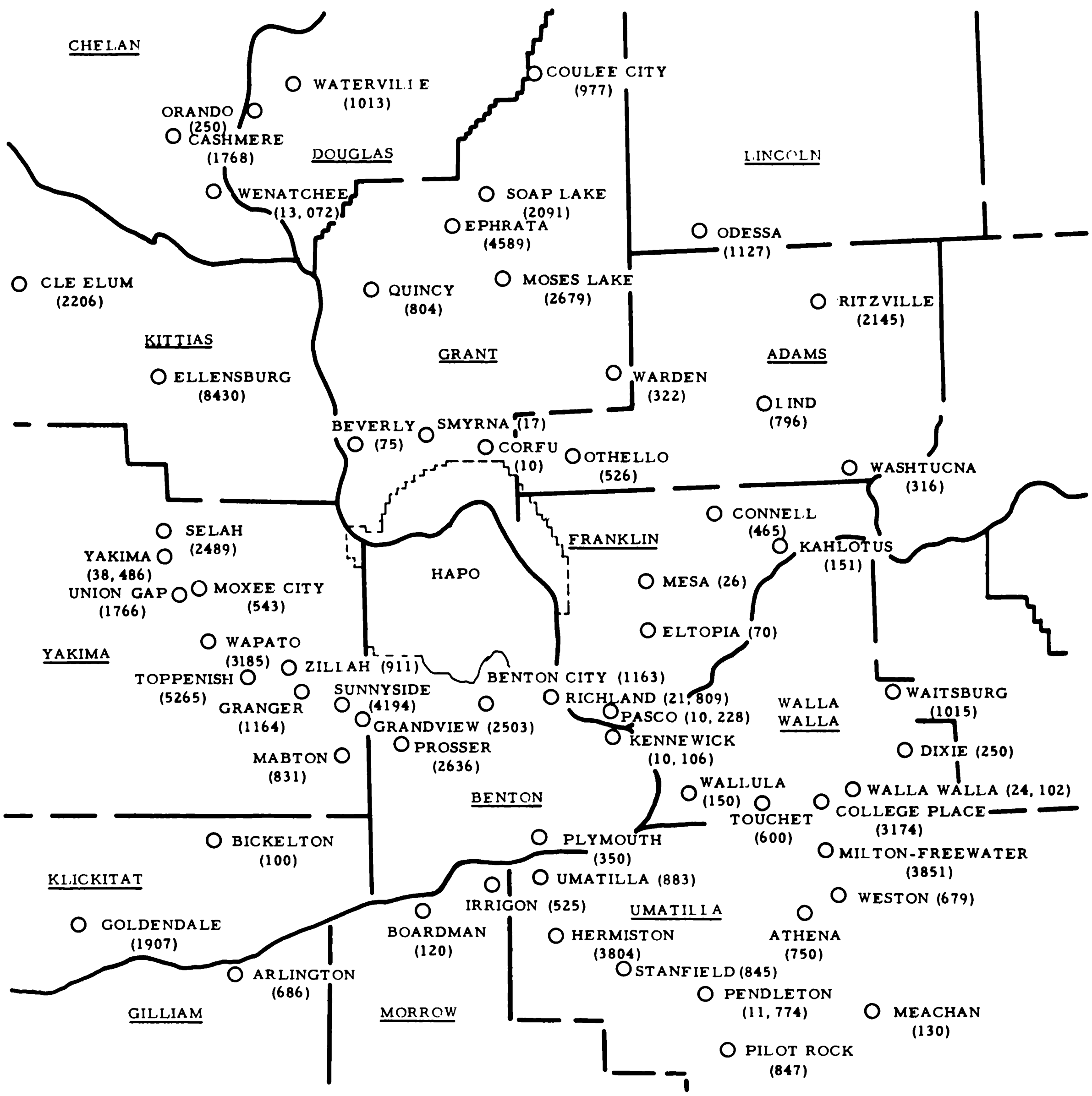

FIGURE 31

Population Distribution: Vicinity of Hanford Site: 1950

TABLE II

POPULATION CHARACTERISTICS - 1950

County	Area sq mi	Pop.	Urban[a] Pop.	Rural Pop.	Rural Pop. Density, People/mi^2
Benton	1,738	51,370	45,308	6,062	3.5
Franklin	1,262	13,563	12,343	1,220	1.0
Yakima	4,273	135,723	102,743	32,980	7.7
Grant	2,691	24,346	22,131	2,215	0.8
Adams	1,895	6,584	4,260	2,324	1.2
Kittitas	2,315	22,235	17,343	4,892	2.1
Walla Walla	1,288	40,135	35,239	4,896	3.8
Umatilla, Ore.	3,231	41,703	----	---	---
Morrow, Ore.	2,059	4,783	----	---	---

(a) includes those in communities greater than 2,500 and the rural non-farm group.

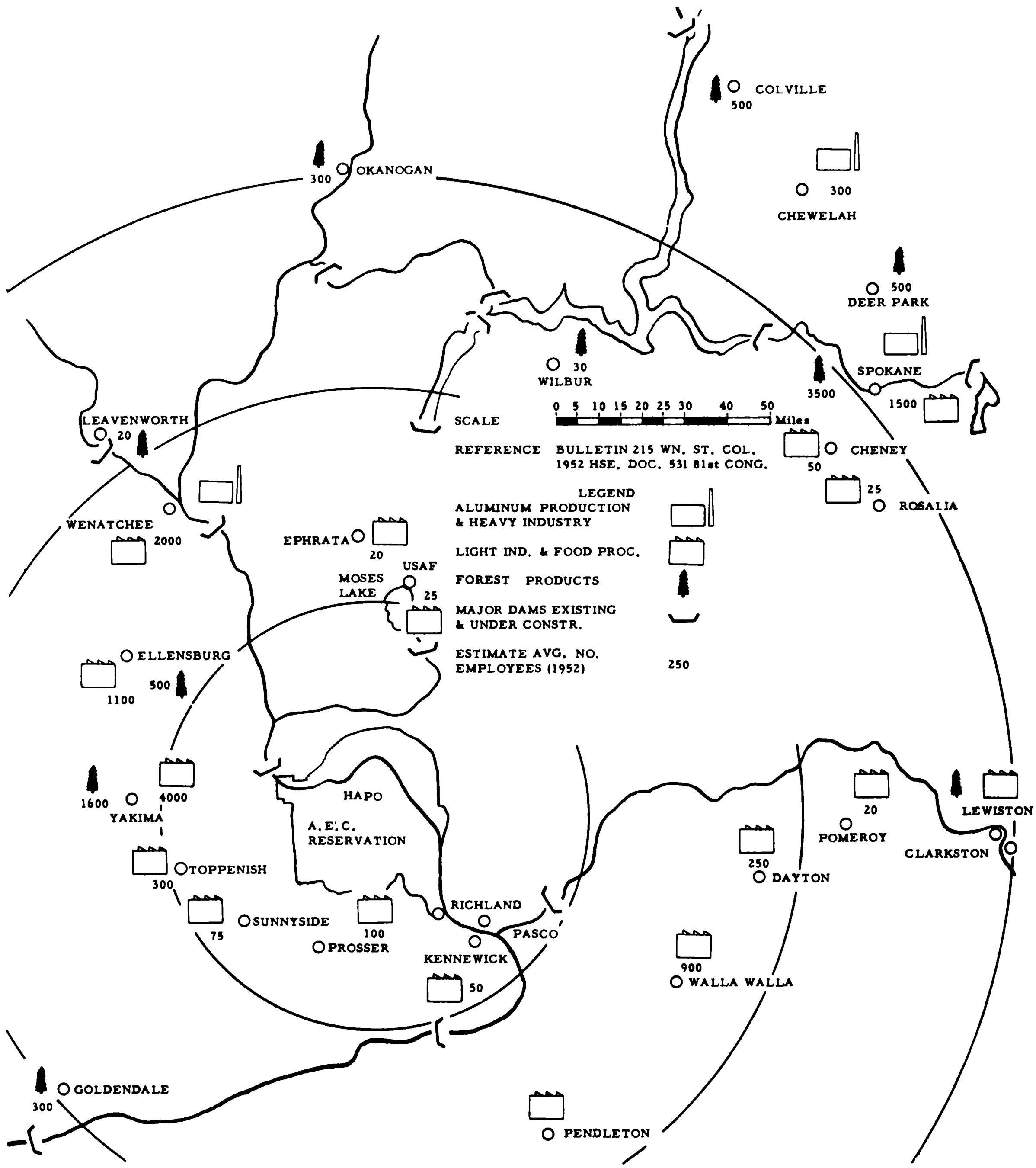

FIGURE 32

Major Industries and Dams

V. REACTOR OPERATION

V. REACTOR OPERATION

A. Operating Program

The operating program for the PRTR is primarily geared to the effective evaluation of the plutonium recycle concept.

1. The reactor will be utilized as a research and development facility to test and demonstrate the feasibility of various plutonium recycle concepts. Emphasis will be placed on investigation of physics parameters including isotope cross-sections, cycle high exposure characteristics, control characteristics and reactor dynamics.

2. Irradiation testing and demonstration of the use of both plutonium-bearing and uranium-bearing fuel elements in special purpose reactors and nuclear power reactors will be carried out.

3. A limited program of irradiation tests to determine improved materials of reactor construction is planned including improved process equipment, new pressure-tube materials, and new types of monitoring and control instrumentation. Conversion of the reactor to use of a different coolant is a contingent possibility for the mid-1960's.

4. Operating periods will be utilized to provide experience and data to assist in the optimization of the economics of plutonium recycle technology.

After acceptance and functional testing of the process equipment, criticality and zero power experiments will be performed. A series of low power operating periods will then be utilized to gradually bring the reactor to full power (70 MW). The testing program outlined above will be started after the reactor has operated for a time at full power.

B. Organization for Operation

Responsibility for directing the experimental and operating program for the PRTR is shared by several existing components of Hanford

Laboratories. However, responsibility for actual reactor operation rests with the Manager, PRTR Operation. An organization chart for operation of the PRTR is given in Figure 33.

The Manager, Reactor and Fuels Research and Development, will be responsible for the operation and over-all safety of the PRTR. Responsibility is delegated as follows to:

1. Manager, Reactor Engineering Development
   a. Conduct of reactor safeguards analyses and publication of safeguards reports.
   b. Preparation of PRTR process specifications.
   c. Audit of PRTR operation for conformance to process specifications.
2. Manager, Plutonium Recycle Test Reactor Operation
   a. Plan and direct operation and maintenance of PRTR.
   b. Plan and direct engineering studies for maximization of the utility of PRTR as test facility.
   c. Assure operational safety of the reactor.

The Manager, Programming, will be responsible for establishing priorities for tests conducted in the reactor.

The tentative organization of the PRTR Operation is shown on Figure 34. Responsibilities of the staff are as follows:

1. <u>Manager</u>
   a. Plan and direct operation and maintenance of the PRTR facility in accordance with prescribed process specifications. The facility includes the reactor proper, in-reactor and mock-up test loop facilities, and the fuel examination facility.

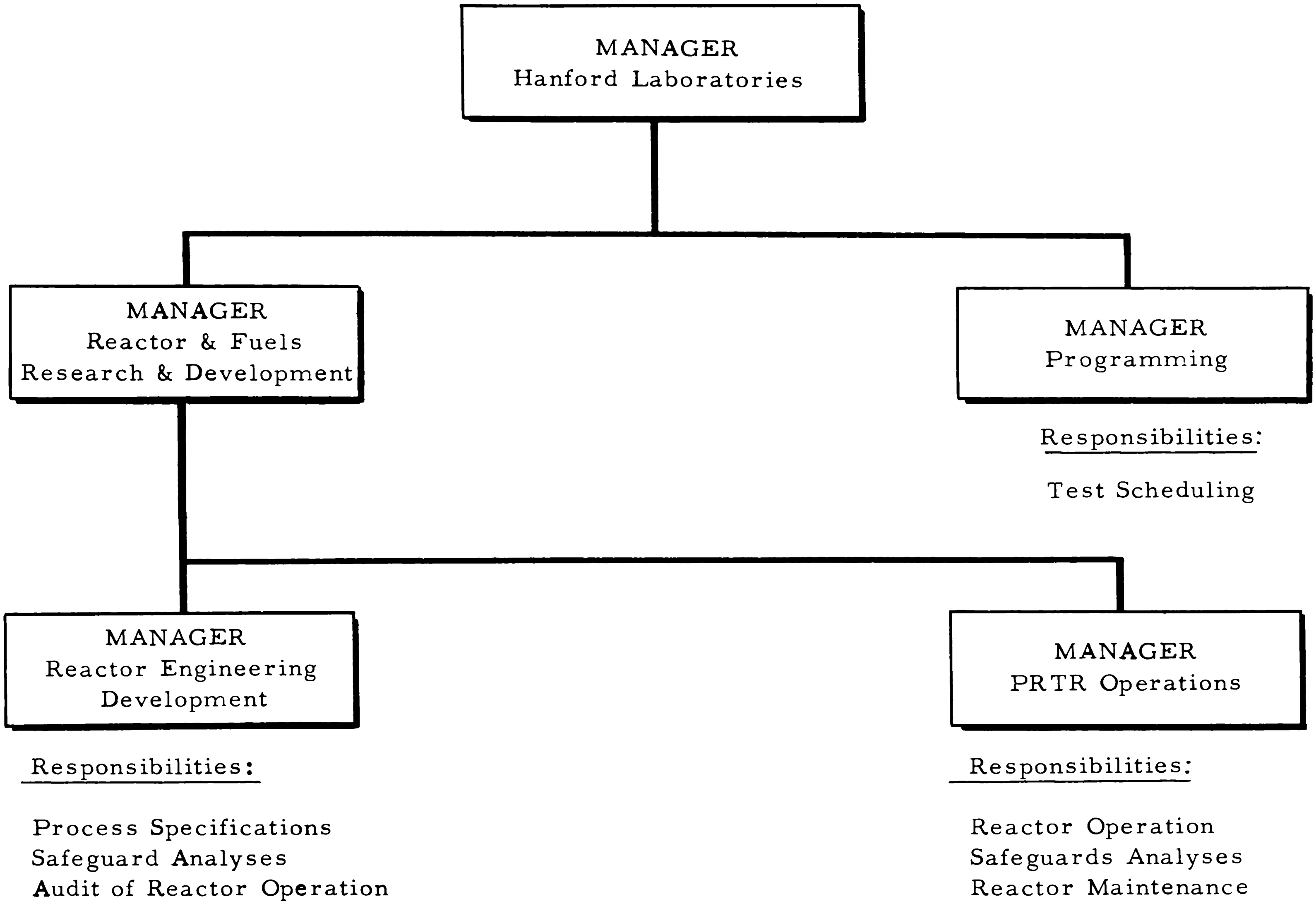

FIGURE 33
Organization for Operation of PRTR

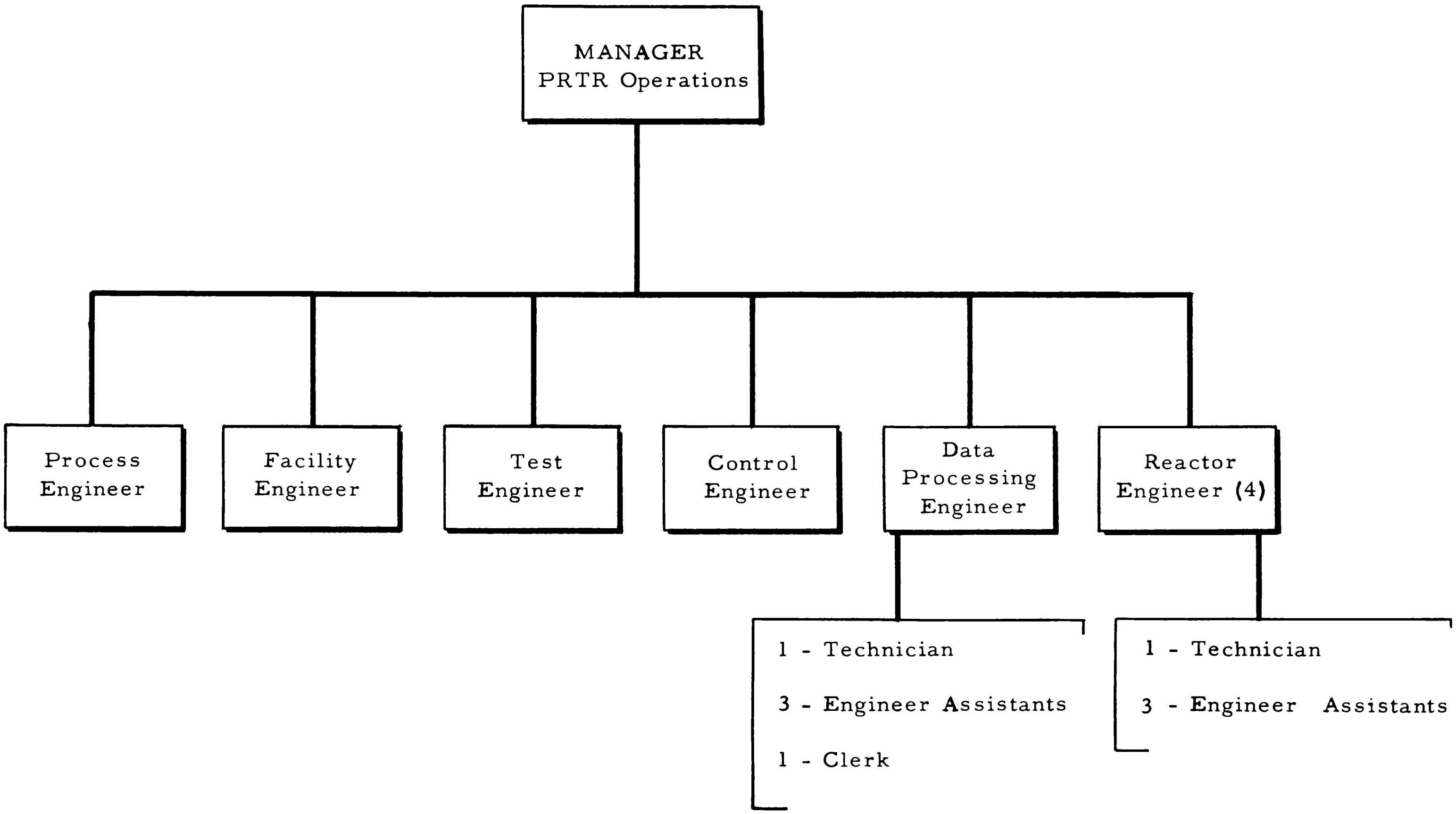

FIGURE 34

Tentative Organization of PRTR Operation

b. Plan and direct engineering studies providing bases for maximization of the utility of PRTR as an experimental test facility combined with complete assurance of operational safety.

2. Process Engineer

a. Liaison with Research and Development groups.

b. Direction of planning and analysis of the PRTR operation.

c. Process standards and engineering assistance.

d. Over-all operational analysis on reactor safeguards.

3. Control Engineer

a. Required operational physics guidance.

b. Reactor control, safety and instrumentation systems.

c. Safeguards analysis from the control standpoint.

4. Facility Engineer

a. Process and utilities equipment.

b. Plan and direct preventive maintenance program.

c. Safeguards analysis from the process equipment standpoint.

5. Test Engineer

a. Test loop and other test equipment, approving all test equipment designs.

b. Inspection, dry-run testing, installation, and removal of all installations.

c. Safeguards analysis from test loop facilities standpoint.

6. Data Processing and Relief Reactor Engineer

a. Process test data to desired storage or report form.

b. Maintain adequate SS accountability procedures.

c. Operate PRTR and associated experimental test facilities in relief of shift reactor engineers.

7. Shift Reactor Engineers

Operate PRTR and associated experimental test facilities on assigned shifts.

C. Training Program

Staffing and training personnel to operate the PRTR was accomplished in several phases over a 30-month period prior to reactor startup. The first phase of the staffing enabled the operations group to follow the initial stages of reactor scope and detailed design. In the second phase, engineers who will eventually operate the reactor were added to the operations staff. These engineers contributed to the writing of the technical manual and the operating procedures manual which were needed for training and reactor operation. The final phase of the training program concentrated on training technicians, engineering assistants, and maintenance personnel in preparation for reactor startup and operation.

Supervisory and technical personnel of PRTR Operation have accumulated 100 man-years of Hanford experience, with 80 man-years in the reactor field. The minimum reactor experience of any man is two years.

In the first phase of the operations training program, four specialist engineers were assigned to the operation to follow detailed design of all phases of the reactor. These engineers participated in design discussions, and visited various vendor plants, acting on behalf of design personnel in connection with equipment design. All design drawings and revisions to the drawings were reviewed by the operations staff before the drawings were approved. Through this teamwork, the specialist-engineers gained a detailed understanding of the design, and the supporting theory.

In the second phase of the training program five reactor engineers were added to the staff, approximately four months after the acquisition of the specialist-engineers. The first months were spent studying the design criteria for the reactor, becoming familiar with the various reactor systems. Following this, the reactor technical manual was written by the reactor engineers. Drafting the technical manual required a thorough knowledge of the reactor design and system functions.

Concurrent with the technical manual preparation, each reactor engineer spent approximately one week in training at either the Savannah River Plant, Aiken, South Carolina, or at Chalk River, Canada. The training at these heavy water reactor sites was intended to broaden the reactor engineers' understanding of heavy water reactors and problems peculiar to this type of reactor.

The third phase of the training program consisted of two parts.

1. Classroom instruction for the reactor technicians and engineering assistants.
2. Acceptance, design, and startup testing of all reactor equipment and systems.

In the class room instruction, each reactor technician or engineering assistant will receive approximately 270 hours of instruction, prepared and presented by the reactor engineers. An outline of this instruction and the amount of time scheduled is presented in Table III. Most of the sessions will include discussions of background or supporting theory followed by functional discussions. Application exercises and written examinations will be given to measure the student's mastery of the program and his preparedness to operate the reactor.

After the classroom work, the reactor engineers and the technicians will devote most of their time to preparing operating procedures and startup tests. Reactor crews will be formed and the reactor manned on a 24-hour basis for this phase of the training program. It is anticipated that the five reactor shift crews will have operated as crews for approximately six months before the reactor goes critical, and each crew will be fully capable of conducting all phases of reactor operation.

TABLE III

REACTOR TECHNICIAN AND ENGINEERING ASSISTANT TRAINING PROGRAM

	Hours
A. INTRODUCTION	
1. The Plutonium Recycle Program	1
2. PRTR Organization and Relationships with Other Groups	1
B. GENERAL	
1. Mathematics Review	6
2. Use of Slide Rule and Calculator	4
3. Radiation Training	24
4. Reactor Physics	10
5. Heavy Water	1
6. Blue Print Reading	1
C. PLUTONIUM RECYCLE TEST REACTOR	
1. Reactor and Service Facilities	5
2. Reactor Description	9
3. Fuel Elements and Fuel Element Loadings	6
4. Fuel Element Handling and Storage	6
5. Containment	4
D. REACTOR SYSTEMS	
1. Primary Coolant System	7.5
2. Moderator Coolant System	2

TABLE III (contd.)

		Hours
	3. Reflector Coolant System	3
	4. Secondary Coolant System	10
	5. Shield Coolant Systems	5
	6. Ion Exchange Systems	8
	7. Helium System	5
	8. Rupture Detection System	2
	9. Raw Water System	4
E.	REACTOR UTILITIES AND AUXILIARIES	
	1. Process and Sanitary Water System	6
	2. Heating and Ventilating System	6
	3. Electrical System	14
	4. Chilled Water System	4
	5. Compressed Air System	4
	6. D_2O Recovery System	7
	7. Waste Disposal System	4
	8. Cranes and Hoists	2.5
	9. D_2O Cleanup Facilities	2
	10. Communication System	2
	11. Experimental Loops	3
	12. Fuel Examination Facility	10
F.	REACTOR INSTRUMENTATION	
	1. Reactor Control Systems	12
	2. Reactor Coolant Systems	10
	3. Helium System	2
	4. Miscellaneous Systems	6

TABLE III (contd.)

	Hours
G. REACTOR OPERATION	
1. Reactor Startups	6
2. Equilibrium Operation	2
3. Reactor Shutdowns	8
4. Process Sampling	8
5. Charge-Discharge	10
6. D_2O, Fuel Element Accountability	2
7. Data Processing	2
H. SAFETY	
1. Chemical, Thermal Hazards	4
2. Radiation Hazards	2
3. Criticality Hazards	1
4. Evacuation, Fire Alarm Systems	2
I. MAINTENANCE WORK	
1. Decontamination	2
2. Special Tools	2
3. Work Coordination	2
TOTAL	272

D. Startup Program

The startup of the reactor is recognized as a critical period of operation. An eleven man Startup Council will direct the startup program for the PRTR and is charged with the following duties and responsibilities:

1. Specify integrated plans and schedules that will permit timely accomplishment of all phases of the total work required.

2. Transmit these plans and schedules to individuals and normal line organizations for action. Keep the Manager, Hanford Laboratories informed of progress on all phases.
3. Review and approve PRTR startup process specifications.
4. Review and approve PRTR startup operating procedures.
5. Formulate plans and schedules for startup experiments.
6. Analyze and review the results of startup experiments; recommend process specifications and operating procedures for normal operation based on the results obtained.

Various tests of the reactor are planned during the startup phase. In addition to the normal equipment acceptance tests, a program of design tests, critical tests, and power tests will be accomplished. The scope of these tests is such that a period of about six months will be required.

1. Design Tests

a. Purpose

The purpose of these tests is to determine that all of the reactor systems will functionally operate as designed. The tests will be performed prior to the charging of fuel elements and filling of systems with heavy water. Any system modifications deemed necessary from reactor safety considerations will be made prior to the reactor startup.

b. Tests

The following tests are planned:

(1) Primary Coolant System Tests

These tests will be made with dummy fuel elements in the reactor using various combinations of pumps and pump speeds. Tests will be made with the primary system cold and unpressurized, cold and pressurized, and hot and pressurized. For the hot system tests it is expected that primary coolant temperatures on the order of 450 F can be achieved by utilizing the heat generated by the primary pumps. Items to be measured are:

(a) Flows and pressure drops for the over-all system and for various portions of the system.
(b) Leakage rates.
(c) Flow decay for power loss to both one and two primary pumps.
(d) Operation of valves and the effects on the system.
(e) Vibration of system components.
(f) Volume of the primary system.
(g) Operation of auxiliary systems and components.
(h) Thermal expansion and stresses.
(i) Stability of the pressurizer level.

(2) Moderator System Tests

Tests will be made to check the moderator system and the moderator level control system. Test data will include:

(a) Flow rates and pressure drops.
(b) Dump rates; leakage rates.
(c) Vibration of system components.
(d) Stability of moderator level.
(e) Effect of loss of power to one or more pumps.

(3) Other Cooling Systems Tests

Measurements of capacities, flows and pressure drops, and tests of operability of various other cooling and auxiliary water systems will be conducted. Specific systems to be included in these tests are:

(a) Reflector coolant system.
(b) Raw water system.
(c) Secondary coolant system.
(d) Shield coolant system.
(e) Effluent system.

(4) Helium Systems Tests

Flows, temperatures and pressures in various sections of the helium systems will be measured. Operation of the following helium systems and components will be tested:

(a) Blowers
(b) Compressors
(c) Storage tanks
(d) Primary coolant system pressurizer
(e) Leak detection system
(f) Gas drying system
(g) Recombiner
(h) Deuterium analyzer

Operation of the controls and valves in the above systems will be tested.

(5) D_2O Recovery System Tests

These tests will determine the efficiency of the system. Data to be obtained includes: pump and blower flows and pressure drops, operation of the drying towers, operation of the precooler and the cooler condenser, and operation of the valves and controls. It is planned to inject known amounts of water into either the upper or lower access space to measure the efficiency of the system.

(6) Electrical System Tests

Tests will consist of interruption of normal power and the startup of the emergency system. Test data will include: time necessary for switch-over, sequencing of contacts and relays, and effects on reactor operating systems of both electrical power failure and return to normal power.

(7) Safety Circuit Tests

All components of the safety circuit will be tested by simulating off-standard or emergency conditions. It is planned to check each trip individually. Actuation times of each of the trips will be measured.

2. Critical Tests

a. Purpose

The purpose of these experiments is to determine the physics characteristics pertinent to operation and control of the reactor. Specific objectives are:

(1) Establishment of reactor fuel loading and reactivity control characteristics for the benefit of the general reactor operations program, including demonstration of nuclear safety of the reactor and its instrumentation.

(2) Testing of measurement and calculation techniques for subsequent fuel cycle experiments.

(3) Establishment of initial (zero burnup) characteristics of plutonium and uranium oxide fuels in the PRTR.

(4) Analysis of the effects of plutonium enrichment on reactor kinetic behavior and other operating characteristics.

The experiments may be classified as approach-to-critical, calibration, substitution, activation, and kinetic experiments, according to the techniques used. Many of the experiments will require special instrumentation with appropriate period and high level trips in the safety circuit.

b. Experiments

The experiments will be conducted in the sub-critical and low power (0 to 5 kw) ranges. Although detailed planning has not been completed, experiments of the following types are visualized:

(1) Approach-to-critical Experiments

These tests will provide a partial reactivity calibration of the moderator level in terms of fuel loading type and radial buckling.

Experiments will be conducted with several types of loadings: various combinations of plutonium-aluminum and UO_2 elements and, possibly, all plutonium-aluminum or all UO_2 element loadings. The loadings will be made with the moderator dumped and with one or more neutron sources placed in appropriate locations. The moderator level will be increased slowly so that steady-state neutron multiplication values may be measured. The reciprocal of the flux at the base of the core will be plotted against the moderator level as the experiment proceeds, so that the level rise may be terminated before the critical level is reached.

The radius of the more highly enriched loadings will be restricted such that criticality cannot be achieved with the moderator at full level. With less reactive types of loadings the first runs will be made using loadings too small to achieve criticality with the moderator at full level to provide accurate estimates of critical levels for larger loadings of the same types.

Calibration of moderator level as a function of radial buckling and measurement of reflector savings will be made by varying the radius of particular types of loadings and by comparing results obtained with and without the reflector.

(2) Calibration of Level and Shim Controls

These experiments will extend the calibration of the moderator level control over the full range of possible levels, with loading patterns similar to those planned for normal operation of the PRTR.

Control sensitivity at criticality will be calibrated by measurement of reactor periods. Operation with the reactor controller on manual will be required to permit observation of rising and falling periods. These measurements will be made at several moderator levels, by adjusting fuel loadings and shim rod positions to vary the critical moderator level. An indeped-dent calibration of moderator level may be made by progressively poisoning the reactor.

Shim control sensitivities will be determined by period measurements. Sufficient combinations of shim controls will be tested to provide a basis for use of the shim system as a reliable and reasonably predictable secondary control for the reactor.

(3) Substitution Experiments

These measurements include all those in which reactivity changes resulting from removal of a reactor component, and its replacement by a different component, are observed. Very small reactivity changes will be rather hard to measure without some means of establishing an accurately reproducible moderator level. The most useful substitutions will be those in which the reactivity change is large compared to the reactivity uncertainty in the level setting.

One substitution experiment, the removal of the reflector to determine reflector savings, has already been mentioned. Removal and addition of single-plutonium fuel rods should produce measureable effects, but the effect of single uranium rod removal is probably not accurately measurable. Replacement of rings of uranium fuel by plutonium fuel, and *vice versa* would be feasible, and might be carried out at intermediate stages during alteration of fuel loadings. In most cases, however, replacement experiments are better suited to other facilities designed specifically for measurement of small reactivity changes.

Coolant substitution experiments are of major importance in this testing phase. The coolant throughout the reactor may be removed to give a coolant loss coefficient, or replaced with H_2O.* The latter measurement might well be carried out at a time when H_2O was in the primary system for engineering tests, and the same fuel loading and shim settings restored with D_2O coolant at another time. Single-tube coolant replacements can also be done.

Measurement of the moderator and fuel-plus-coolant temperature coefficients, though not strictly substitutions, will be conducted in much the same way, by observing the critical level changes resulting from non-nuclear heating of the D_2O. These may be carried out in several ways, e.g. moderator heating with no coolant, with constant-temperature (hot or cold) coolant, and coolant heating with constant-temperature moderator. The reflector temperature coefficient may also be measured, although the uncertainty of the moderator level measurement may limit the accuracy of this measurement. These measurements will require temporary modifications of the coolant and moderator systems, but are of considerable importance to the startup program.

(4) Flux Measurements

In many of the above experiments flux and spectrum measurements are planned. Routine vertical flux traverses will be taken by means of ion chambers or cobalt wires in the

* It is planned to operate the PRTR with heavy water coolant. Reactivity measurements with H_2O coolant are included to provide data for the eventuality that a change to H_2O may be made in the future. A supplement to this report will be submitted for review before changing to H_2O coolant.

instrumentation channels. Short-range horizontal traverses can probably be taken by use of the access holes in the top of the calandria. Foil activation measurements may be used for determination of neutron temperature, cadmium ratio, and current. Special preparation of fuel elements for physics testing during the startup phase is not planned, with the possible exception of a few fuel rods which may be slotted for the insertion of foils.

(5) Kinetics Measurements

A very limited program of measurement of kinetics characteristics is contemplated at this time. The moderator dump transient will of course be observed, also the response of the reactor to small reactivity steps and ramps. Low-frequency oscillation of the reactor may be carried out also, possibly using the shim system for the purpose. Xenon and fuel temperature reactivity effects may be measured best when full-power operation has been achieved. The PRTR kinetics measurements, together with some of the period measurements, will require manual operation of the reactor controller, and further analysis is required to establish safe procedures for each case.

3. Power Tests

a. Purpose and Scope

The power tests are the final tests for proving operability of the reactor and associated facilities. They can be defined as low power cold and hot tests and high power tests.

Objectives of the tests are:

(1) Demonstration of the operability and safety of the reactor under power conditions.

(2) Determination of operating physics parameters such as xenon transient and reactivity coefficients.

(3) Initiation of the long range engineering physics and operational testing program.

All exploratory operation of the reactor will be accomplished on manual control. The automatic controller will be used only for operating situations which previously have been checked out on manual control.

Prior to performance of the power tests, normal startup checks will be made as well as special checks as necessary.

The following is a list of the planned types of power tests:

b. Low Power Tests

(Low power is here defined as 1 - 10 MW).

(1) Reactor Cold

The power test fuel loading will be defined and arranged following evaluation of critical tests. A series of startups will be made on manual control to determine control characteristics, test neutron and other instrumentation and equipment, and establish desired ion chamber locations. Manual startup to 2 MW will be made and the level held for a sufficient period of time to make a radiation survey of the facility. The power level will be raised in steps to about 10 MW to perform further building radiation surveys and to observe reactivity changes, control characteristics, flux distribution, and temperature coefficients. A series of startups to 2 MW on automatic period control will then be made to test the reactor controller under period control conditions. From 2 MW, the reactor power will be increased with the automatic controller to various levels below 10 MW and then decreased to 2 MW to evaluate linear rate of rise control features.

A series of tests to verify the adequacy of the safety and containment systems will be performed as appropriate.

(2) Reactor Hot

A series of tests at power levels from 2 - 5 MW will be made to test the heating characteristics of the primary coolant and operating characteristics of the secondary coolant and other systems. Further building radiation surveys will be made and reactivity, control, flux, and temperature coefficients observed.

c. High Power Tests

(1) Power Increases to Design Level

With the reactor at 2 MW and the primary coolant hot and pressurized, the reactor power will be increased in steps to levels up to 70 MW. During this period, nuclear instrumentation calibrations, general instrument performance, control characteristics, flux distributions, reactivity variations, temperature characteristics, power distributions, building radiation surveys, equipment and system performance, etc. will be determined.

(2) Full Power Tests

It should be recognized that the reactor loading for power tests may have to be altered slightly to obtain the operating characteristics desired for "normal" operation. Observation of flux and power distributions and reactivity determinations may necessitate loading changes on attainment of full power. These possible loading changes should be minor consisting of repositioning of some fuel elements or addition or deletion of of a few spike enrichment elements.

Full power tests will be designed to obtain physics and engineering data pertinent to the continued safe and efficient operation of the reactor and the evaluation of various aspects of the plutonium recycle program. Xenon transient and samarium determinations will be made to confirm theoretical calculations and aid in reactivity predictions. Reactivity lifetime tests are envisioned to aid in fuel cycle analysis and charge-discharge planning. Temperature coefficient tests will be performed routinely as a function of exposure, as will some of the critical tests, to form the basis for reactivity predictions and aid cycle analysis.

Other tests to be performed include:

(a) Flux distribution determinations and flux shaping experiments.

(b) Periodic building radiation checks to evaluate shield life and adequacy.

(c) Automatic controller tests to evaluate design.

(d) Exposure correlation tests to provide data for calculating tube exposures. Radio-chemistry determinations of exposure by comparing isotopic ratios of B^{10} and B^{11}, U^{235} and U^{238}, etc., are possibilities.

(e) Fuel substitution reactivity and power distribution tests on discharge of fuel elements.

(f) Containment integrity and safety system tests.

E. Operating Procedures

1. Startup

a. Types of Startups

Reactor startups will be of two types:

(1) Startup after an outage, and

(2) Startup within Xenon override time.

The most significant differences between the two types of startup are the amount of preparation necessary before the reactor may be started, the reactivity changes caused by temperature increases of the primary coolant, moderator, and fuel and the time required to bring the reactor to full power.

b. Startup After an Outage

(1) Preparations

In a startup following a reactor outage the primary and secondary coolants will have cooled to approximately 100 F. The startup preparations begin with a check of all auxiliary and utility systems and by placing the systems in their operaing condition. Valving, temperatures, pressures, indicating lights, radiation levels, tank levels, diesel fuel supplies for the emergency generator and the deep well pump, and equipment lubricating oil levels are checked and corrected if necessary.

The secondary coolant system is conditioned for startup by stopping the shutdown cooling water flow through the steam generator and by adding excess phosphate and sulfite to the feedwater to prepare the water for the warmup period. The main steam pressure control valve is left open at this time. Normal operating flows are restored through the small heat exchangers and all temperatures, pressures, and flows of secondary coolant system streams are then observed and adjusted if necessary.

Normal operating cooling water flow rates are restored through the shields, the reflector, and the moderator coolant systems and all applicable instrumentation is surveyed for abnormal conditions.

Before pressurizing the primary coolant system, the fuel element rupture detection equipment is placed in operational status and the reactor inlet temperature controller is set at 478 F. The primary coolant system pressure is slowly increased to 1040 psia. During the pressure rise, and at full pressure, locations with high potential for leakage are observed very closely. Primary coolant flow is raised to 8400 gpm and a final leak check is made. Any detectable leaks are repaired prior to reactor startup. Individual process tube coolant flows are checked for unusual flow rates and all other primary coolant system instrumentation is surveyed to assure proper equipment functioning.

Final control room preparations include the setting of flux monitor trips and ranges. In addition, each channel of the safety circuit is actuated and de-energizing of the safety circuit relays is observed. Artificial signals are introduced to the automatic controller and the response of the controller is checked. Dump valves are checked for proper operation by closing the valves, de-energizing the safety circuit, and observing that the valves open. Any annunciator alarms are investigated and corrected.

Core excess reactivity is estimated from a reactivity prediction curve and the shim rods are placed in the maximum poisoning position which will still allow the reactor to be operated; that is, the rods are positioned such that the maximum possible moderator level will result. Dump valves are then cocked and armed. All control room instrumentation is rescanned and corrective action taken where necessary.

(2) Approach to Critical, Warmup, Rise to Power

All flux monitor and moderator level instrumentation is continuously observed during the reactor startup. The automatic controller is set for the startup period. The helium gas balance compressor is then started and the increase in moderator level observed. Sub-critical flux level changes and the reactor period are watched closely.

The automatic controller brings the power level to 2 MW on period control. The difference between the predicted and actual moderator levels at the critical point is calculated and the shim rods are repositioned to compensate for this difference. The calibration of the automatic controller with respect to the heat generation recording instruments is checked and corrected if necessary.

By means of the automatic controller, the power level is then increased to 5 MW for the warmup period. Individual process tube flow and outlet temperature records are inspected for unusual values.

Steam generator pressure control instrumentation is programmed to increase the pressure linearly to 425 psia over a period of not less than 2 hours.

As the primary coolant and the steam generator temperatures increase with the reactor operating at 5 MW, shim rods are repositioned to maintain the maximum possible moderator level. Process tube flow, temperature, and other instruments are scanned periodically during the warmup period.

When the steam generator pressure reaches 425 psia, the reactor power level is increased by means of the automatic controller at a maximum rate of 13 MW/min to full nominal power of 70 MW. Process tube coolant outlet temperatures are observed closely during the rise to full power and all control room instrumentation is thoroughly scanned and tube temperature and flow recordings are obtained when full power is reached. Unusual readings and indications are immediately investigated.

c. Startup Within Xenon Override Time

(1) Preparations

Startup within xenon override time follows a scram or other emergency or semi-emergency shutdown after which the primary and secondary coolants have not been depressurized and cooled.

The first preparation for this type of startup is correction of the condition which caused the reactor to be shut down. Flux monitor trips and ranges are then reset. Core excess reactivity is estimated from a reactivity prediction curve and shim rods are repositioned as necessary to assure a moderator level near the maximum with the reactor operating. The automatic power level controller is set for the startup period. The moderator dump valves are then cocked and armed for normal operation.

(2) Approach to Critical and Rise to Power

All flux monitor and moderator level instrumentation is continuously observed during the reactor startup. The helium gas balance compressor is started and the increase in moderator level observed. Sub-critical flux level changes and the reactor periods are watched closely. The automatic controller brings the power level to 2 MW on period control. At this point the automatic controller is set to increase the power level to 70 MW at a maximum rate of 13 MW/min. (If the shutdown occurred prior to completion of the warmup period of a startup, the power level is brought to 5 MW for completion of the warmup.) Outlet tube temperatures are observed closely during the rise to power and all control room instrumentation is thoroughly scanned and tube temperature and flow recordings are obtained when full power is reached. Unusual readings and indications are immediately investigated.

2. Equilibrium Operation

a. Definition

The short term reactivity transients are the fuel and moderator temperature coefficients and the xenon concentration effect. When these reactivity contributions become stabilized following a startup, the reactor is said to be operating at equilibrium. The equilibrium operation is achieved after approximately 48 hours of full power operation.

b. Use of Automatic Controller and Shim Rods

Except in the event of special tests or controller failure, the automatic controller is used to maintain constant power level.

Long term reactivity changes cause variations in moderator level. To keep the moderator level within the acceptable range of height in the calandria tank, shim rod adjustments are made as necessary. Besides being used for coarse moderator level adjustments, shim rods are used for flux flattening throughout the reactor core. Also, when high flux experiments are run, shim rods are employed to provide the necessary perturbations.

c. Equipment and Instrumentation Surveillance

The power level deviation recorder and indicating galvanometers and the moderator level indicators are observed frequently. All other instrumentation and accessible equipment are observed at intervals which are determined by the relative importance and need for attention of the systems. Appropriate checks and readings are recorded.

3. Shutdown

a. Types of Shutdowns

Shutdowns are classified as two types, normal or emergency. During normal shutdowns, the reactor power level is reduced at a predetermined rate and depressurization and cooling of the primary and secondary coolant systems is done slowly. Emergency shutdowns result from unusual conditions which require that the power level be reduced rapidly.

b. Normal Shutdown

The reactor is regularly shut down for charge-discharge of fuel elements, for routine maintenance, and for installation, removal, and modification of test facilities. Briefly, the normal shutdown consists of reducing the power level to zero over a short period of time, cooling off the primary and secondary coolants in approximately four hours, and then reducing the flows and pressures of coolant streams and other miscellaneous streams.

To reduce the power level to zero, the automatic controller is set at 2 MW and a shutdown rate of 10 MW/min. When the power level reaches 2 MW, the reactor is scrammed manually. A short time after shutdown, the primary coolant flow rate is reduced to 1200 gpm.

Steam generator pressure control instrumentation is then programmed to decrease the pressure linearly over a 2-hour period to atmospheric pressure. To further reduce the temperatures of the steam generator and of the primary coolant, water flow is initiated through the steam generator by opening the steam generator-to-sewer valve.

When the primary coolant temperature is less than 140 F, the primary coolant system pressure is reduced to atmospheric pressure. The secondary coolant flow through the heat exchangers is reduced to the flow rate required during shutdown conditions. Shim rods are moved to their maximum poisoning position and flux monitors are moved to most sensitive positions. The containment vessel air lock doors are then opened and the process area exhaust fan is placed on high speed.

c. Emergency Shutdown

Reactor scrams and shutdowns for reasons other than normal charge-discharge, routine maintenance, or planned installation, removal, or modification of test facilities are classified as emergency shutdowns. Automatic reactor shutdown by means of moderator dump results from the several emergency conditions listed on page 40.

In addition to automatic emergency shutdowns, emergency shutdowns may be manually initiated for any of the following reasons:

(1) Failure of auxiliary equipment which cannot be repaired during

operation and/or which is considered serious enough to warrant a shutdown.

(2) Unexplained radioactivity increase within the process area.

(3) Activity increase in the primary coolant stream.

(4) An inadequate supply of helium or other essential operating material.

(5) An increase in leak detector dewpoints which are indicative of possible D_2O leak.

(6) Excessive loss of D_2O from primary coolant or moderator coolant system.

(7) Unexplained changes of temperature, pressure, flow, conductivity, or reactivity.

Rapid cooloff of the steam generator and primary coolant is possible and is resorted to only when conditions warrant. The cooloff may be programmed for one hour with the steam generator pressure control instrumentation or, if more rapid cooloff is required, the steam generator pressure may be reduced manually. Also, the primary coolant pressure may be reduced rapidly during the cooloff to minimize D_2O losses.

4. Charge-Discharge

Charge-discharge is accomplished using the fueling vehicle. This vehicle is permanently mounted on rails which extend the diameter of the reactor hall. A lead-shielded discharge cask and an unshielded charge tube are mounted on the vehicle. Winches with cables and fuel element lifting attachments are employed to raise and lower the fuel element assemblies.

To charge an empty process tube, an element is lifted by the fueling vehicle from the fuel storage pit into the charge tube. The rotating disc shield is indexed for the proper tube, the access plug is removed from the inner rotating shield, and the process tube nozzle cap is removed. After positioning the fueling vehicle charge tube over the process tube, the element is lowered into the process tube. Disengaging the cable from the fuel element and replacing the nozzle cap complete the charging operation.

To discharge a fuel element from a process tube, the process tube nozzle cap is removed, the fueling vehicle discharge cask is positioned over the process tube, and the cable is lowered and attached to the irradiated fuel element. The cooling air blower is started and the element is raised into the cask. The vehicle is then moved to a position over the discharge water pit. The fuel element is lowered into the carriage of the transfer conveyor in the water pit for transfer to the storage basin. In the storage area an extender is attached to the fuel element hanger and the entire assembly is moved to its storage location by means of a bridge crane.

A similar procedure is used for removing empty process tubes and shim control elements except that special adaptors are employed.

F. Control of Experiments

Responsibility for the control of tests in the PRTR will be shared, as described below, by the sponsor of the test, PRTR Operation, Reactor Engineering Development Operation, and Programming Operation.

1. Planning and Approval of Tests

The following steps will be followed in planning and approving tests.

a. Design of test and request by sponsoring component.

b. Review by PRTR Operation of request for feasibility and conformance with established specifications.

c. Establishment of any necessary special process specifications by Reactor Engineering Development Operation.

d. Establishment of priority by Programming Operation.

A proposed test not conforming to process specifications will be conducted in the PRTR only after the written recommendation of the Manager, Reactor Engineering Development, with concurrence by Manager, PRTR Operation, and written approval by Manager, Reactor and Fuels Research and Development, have been obtained. Tests which can be conducted within process specifications will require only the approval of the Manager, PRTR Operation.

2. Responsibilities of Sponsor of Test

The sponsor of a test will be responsible for the design of the test and the initiation of a request for approval of the test. In planning the test the sponsor should consider the value of the information which will be produced as well as the technical feasibility of conducting the test. In addition to the above the sponsor will be responsible for:

a. The design of special equipment required. (Specialists in PRTR Operation will be available for consultation with sponsor's representatives on the design of test equipment);

b. Procurement of the special test equipment;

c. Furnishing information on the effects of the test on reactor physics and reactor process;

d. Preparation of the operating procedures for the test; and

e. Specifying the form and frequency of experimental data to be obtained in conjunction with the test.

3. Responsibilities of PRTR Operation

PRTR Operation will be responsible for the safe conduct of the test in such a manner that desired data are obtained. PRTR Operation will also be solely responsible for taking the necessary action in the event that a difficult or hazardous situation arises as a result of the test. In addition, PRTR Operation will:

a. Conduct analyses of the proposed test to determine the feasibility, and availability of suitable reactor space;

b. During initial contacts between the test sponsor and Programming, assist Programming in setting priorities based on the practicability of performing the test;

c. Determine if the test can be conducted within the approved process specifications;

d. Review the effect of the test on reactor safety;

e. Provide consulting assistance to the sponsor in the design of special test equipment;

f. Review design of special test equipment for proper fit and compatibility with PRTR;

g. Insert all experimental equipment and materials into the reactor and install equipment service lines and instrument connections; and

h. Remove all equipment and dispose of radioactive components at termination of the test.

4. Responsibilities of Reactor Engineering Development Operation

Reactor Engineering Development Operation* (REDO) is responsible for conducting reactor safeguards analyses, publishing safeguards reports, writing the process specifications for PRTR, and auditing reactor operation for compliance with process specifications. REDO responsibilities for the approval of proposed tests will be limited to those tests which cannot be conducted within approved process specifications.

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* In the Hanford Laboratories, Reactor Engineering Development Operation is a component of Reactor and Fuels Research and Development. REDO personnel include technical and scientific specialists in the functions of heat transfer, fluid flow, design development, reactor safeguards, reactor process engineering, reactor physics, reactor instrumentation, shielding, and mechanical equipment development.

a. If a proposed test cannot be conducted within approved process specifications or involves conditions outside the bounds of reactor safeguards approval REDO will,

(1) Conduct an independent safeguards analysis of the test and recommend design changes to permit performance of the test without compromising reactor safety;

(2) Publish a revision or addition to the Final Safeguards Report presenting the results of the above analysis; and

(3) Write special process specifications for the proposed test provided the specifications do not violate the provisions of the Final Safeguards Report and subsequent revisions.

5. Responsibilities of Programming Operation

In conjunction with planning functions for the Plutonium Recycle Program the Programming Operation will review each proposed test to determine the technical value of the test to the Plutonium Recycle Program and to assign a priority to the test. During the initial contacts with the test sponsor, Programming Operation will obtain PRTR Operations advice on the practicability of performing the test. All tests will be reviewed to determine that they will not seriously interfere with the Plutonium Recycle Program.

6. Performance of Tests

a. Responsibility for Operation of PRTR

(1) The Manager, PRTR Operation, is responsible for and held accountable for the operation of the PRTR and its associated experimental tests.

(2) No one outside this organization is permitted to perform any manipulations in or around the reactor, including the instrument consoles of experimental equipment, without direct authorization from the operating supervisor on duty at the time.

b. Sponsors' Representatives

A reasonable number of technical representatives of sponsoring components will be granted approval by Manager, PRTR Operation, on request of the sponsor, for the purpose of observing and facilitating the conduct of their experiments.

G. Process Specifications-Operating Limits

Process specifications are provided for the PRTR where reactor safety is concerned. The specifications will consist of limits to critical process conditions, safety circuit trip point settings and annunciator alarm settings. Specifications will be established, modified, or rescinded by the recommendation of Manager, Reactor Engineering Development; with concurrence by Manager, PRTR Operation; and approval of Manager, Reactor and Fuels Research and Development. The essential process specifications for power operation are presented in brief form in Tables IV and V. It should be recognized that many of these values are conservative and will be revised in a realistic direction and submitted for ACRS review after the period of initial reactor startup pending the results of current analyses and the startup tests.

H. Evacuation Procedure

The 300 Area Evacuation Plan is followed by all personnel and visitors in the 300 Area and the PRTR site. The plan is initiated whenever emergencies arising from plant radiation hazards, natural disaster, or enemy action may compel evacuation of the premises.

The general signal that any type of evacuation has been initiated is a steady sounding of outdoor sirens and indoor building evacuation claxons throughout the 300 Area. In addition to these signals, key buildings in the 300 Area, and the PRTR site, are notified of the evacuation by telephone over a crash alarm system.

TABLE IV

SAFETY CIRCUIT TRIP POINTS

Item	Minimum Value	Design Operating Value	Maximum Value	Unit
Pressurizer Pressure	1005	1025	1130	psig
Tube Coolant Flow	90	100	120	Per cent of setting
Pressurizer Liquid Level	-4	Arbitrary Setting	+4	inches
Liquid Level in Steam Generator	65	75	--	inches
Steam Generator Pressure	--	425	465	psia
Period During Startup	10	--	--	sec
Neutron Flux During Startup and Warmup (at 10 MW)				
Logarithmic Channels	--	--	0.15	Ratio of actual flux to flux at maximum operating level
High Level Channels-High Trip	--	--	0.15	Ratio of actual flux to flux at maximum operating level
High Level Channels-Low Trip	2	--	--	Per cent of full scale
Neutron Flux After Warmup (to Full Power)				
Logarithmic Channels	--	--	1.3	Ratio of actual flux to flux at maximum operating level
High Level Channels-High Trip	--	--	1.3	Ratio of actual flux to flux at maximum operating level
High Level Channels-Low Trip	2	--	--	Per cent of full scale
Secondary Coolant Activity	--	--	1.2×10^{-2}*	$\mu c\gamma/cc$
Exhaust Air Activity	--	--	4.5×10^{-8}**	$\mu c\gamma/cc$
Seismoscope				
High Sensitivity	--	--	II	(Intensity on Modified Mercalli Scale of 1931)
Low Sensitivity	--	--	V	(Intensity on Modified Mercalli Scale of 1931)
Instrument Air	75	100	--	psig

* Also actuates "total containment".
** Also actuates "ventilation containment".

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TABLE V

RANGE OF OPERATING VARIABLES

Item	Minimum Value	Design Operating Value	Maximum Value	Units
Reactor Thermal Power	--	70	70	MW
Maximum Heat Flux				
Mark I Fuel	--	300,000	400,000	BTU/(sq ft)(hr)
Mark II Fuel	--	400,000	400,000	BTU/(sq ft)(hr)
Tube Power	--	Variable	1,200	KW
Tube Outlet Temperature	--	542	545*	F
Bulk Outlet Temperature	--	530	535*	F
Process Water Supply Reservoir Level	5*	---	---	feet
Liquid in Deaerator	1050*	1,200	---	gallons
Boiler Feedwater Pressure	425*	450	---	psig
Spike Enrichment Pu in Reactor	--	---	14	kg
Plutonium in One Fuel Element	--	---	0.33	kg
Number of UO_2 Fuel Elements	42	---	---	Number of fuel elements

* Annunciator alarm setting

When any of the signals is heard, personnel evacuate the area as rapidly as possible by means of evacuation buses, private vehicles, and government vehicles. These vehicles are located in the parking lots or near the 300 Area gates.

Key personnel perform assigned duties as prescribed by established procedures and then evacuate in the available transportation.

Evacuation of the 300 Area is normally through the south gate and badge house and through the parking lot. In an evacuation occasioned by an accident in the PRTR the west gate will be used.

The caravan of evacuating vehicles proceeds south on state highway 4-South for approximately two miles and then west to Whitstran, approximately 25 miles southwest of the PRTR.

Following an emergency evacuation, the caravan is allowed to return to the area only after an "all clear" is received.

VI. SAFETY ANALYSIS

VI. SAFETY ANALYSIS

A. General Safety Features

1. Inherent Safety

This group of characteristics includes those safety features of the reactor which do not depend upon external systems or operating procedures for their effectiveness in the limitation of power excursions.

a. Doppler Coefficient of Uranium Oxide

Due to the comparatively low thermal conductivity of UO_2 as compared to uranium or other metals, almost the entire heat generation in an excursion is initially available for the increase of the fuel temperature, except for about three per cent which is generated in the moderator and coolant by gamma rays and neutron moderation, and the still smaller contributions from neutron-induced reactions outside the fuel element. As compared to uranium metal fuel, UO_2 has a far higher heat capacity, $\sim$22 cal/(mol)(C), and a somewhat lower Doppler temperature coefficient, so that higher energy releases are required to produce a given negative reactivity. Nevertheless, with 50 per cent or more of the reactor loaded with UO_2 fuel the prompt temperature coefficient will be the predominant effect in the termination of a fast excursion, aside from the scram system, and of major significance even in a slow runaway. It is completely instantaneous in its response and limited in magnitude only by the attainable fuel temperature in the excursion. Since the negative reactivity due to the Doppler effect is sensitive to the total amount of U^{238} present in the reactor, the total release of energy during an excursion terminated by the Doppler effect alone is sensitive also to the amount of U^{238}. Cores containing large amounts of Pu-Al fuel elements or other enriched types are not as inherently safe as cores with a large U^{238} content. This point is illustrated by Figure 35, where the highest relative energy release results from the most highly enriched

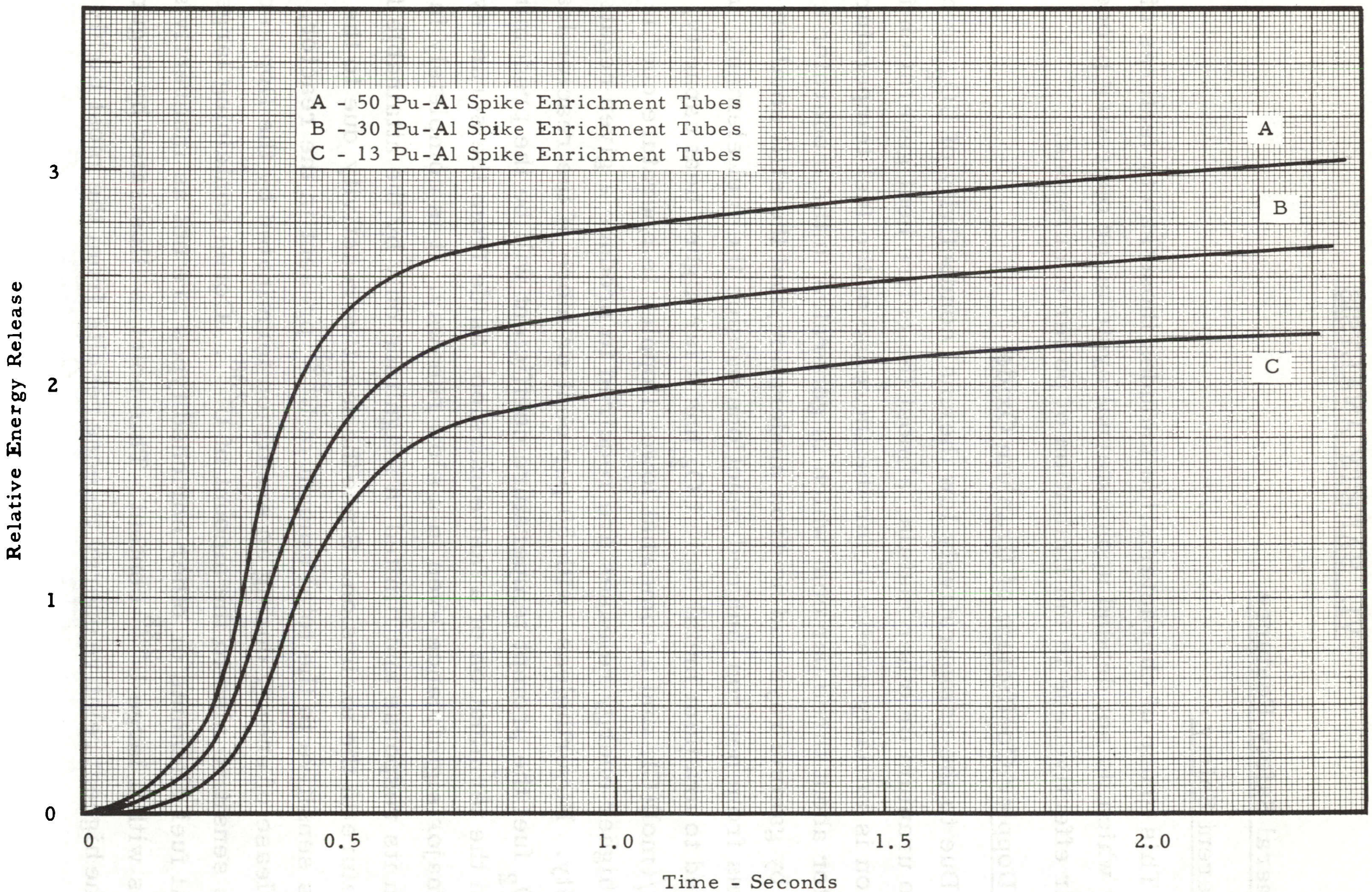

FIGURE 35

Effect of Plutonium Spike Enrichment on Energy Release core for a given reactivity input. The Doppler coefficient measured in the Physical Constants Test Reactor for a full UO_2 fuel loading in the PRTR lattice without coolant was

$$\frac{1}{p}\frac{dp}{dT} = -(2.3 \pm 0.2) \times 10^{-5}/C$$

which corresponds to an effective resonance integral coefficient

$$\frac{1}{\Sigma_{res}(20C)}\frac{d\,\Sigma_{res}(T)}{dT} = (2.0 \pm 0.2) \times 10^{-4}/C$$

Although this value appears satisfactory in the range of fuel temperatures from 20 to 300 C, fuel temperatures reached in a nuclear excursion will, in general, be much greater than this. Recent measurements and calculations indicate that the resonance integral coefficient of UO_2 decreases with increasing temperature and may be of the order of half of the PCTR measured value at fuel temperatures reached in an excursion.[15] Therefore, the following two values have been adopted for this analysis:

$$\frac{1}{\Sigma_{res}(20C)}\frac{d\,\Sigma_{res}(T)}{dT} = \begin{cases} 1.5 \times 10^{-4}/C & \text{"Best guess"} \\ 1.0 \times 10^{-4}/C & \text{"Minimum value"} \end{cases}$$

b. Formation of Vapor Voids

If the coolant and moderator are substantially subcooled, as during the early portion of a reactor startup, an excursion probably would be terminated by other effects long before the coolant and moderator reach the boiling point. Radiolytic decomposition of the moderator may, however, take place to a small extent in a severe excursion. At normal operating temperatures, voids would form fairly rapidly in the coolant. The coolant

(15) Blomberg, P. and E. Hellstrand, Ibid, page 58.

void coefficient is dependent to some extent upon reactor loading as well as fuel element design. However, for contemplated loadings it is expected to be zero or slightly positive and thus would contribute nothing towards the termination of an excursion. Voids in the moderator could not occupy a significant fraction of the moderator volume until a substantial amount of moderator was expelled from the core. This process would be of importance only if, by some undefined means, the reactor safety circuit were to fail.

c. Moderator Temperature Coefficient

The bulk of the moderator temperature coefficient ($\sim -10^{-4}$/C) is the result of thermal expansion of the moderator. The bulk of the heating of the moderator is a result of the direct heat generation in the moderator ($\sim$ 3 per cent of the total heat) and does not involve much time delay. However, the total temperature increase possible is not large (30 to 40 C) and the negative reactivity represented by this is only 3 to 4 mk, which is in general less than the Doppler effect. The expansion of the moderator increases top reflector thickness, but this means only that the vertical leakage (about 25 per cent of the total leakage) is more or less unaffected, since the migration is inversely proportional to the square of the moderator density and the vertical geometric buckling is essentially directly proportional to the square of the density. The multiplication constant and radial nonleakage probability are the predominant effects in the moderator temperature coefficient, and these are independent of the moderator level but strongly dependent upon the density.

It is concluded that the prompt or Doppler negative temperature coefficient of the fuel is the chief inherent safety mechanism of the PRTR. It should again be noted that the inherent safety characteristics of the PRTR are not unlike those of light-water-moderated reactors; the presence of larger amounts of plutonium has an unfavorable effect on the total delayed-neutron yield, but this is compensated for by the favorable effects of delayed photo-neutron production in heavy water.

2. Primary Controls

The primary control system will be capable of controlling reactivity by adjustment of moderator level in the range from 36 to 111 in. through varying settings of the control valves. This range of control represents a control strength of some 300 mk. This range can be extended by operation of the shutdown valve in conjunction with the control valves to a lower limit of moderator level of 20 inches. This increases available strength in the primary system to 890 mk.

The limiting rates of reactivity change have been shown in Figure 13, page 34. Maximum rates of reactivity decrease are shown for control valve action only, and for control valve action supplemented by shutdown valve action. Operation of the shutdown valve by itself (with control valve closed) would provide reactivity decrease rates similar to the limiting rates for control valve action.

Maximum reactivity decrease rates attainable by control valve action are about 0.25 mk/sec in the normal operating range of moderator levels, and increase to a peak of 0.85 mk/sec at a moderator level of 3.8 feet. The additive action of the shutdown valve increases the capacity of the control system considerably; rates of decrease of about 0.65 mk/sec can be attained at maximum moderator levels. The rate of decrease is much greater at lower moderator levels; at any level above 2.6 feet, the rate of reactivity decrease attainable through combined control-shutdown-valve action exceeds the maximum rate of increase which can be obtained.

The increases in reactivity attainable with the control system are limited by the capacity of the compressor which produces the balancing pressure. In the normal operating range the maximum increase rate is about 0.15 mk/sec. The maximum rate of reactivity increase is greater at lower moderator levels, where the change of reactivity with level is greater.

During equilibrium operation (after startup reactivity transient), the moderator level will normally be maintained within the top six inches of the calandria. By maintaining the level within this range, the maximum excess reactivity which can be introduced by changes in moderator level is about 4 mk under the worst conditions. Recovery from such a situation could be quickly effected by the safety system. The reactivity could increase no faster than the maximum rate shown on Figure 13, page 34.

An alarm system will indicate when the moderator level is outside the normal operating range, requiring reactivity adjustment with the shim controls.

Operation within the normal operating range is not feasible during startup of the reactor. During this period of operation, with the moderator at relatively low levels in the calandria and with high attainable rates of reactivity increase, a complex sequence of control and safety system failures could, conceivably, permit introduction of sufficient reactivity to initiate a serious power surge. Such a sequence, however, must entail failure of the controller and of the period and/or power level safety trips, and must assume non-operation of the fail-safe features of these components; or, it must assume a reactivity increase so large and so rapid that neither the controller nor the safety system can respond in time.

Figures 36 and 37 show the lower limits for the subcritical period during approach to critical for a number of varied reactor loading conditions. The curves were developed from the approximate formula:[16]

$$T = \frac{-\Delta k}{\frac{dk}{dt}}$$

where T is the period in seconds. Any real period must be equal to or greater than the value indicated by the formula, owing to delayed neutron effects. The curve for any real case will deviate more and more from the limit as criticality is approached. The curves shown were developed, based

(16) Schultz, M. A. Control of Nuclear Reactors and Power Plants, McGraw-Hill, N. Y. 1955.

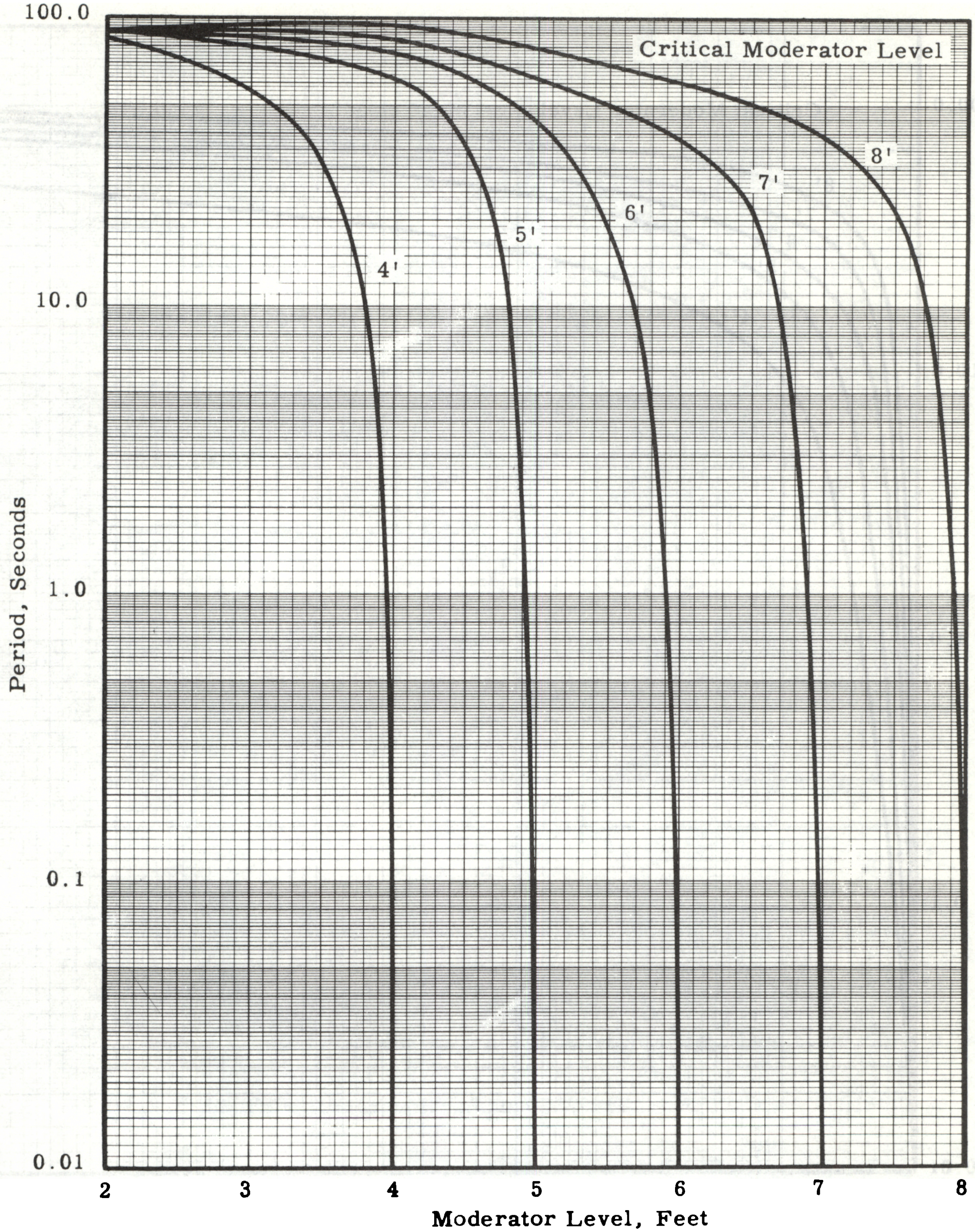

FIGURE 36

Lower Limits for Subcritical Period During Startup

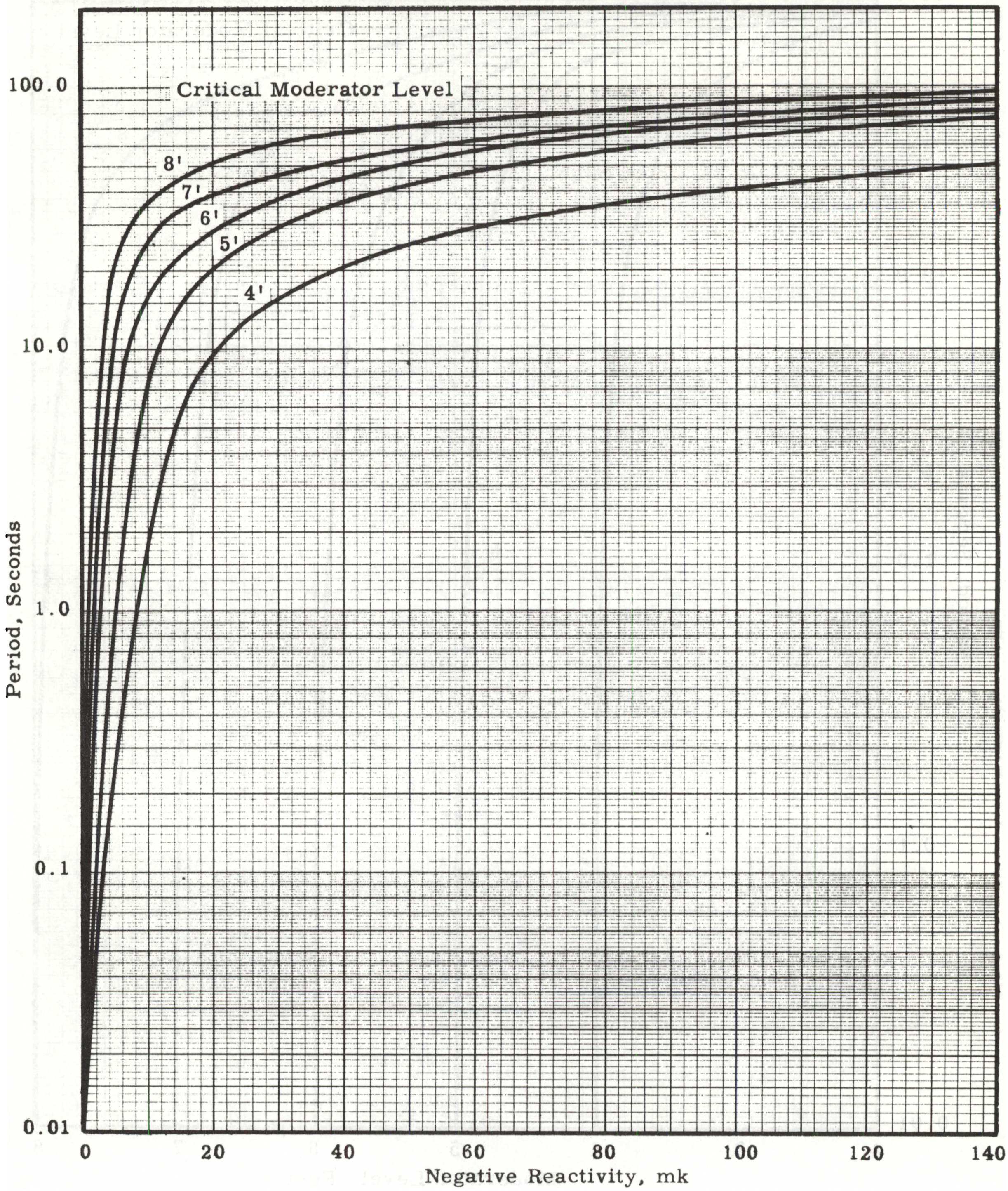

FIGURE 37

Lower Limits for Subcritical Period Versus Reactivity on the maximum rate of increase of reactivity, Figure 13, page 34. In reality, once a finite period signal is received from the monitoring channels, the controller will assume control of the reactivity increase rate and will partially open the control valves, slowing the rate of increase.

The period limits shown indicate that it would be very difficult to develop a situation whereby the reactor approached criticality with a dangerously short period, except possibly at very low levels of criticality. Even in these cases, adequate time would be available for control action, or, supposing that to fail, for actuation of the safety circuits.

Analyses of credible accidents during both startup and steady state operation are presented in Section VI-C.

3. Shim Controls

Total control strength of the shim system, with optimum rod distribution, is ~ 115 mk; the average strength for each half rod is 2.1 mk, and the nominal maximum, ~5.3 mk.

Movement of the rods within the reactor is accomplished through manual controls in the control room. The maximum rate of reactivity change due to controlled movement of the shim rods is ± 0.12 mk/sec. This rate is well within the range of control action of the primary control system.

The only apparent way in which significant rates of reactivity increase might be caused by the shim system is the accidental dropping of half rods from the calandria. The maximum measured rod fall rate, upon complete disengagement from the sprocket drive assembly and shaft hubs, was 30 ft/min. Two half rods falling at this rate would introduce reactivity at an average rate of 0.68 mk/sec as shown on Figure 38. As seen on Figure 13, page 34, the moderator level control system is capable of decreasing the reactivity at a minimum rate of 0.65 mk/sec at normal moderator level, using both the control valve and the shutdown valve. With the

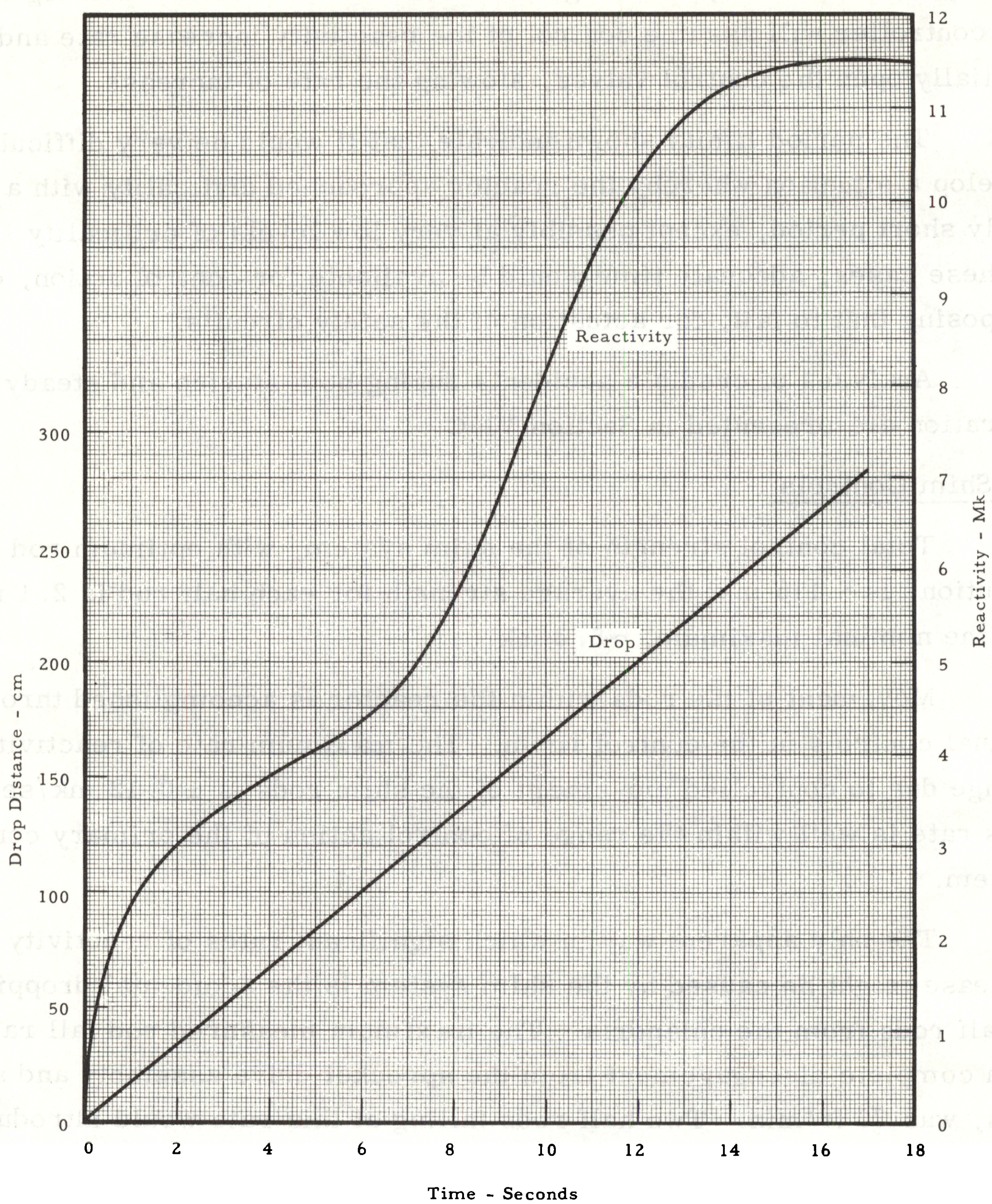

FIGURE 38

Drop Rate and Reactivity Effect of Two Falling Maximum Strength Shim Rods controller properly functioning, the excursion caused by two half rods failing could be essentially negated. A case in which it is assumed that the controller does not react to decrease the reactivity is analyzed in Section VI-C.

4. Safety System

The safety system is capable of rapidly decreasing reactivity in the reactor by about 1400 mk by draining the calandria. The rapid initial decrease in reactivity during a scram, some 18 to 20 mk within 1 second, provides a high degree of protection for the reactor.

The reactivity decrease afforded by the safety system has been shown in Figure 17, page 44. This curve does not provide any allowance for instrument time delays. In analyses of nuclear excursions presented in this report an allowance of 0.2 seconds was made for safety circuit and valve delays.

The curves shown are for dumping the moderator when the moderator level is initially at its maximum. For lower initial moderator levels, the actual rate of level decrease during dump is less during the initial stages of dump. However, since at lower levels reactivity is more strongly dependent on level, the rates of reactivity change under these conditions are as great as or greater than those shown.

Design of the safety system is such that failure of any two of the four dump valves to open would not appreciably reduce scram rates. The valves themselves are of fail-safe design, each valve being held closed by a solenoid mechanism and opened by a powerful spring aided by the gas pressure existing across the valve. The dump valves are opened by the action of a scram signal which cuts current to the solenoids. Failure of electrical power to the valves would thus be equivalent to a scram signal, and would cause the dump valves to open.

Circuitry connecting instrumentation to the safety system is of fail-safe character. Relays involved in the circuit supplying the dump valve solenoids are triplicated and connected in series - parallel such that dropping of any one relay will open the four valves. The instruments themselves, are also largely fail-safe. Exceptions are the period meters which indicate an infinite period on power failure or certain component failures and mechanical liquid level and pressure sensing devices. The latter are triplicated and coincident circuitry is used; trip of two of the three sensing elements is required to initiate scram. The multiplicity of detectors and coincident circuits provides protection against non-safe failures and eliminates the need for bypass switches.

The only credible mechanisms for failure of the safety system which do not involve extremely improbable sequences of failures, and failures overriding fail-safe features, are those involving deliberate sabotage of the system.

B. Reaction to Off-Standard Conditions

1. Reaction to Pressure and Temperature Change

Pressure and temperature changes (exclusive of the fuel temperature) result in reactivity changes primarily through density changes in coolant and moderator. The moderator temperature coefficient is calculated to be

$$\frac{1}{k}\frac{dk}{dT} = -1.06 \times 10^{-4}/C.$$

Thus, cooling of the moderator can result in a reactivity increase, but the length of time required to change the moderator temperature appreciably rules out any dangerous rate of reactivity introduction. The total average rise in moderator temperature is 44.4 C (80 F), going from cold to hot operating conditions so that only ~5 mk is potentially available from moderator cooling under normal conditions.

The moderator void coefficient is

$$\frac{dk}{k} = -0.323\frac{d\rho}{\rho}$$

where ρ represents moderator density. Thus, if the moderator contained one per cent voids, collapse of these voids would introduce 3.23 mk. Two ways have been postulated in which boiling can occur in the moderator.[17]

Gamma heating could possibly cause boiling in a region of low circulation. Extensive investigation on a full scale mockup of the calandria has failed to disclose any regions of zero circulation. Further, it has been determined that boiling could occur from this source only if it were possible to form a hot spot of moderator 65 F above the outlet temperature, or a vertical temperature inversion of moderator of at least 10 F/ft. Since the moderator is in contact with the shroud tubes, which in turn receive heat by conduction and radiation from the process channels, eccentricity of a process tube by 0.219 inches may result in mild surface boiling.[18] Such a misalignment is extremely unlikely and could occur in only a few tubes. It is also possible that mild boiling will occur on the shroud tubes at the moderator surface. This is due to the poor heat transfer from the exposed shroud tubes to the gas above the moderator surface. Again, this effect is not believed to be significant.

Although the moderator void coefficient is relatively large, any voids should be confined to small local areas and as such cannot represent dangerous amounts of reactivity.

Only a weak coupling exists between the temperature and pressure of the primary coolant and the reactivity. Measurement in the PCTR of complete loss of D_2O coolant was found to result in a 12 mk increase in k_∞ for an all UO_2 loading. Taking account of the increased neutron leakage results in a net reactivity gain of only about 4 mk and even less with a spike enrichment loading. Thus, formation of voids in the primary coolant due to pressure or temperature fluctuation will not result in any large changes in reactivity.

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(17) Nelson, H. K. Dynamic Control Characteristics of the Plutonium Recycle Test Reactor, HW-58706 January 2, 1959.

(18) Peterson, R. E. and J. Muraoka. PRTR Total Energy Distribution Calculations, HW-61346. July 31, 1959.

2. Neutron Kinetics

The transient response of the PRTR to sudden changes in reactivity is governed by the time dependent neutron density. This in turn is dependent upon the form of the reactivity disturbance, the delayed neutron characteristics, the fuel temperature and other smaller effects. The analytical formulation of this problem is given in Appendix C. Solutions for a range of reactivity inputs corresponding to various errors and malfunctions were obtained by means of analog computation. Also, the response of the PRTR to "ramp" reactivity disturbances was investigated. A ramp input assumes the linear introduction of reactivity, Δk, with time. Of particular interest in these calculations is the termination of an excursion by the Doppler coefficient of the fuel. Extremely rapid increases in neutron density and, hence, power are experienced when Δk is much greater than the total delayed neutron fraction, β. However, as has been pointed out, nearly all the energy released is available for heating of the UO_2 and the Doppler coefficient overrides the disturbance and brings the power back to normal rapidly.

Results for a ramp disturbance of 10 mk/sec for two seconds duration and for two different values of the resonance integral coefficient are shown in Figure 39. The greater energy release as well as higher fuel temperatures accompanying the lower coefficient are readily discernible. Termination of the excursion occurs approximately 0.25 sec. sooner with the higher value. Although this disturbance is relatively severe compared to conceivable reactivity insertions, it illustrates the important role played by the Doppler effect.

C. Nuclear Excursions

1. Procedural Errors

Historically, most inadvertent nuclear excursions have resulted from personnel errors coupled with equipment failure. The best defense against such errors is to minimize the dependency of reactor safety on

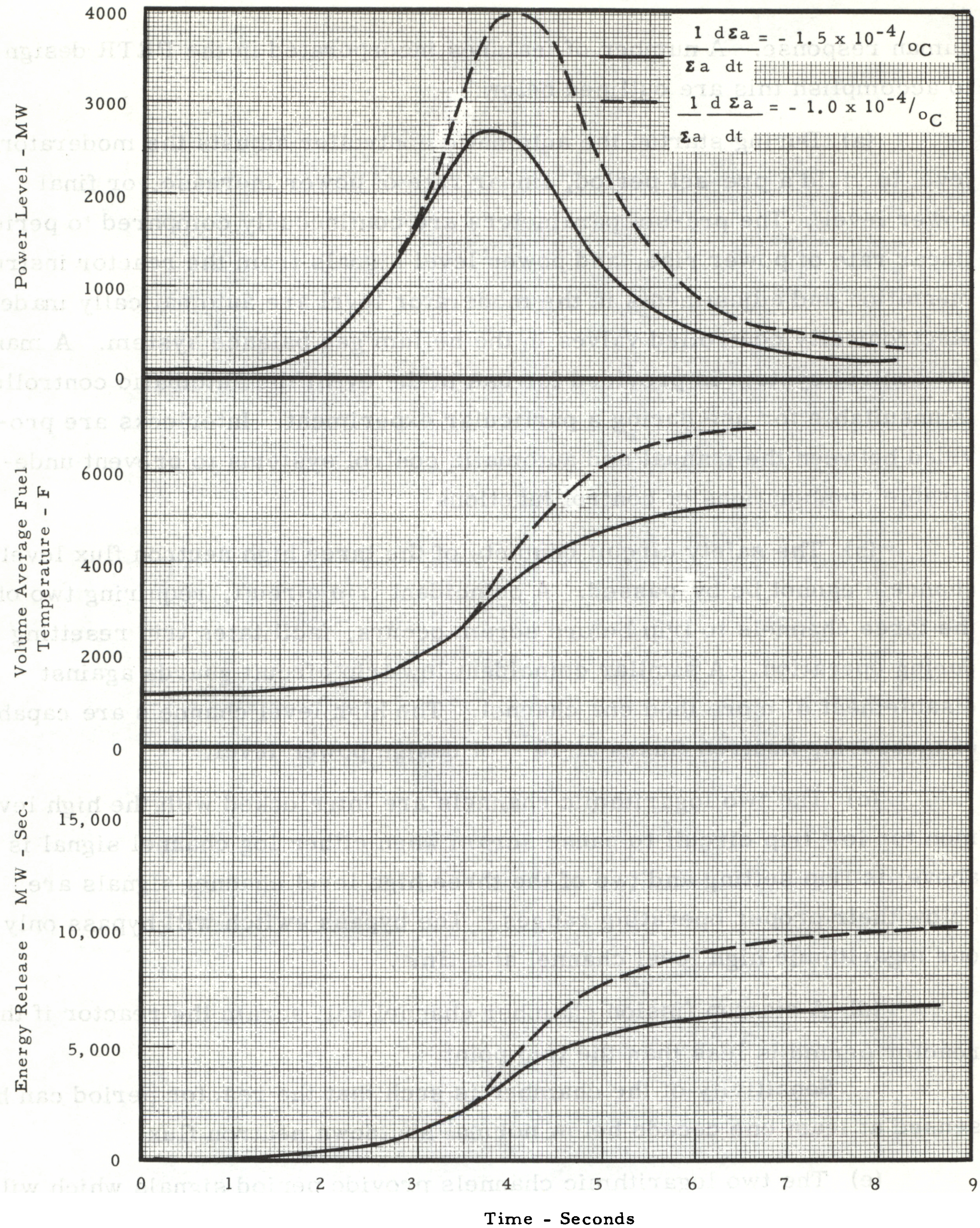

FIGURE 39

Response to A 2 Second, 10mk/sec Ramp

human response. A number of features incorporated in the PRTR design to accomplish this are outlined below:

(a) During startup the automatic controller adjusts the moderator level to yield a pre-set period, linear rate of power increase, or final power level. The pre-set parameters are continuously compared to period, linear rate of power rise, and power level signals from the reactor instrumentation and adjustments in the moderator level are automatically made by electronically controlled valves in the helium gas balance system. A manual control system is provided for use in the event the automatic controller is unsuitable for use during a particular experiment. Interlocks are provided between the manual and automatic control systems to prevent undesirable combinations of control functions.

(b) The safety circuit contacts of the three high neutron flux level channels cannot be by-passed. A coincident trip circuit, requiring two of the three channels to trip before scram occurs, facilitates trip resetting during operation. A similar coincident low trip circuit guards against disablement of more than one channel. The high level channels are capable of measuring neutron fluxes at 10^{-4} of design power level.

(c) The two logarithmic channels are interlocked with the high level monitor low trip circuit to cause scram when either log channel signal is above its trip setting and two of the three high level channel signals are below their proper operating ranges. The bypass switch will bypass only one logarithmic high level channel at a time.

(d) A startup fission chamber channel will scram the reactor if the reactor period is less than the trip point.

Sensitivity of the chamber is such that the reactor period can be sensed at least one decade below normal shutdown neutron flux.

(e) The two logarithmic channels provide period signals which will scram the reactor if either period signal is less than the period trip point. These channels are capable of measuring periods at 10^{-6} of design power.

2. Startup Accident

In the PRTR, the maximum rate of reactivity insertion is fixed by the dynamic capabilities of the gas balance moderator level control system and the maximum rate of movement and worth of the shim controls. In starting up the reactor, the moderator level is normally raised by the automatic control system or manually by means of the manual control system. The shim system may be adjusted to provide sufficient reactivity to compensate for xenon buildup or to control the flux distribution. For startup accident analysis two types of accidents were considered:

Case A; The moderator level being increased continuously at the maximum possible rate with simultaneous withdrawal of the shim controls, and

Case B; The moderator level being continuously increased at the maximum possible rate after all shim controls are removed.

The latter case is by far the most severe because of the lower initial critical moderator level and the associated increased rate of reactivity addition (shown on Figure 13, page 34). It is also by far the more incredible because it requires four independent errors or failures: failure of the automatic control system or personnel error during a manually controlled startup; failure of the period instrumentation to respond to a fast period; procedural error of removing all shim controls prior to a startup; and failure to set high neutron level trip for startup, 0.15 P. Excess reactivity during these accidents is shown on Figure 40. For each of these cases, it was assumed that reactor scram was initiated by the high flux level instrumentation at levels of 0.15, 1.15 and 1.3 times P (P = design reactor power level). A 0.2 second delay time for relay actuation was assumed before the actual moderator dump began. For each case analogue computer runs were made with several values for the reactor loading, UO_2 thermal conductivity, and UO_2 temperature coefficient. The transient curves in this section were based on the best estimate values of

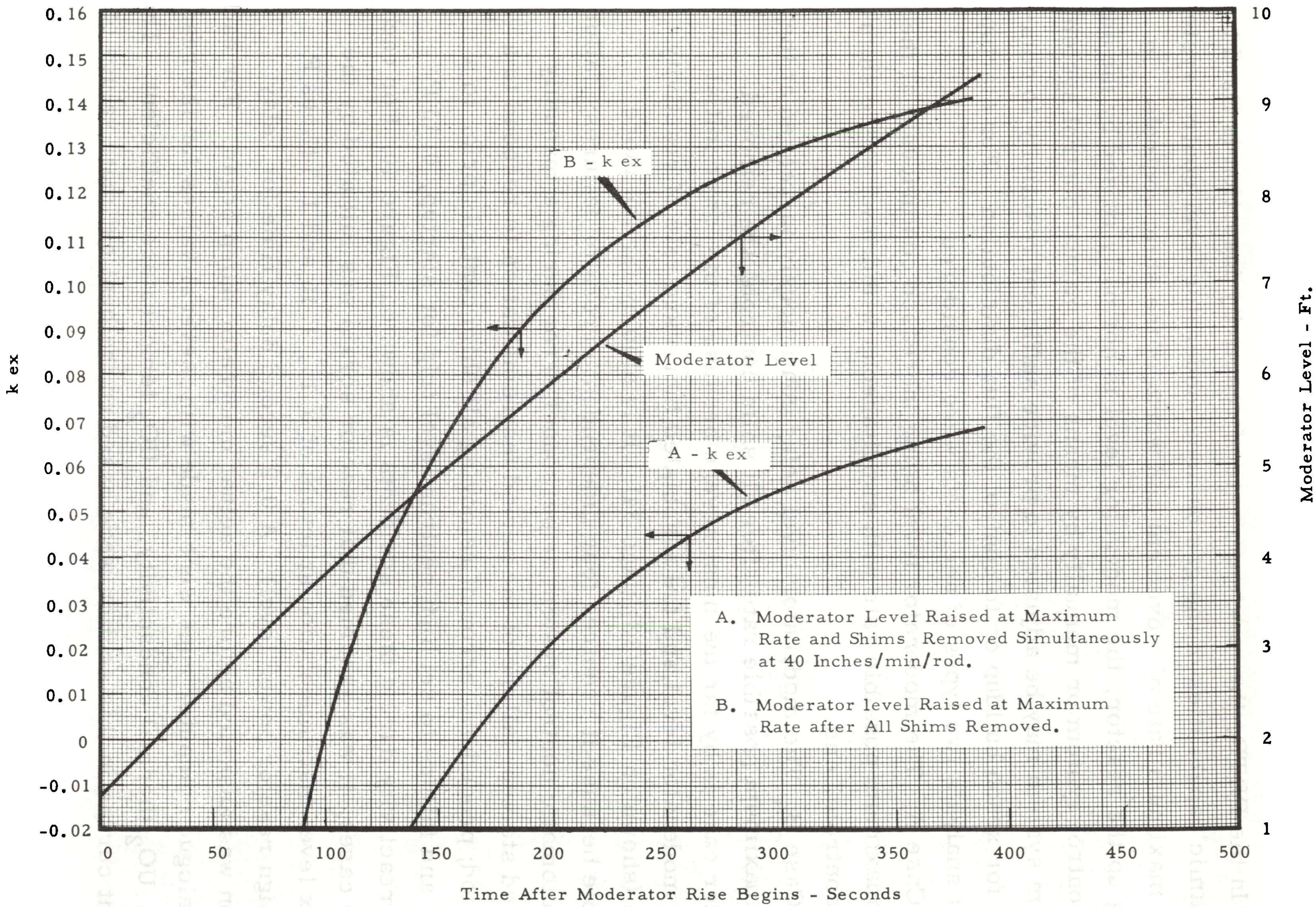

FIGURE 40

Excess Reactivity During Startup Accidents the parameters listed in Appendix C. Mark I fuel elements were used as the basis for the analysis. The transients are more severe for this type of fuel element than they would be for Mark II fuel elements because the higher UO_2 temperatures at a given power level and the greater weight of UO_2 in the latter causes the Doppler effect to be greater and to become effective sooner during excursions. A neutron source of 10^3 neutrons/(sq cm) (sec) was assumed. This is substantially less than the 10^7 neutrons/(sq cm) (sec) shutdown source level expected from photo-neutrons. Results are therefore pessimistic because the assumed low source flux permits a larger amount of reactivity to be inserted before the high flux trips are reached. The reactor period at the high flux trip point is shorter than would be the case with higher source levels and more overshoot above the high flux trips results.

The distributions of temperatures within the fuel elements were obtained by calculations performed on an IBM-709 digital computer using the power versus time data obtained from the analogue computer as input data.

Figure 41 shows temperature and power transients for the more credible startup accident, Case A. During a startup the high flux trip points are normally set at a maximum of 0.15 P. As shown, the maximum fuel surface temperatures and core temperatures are quite moderate even if the trip points are erroneously set for full power range values (1.3 P). Maximum heat fluxes for this case are 145,000 and 297,000 Btu/(hr) (sq ft) for the UO_2 and Pu-Al fuel elements, respectively. Laboratory experiments at heat fluxes of 330,000 Btu/(hr) (sq ft) have been performed without any indications of boiling burnout using an electrically heated cluster element test section under prototypic conditions.

The pessimistic startup accident, Case B, transients are shown on Figure 42. With the high flux trip settings at 0.15 P, fuel temperatures and heat fluxes are adequately limited. With high flux trips erroneously set for 1.3 P, the UO_2 fuel temperatures and heat fluxes are quite moderate. However, the Pu-Al core temperatures substantially exceed the melting point (1220 F). The maximum heat flux, 1,260,000 Btu/(hr) (sq ft), is below the range of predicted boiling burnout. In this highly pessimistic case, some Pu-Al fuel element cores would melt, but the time during which the Pu-Al would be molten is too short for Al-Zr diffusion to result in rupture of the Zircaloy jackets.

3. Control System and Instrument Malfunction

Certain not inconceivable failures of automatic controller components can cause the control valve to close completely resulting in reactivity increases at the maximum possible rate provided by the helium pump. Such failures are expected to be very infrequent, if indeed they ever occur. However, the high level and period channels provide protection for the reactor in the event that this type of failure occurs. Figures 43 and 44 show the results of analogue simulation of this type of failure at power levels of 10 and 70 MW using the same assumptions as used in the startup accident analyses in the previous section. As is seen, the high level flux trips adequately limit fuel element temperatures and surface heat fluxes to quite moderate values. Period trips are even more effective in limiting the severity of the excursion. The nuclear excursion will thus be very minor unless all high level circuit trips and all period trips should be inoperative at the same time, an essentially impossible situation.

4. Loading Error

Charge-discharge in the PRTR will be accomplished segmentally, charging 1 to 4 new fuel elements at each outage. It is conceivable that a loading error could be made in which too many spike enrichment fuel elements are loaded into the reactor. Once the reactor begins operation, a maximum of 15 fuel channels will be reloaded per quarter until the use of uranium-plutonium oxide fuel begins in 1963.[19] Even if a schedule change required

(19) Bradley, J. G. A Proposed Schedule for Plutonium Recycle Program Procurement and Operation, HW-61627. August 28, 1959.

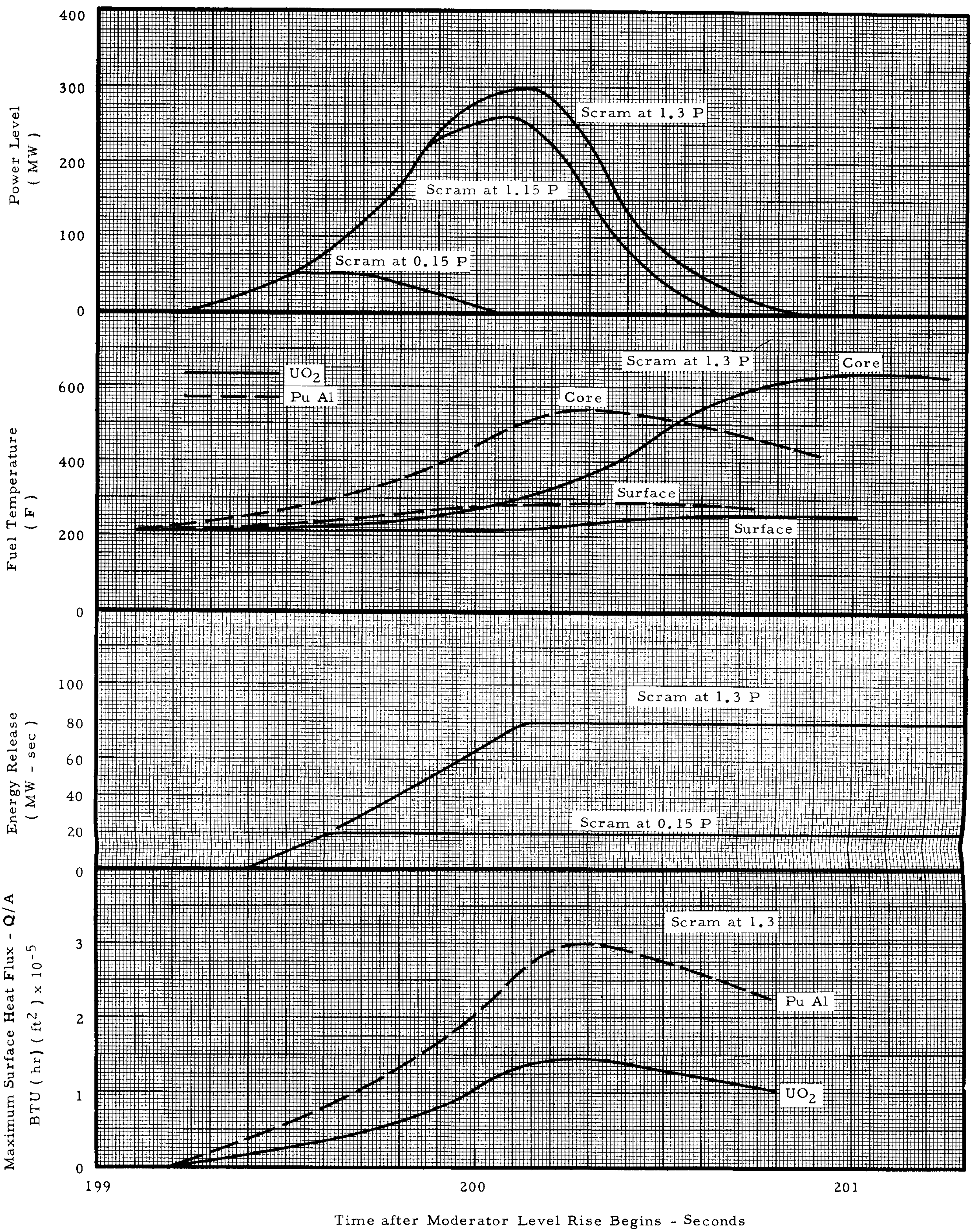

FIGURE 41
Startup Accident - Case A

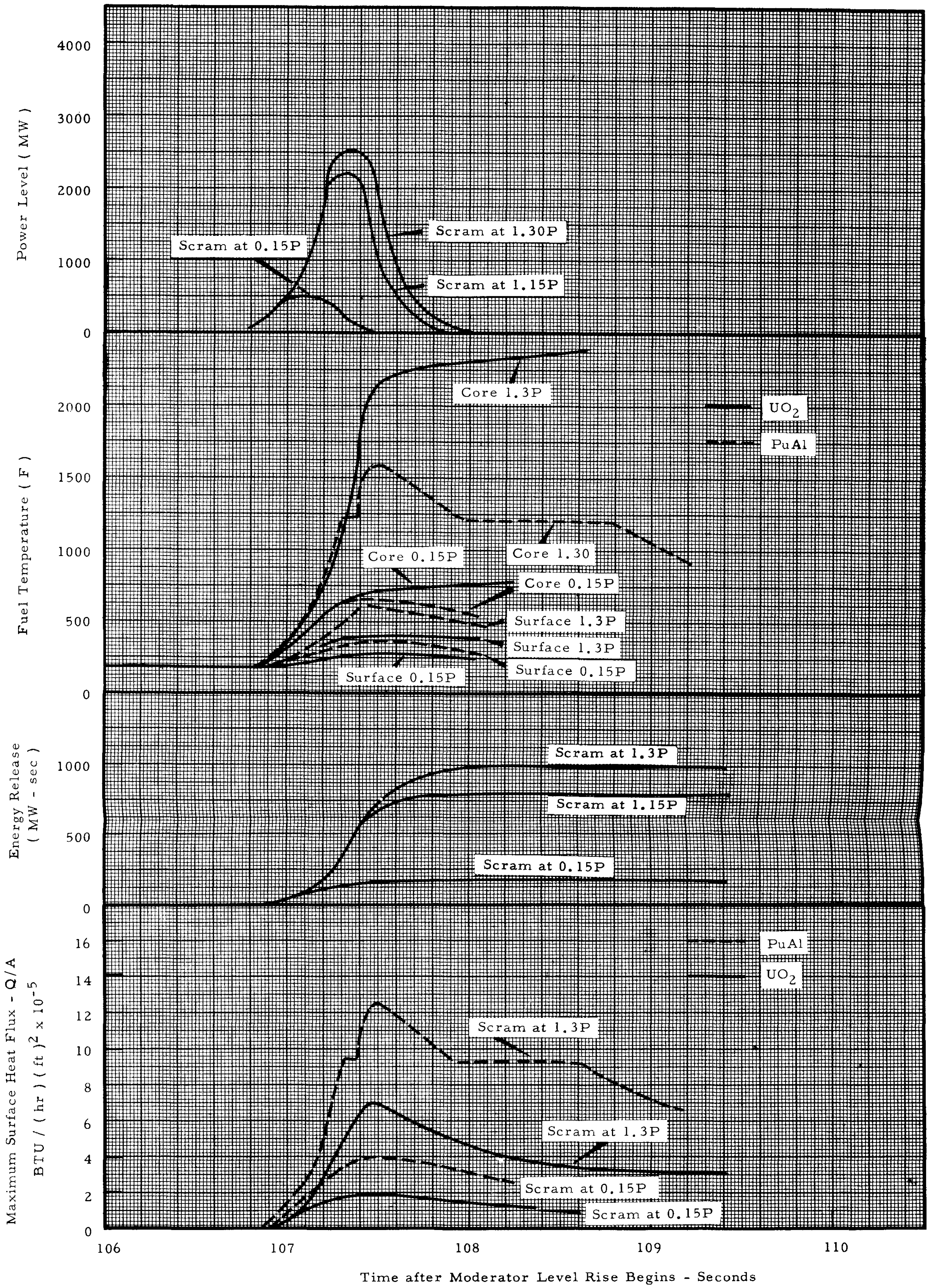

FIGURE 42

Startup Accident - Case B

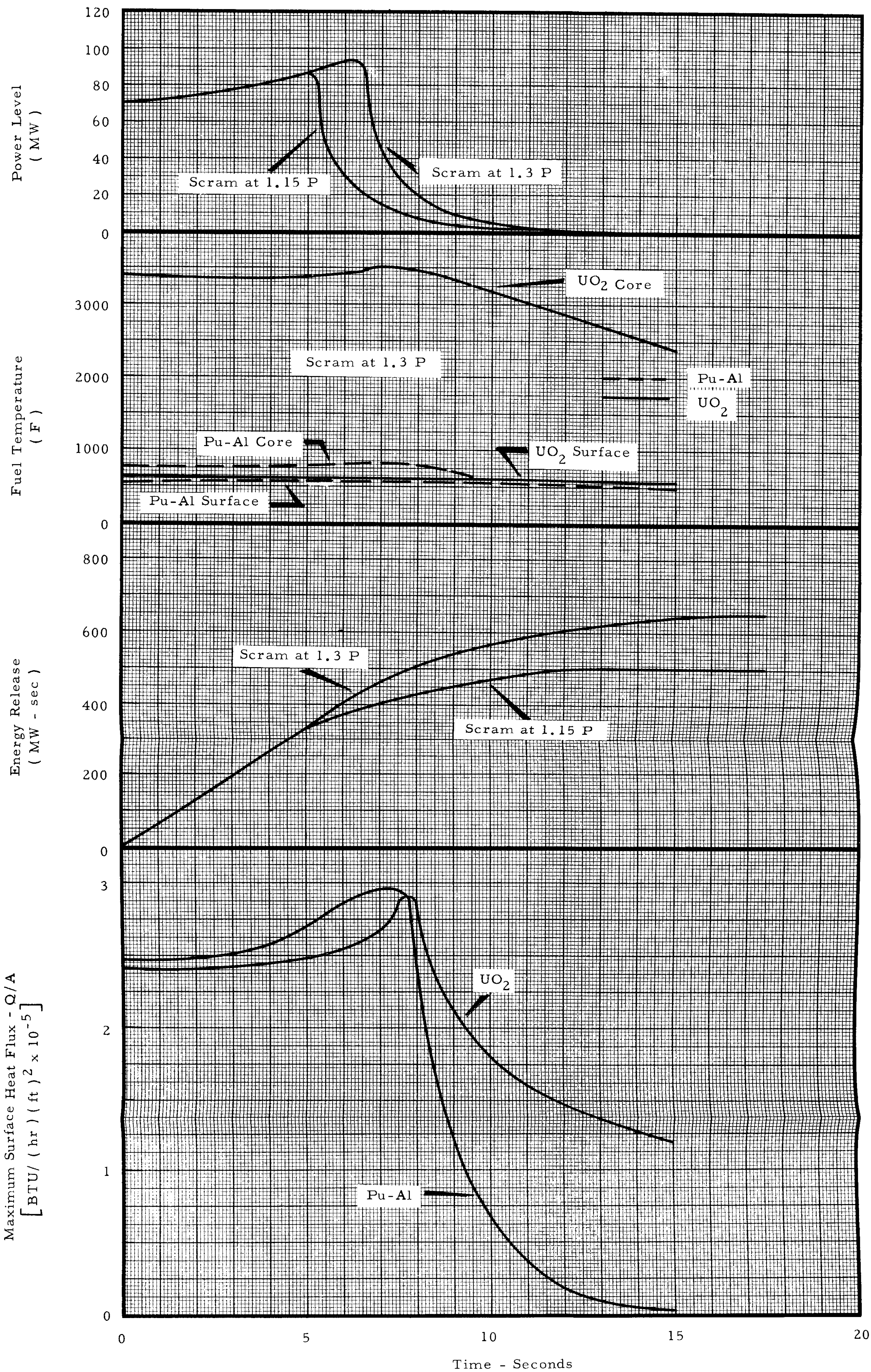

FIGURE 43

Controller Malfunction at 70 MW

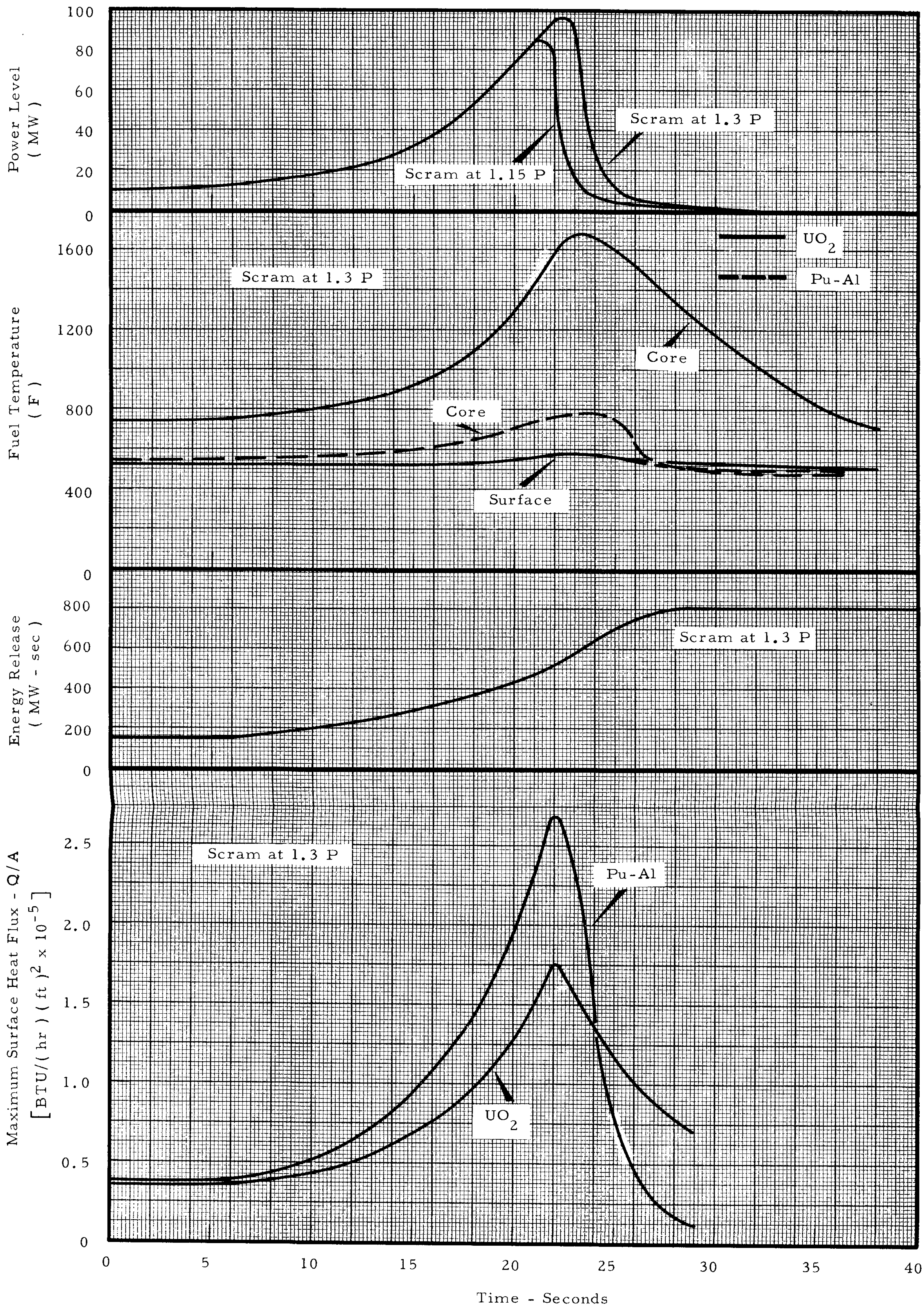

FIGURE 44

Controller Malfunction at 10 MW

the charging of 7 or 8 new fuel elements at a single outage, the mischarging of this number of fuel elements (replacement of eight UO_2 elements with Pu-Al elements) could not result in a disastrous startup accident.

The mischarging of 8 Pu-Al fuel elements would reduce the delayed neutron fraction and negative fuel temperature coefficient. These effects would change the dynamic behavior of the reactor, tending to increase the severity of nuclear excursions. Analogue computations of startup accidents assuming that such a loading error had occurred were performed. The results showed that internal fuel melting or burnout would not occur in the more credible startup accident, (maximum nuclear power 1100 MW and maximum Pu-Al core temperature 750 F). This accident would require, in addition to the loading error, malfunction of the automatic controller or personnel error during manual control, failure of the period instrumentation to detect a fast period or to actuate the scram system, personnel error in having the high neutron flux trips set for the operating range instead of the startup range, and removal of the shim controls during the moderator level raise.

Errors in which larger numbers of Pu-Al fuel elements are erroneously charged to the reactor are not considered credible because of the following factors:

1. Only a limited number of Pu-Al spike enrichment fuel elements will be available. Gross misloading would require recharging elements from irradiated fuel storage and special efforts to obtain new fuel elements from the fuel element fabrication facility. It is difficult to visualize such efforts being taken unless the loading is deliberate rather than erroneous.

2. The identity of plutonium fuel elements will be maintained by the use of specially shaped fuel hangers to which only Pu-Al elements can be attached and UO_2 fuel hangers of a different shape to which Pu-Al elements cannot be attached.

3. Startup procedures will require that all Pu-Al fuel elements be accounted for before reactor startup can be initiated. This includes elements in reactor, in irradiated fuel storage, in unirradiated fuel storage, and in the fuel examination facility. Written approval of the inventory by ranking personnel will be required before reactor startup can be initiated.

5. Experiment Failure

It is currently planned to install several test loops in the PRTR to enhance the experimental flexibility of the reactor facility. Separate hazards analyses will be performed for each loop when the designs are sufficiently advanced. One such analysis has been performed.[20] To approximately evaluate the effects of the various loops on reactor nuclear safety, analogue computations of the reactor behavior on loss of coolant or vaporization of test assemblies were performed. The loops being considered are:

(a) A gas cooled loop for the irradiation of graphite or fuel impregnated graphite.

(b) A high pressure water cooled loop to determine the feasibility of a water cooled reactor operating in the supercritical steam range.

(c) A water cooled loop for the irradiation of high risk or purposely defected experimental fuel elements.

Preliminary analysis of these loops [21] indicated that the loss of coolant or vaporization of the in-reactor portion of a loop could cause up to a 5 mk reactivity addition.

It is emphasized that these loops are being designed for a coolant reliability equal to or greater than that of the reactor proper. Figure 45 shows the analogue results of a 5 mk/sec reactivity addition during a one second period without reactor scram. Even for this pessimistic case, a maximum reactor power level of only 93 MW is attained and fuel temperatures are well within tolerable values. The maximum heat flux is

(20) Wittenbrock, N. G. PRTR Gas-cooled Loop, Hazards Survey of Preliminary Scope Design, HW-59338. April 29, 1959.

(21) Peterson, R. E. Effects of In-reactor Test Loops on PRTR Operation and Program, HW-59391. March 18, 1959.

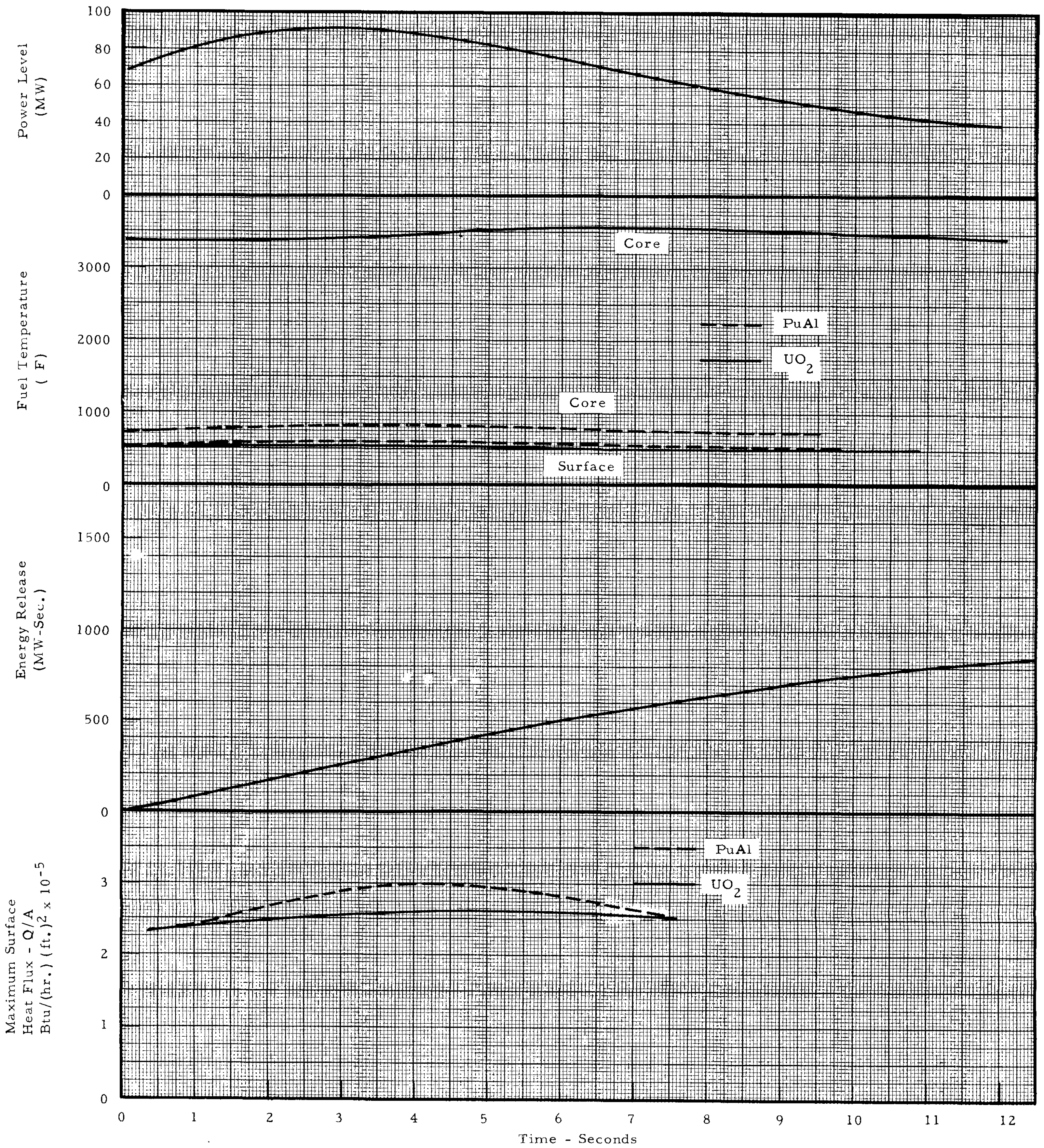

FIGURE 45
Experiment Failure

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300, 000 Btu/(hr) (sq ft). As stated previously, laboratory experiments under prototypic conditions have shown no indication of boiling burnout at a heat flux of 330, 000 Btu/(hr) (sq ft). Coolant loss in any of the loops will immediately initiate a reactor scram. Any possibility of an excursion of even this moderate magnitude is therefore extremely unlikely.

6. Shim System Failure

A description of the shim control system is given in Section III-A-3-b and is not repeated here. Each motor drives its corresponding lead screws by sprocket and steel chain through a jack shaft and from the jack shaft to the lead screws with sprockets and stud chain. In the event of chain breakage, the system will either continue to operate or jam. A chain break will not allow the lead screws to turn independently of the motors. All sprockets and shaft hubs are light press fitted into the large hubs and then pinned with a spring type roll-pin. A pin can be removed only if it is aligned with its installation clearance hole. If any pin were to start out, it would jam the mechanism. In the event that a lead screw became free of its upper support bearing, the screw would drop into a tapered hole below the lower bearing and jam itself, preventing further rotation. The ball nut which operates on each lead screw is equipped with two separate circuits of balls so that loss of one set of balls will not allow the rod to fall. Tests of the free fall of each rod of a prototype assembly, with the rod held by the lead screw and ball nut, were conducted. Two of the rods would not fall under their own weight because of irregularities in the lead screws. The third rod fell completely to the bottom. The maximum speed of fall was 30 ft/min. Using the rod drop rate of 30 ft/min and a strength of 5.3 mk per rod, the reactivity transient caused by the simultaneous dropping of a coupled pair of rods from their maximum worth position was computed for an initial equilibrium power level of 70 MW. The reactivity disturbance is shown on Figure 38, page 140. The results of the analogue computation are shown on Figure 46. As indicated, the high neutron flux channels limit the extent of the

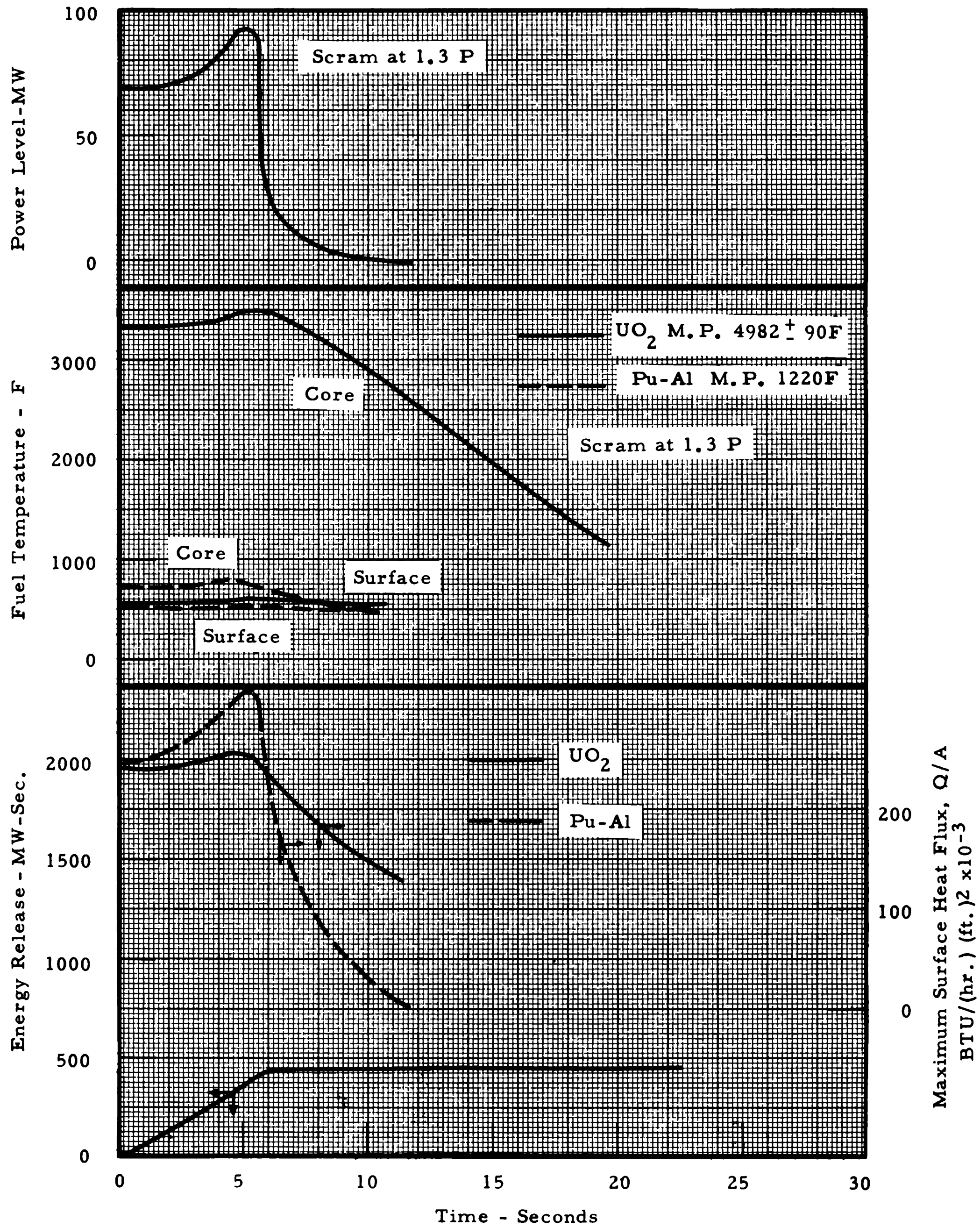

FIGURE 46

Failure of Two Shim Rods

excursion adequately with a maximum power of 94 MW being attained and maximum UO_2 fuel core and surface temperatures of 3500 F and 630 F, respectively, being attained. The maximum Pu-Al fuel temperatures are safely below the alloy melting point of 1220 F.

7. Fuel Element Failure

Violent failure of plutonium fuel elements, resulting in a rapid distribution of plutonium throughout the coolant, and occurring in several elements simultaneously, could initiate a serious power transient. The greatest hazard would be in those elements having the highest ratio of plutonium content to power generation, i. e., the highest self shielding factor. Such elements are not currently under serious consideration for use in the reactor. Plutonium-bearing fuel elements planned for use in the PRTR are segmented into 19 isolated fuel rod sections. Failure of an element will be partial and probably gradual. Since many feet of fuel element would have to fail nearly simultaneously to cause an excursion, it is considered impossible that the failure of Mark I spike enrichment fuel elements could cause an excursion.

8. Coolant System Failure

Failure of the coolant system itself constitutes a serious incident. With heavy water of reasonable purity (97 to 100 per cent) as coolant, no severe power excursion can result from loss of coolant. Process and shroud tube rupture, by increasing calandria pressure, would tend to initiate a dump which would prevent any net increase of heavy water content in the core. Ruptures in the main coolant piping would result in rapid loss of coolant from the process tubes. In most cases this loss would decrease reactivity; for some fuel configurations, however, some increase in reactivity could occur. (See Section III-B-3-b) Even in the most disadvantageous configurations, the reactivity increase would be slight enough that the reactor would be shut down effectively by the safety system, with no severe excursion occurring.

9. Moderator and Gas System Failure

Flooding of the gas space in the reactor due to a leak or rupture in a single calandria shroud tube would be confined to that particular process channel and little effect on reactivity would result. Simultaneous flooding of all channels would produce a reactivity increase (at full level) estimated at 19 mk. Such leaks would not result in flooding if the reactor were in full power operation, since vaporization would occur as soon as the moderator came in contact with the heated process tube. Therefore the only possibility of an excursion resulting from this effect would be simultaneous and drastic rupture of most of the calandria shroud tubes during cold startup conditions. The resulting excursion would be limited to that produced by addition of 1086 gpm of moderator to the core if the entire moderator inflow is used to fill the gas annuli.. At a 200 cm level this would produce a maximum ramp addition of reactivity of about 5 mk/sec. A ramp of this size should be terminated by the safety circuit before excessively high power levels are reached.

The dynamic gas-balance system and its associated instrumentation and controls are of fail-safe type. Short of blockage of the dump weir (7 ft diameter annular slot 12 inches wide), it is very difficult to envision a sequence of events in which the normal dump mechanism would fail to respond to a failure in the moderator or gas systems.

Transient pressure unbalances in the calandria gas balance system will, in general, tend to shut down the reactor. An increase in gas pressure within the calandria, by failure of reactor pressure piping, overheating or accumulation and ignition of an explosive gas mixture, will force the moderator level down. In a severe pressure surge, such as produced by an explosion, the calandria would probably rupture. Rupture of the top gas plenum would have essentially no effect other than aiding removal of the excess pressure. Rupture of the calandria walls proper would result in mixing of the moderator with the reflector and/or draining of moderator from the calandria into the surrounding gas space. Either action results in either no change or decrease of reactivity. The shroud tubes would tend to collapse rather than rupture in this case; in either event, moderator contacting the hot process tubes would flash into steam, increasing pressure and aiding ejection of moderator from the calandria.

An explosion in the moderator storage tank, if severe enough, would rupture the tank; subsequent loss of pressure would cause the moderator to drain from the calandria. A less severe explosion would immediately force heavy water trapped in the weir into the calandria, raising the moderator level by about 1.2 inches, probably followed by violent bubbling of gas into the calandria through the weir. These disturbances would provide reactivity transients sufficient to trip the safety system, but not strong enough to initiate a serious excursion when the moderator is at the normal operating level (0.5 mk reactivity addition). Continuous purging of the gas within the calandria and the storage tank to the helium purification unit makes the possibility of a deuterium-oxygen explosion extremely remote, even if the blanketing effect of the helium gas is ignored.

D. Coolant System Failures

1. Electrical Power Failure

Failure of BPA system power will result in a scram of the reactor. In event of such failure, emergency power will be available from an emergency diesel-powered generator in approximately 15 seconds. This emergency generator is rated at 300 KW, adequate for supplying the 220 KW PRTR emergency load.

Energy for the following services is supplied by the emergency electrical distribution system from the diesel generator.

a. Control instrumentation*
b. Emergency lighting*
c. Compressed air*
d. Process water pumping*

* Automatic switch

e. Boiler feedwater pumping**
f. Primary coolant circulation*
g. Moderator circulation*
h. Process area air conditioning**
i. Helium processing**
j. Primary coolant makeup**

A bank of storage batteries provides DC power for a portion of the emergency lights, all switchgear control power, and other critical circuits and items of equipment. DC powered circuits include the safety, containment, and annunciator circuits.

In the event that both BPA power and emergency power fail, the primary coolant is circulated by pump flywheel energy and then by natural convection. During convection circulation of the primary coolant, boiler feed water would be supplied by the emergency diesel well pump. Figure 47 shows the relative power and primary coolant flow decay with time and the relative adequacy of primary coolant flow in the "hot tube". (Primary coolant flow adequacy of 1.0 is defined as the flow required to give the saturation temperature at the tube outlet.) As indicated, the coolant flow adequacy does not drop below 1.0; no coolant bulk boiling will occur during such a transient.

2. Mechanical Failures

a. Large Header Rupture

An evaluation of the temperature excursion and its possible consequences arising from loss of coolant through postulated ruptures in the PRTR primary cooling system has been made.[22] As the basis for the evaluation, a series of computations were made based on incremental heat and mass balance for sections of Zircaloy-clad UO_2 and Pu-Al fuel elements.

* Automatic switch
** Manual switch

(22) Lemmon, A. W., C. A. Alexander, L. E. Hulbert and R. B. Filbert. Core Temperature Excursions Following a Piping Failure in the Plutonium Recycle Test Reactor, BMI-1356. July 6, 1959.

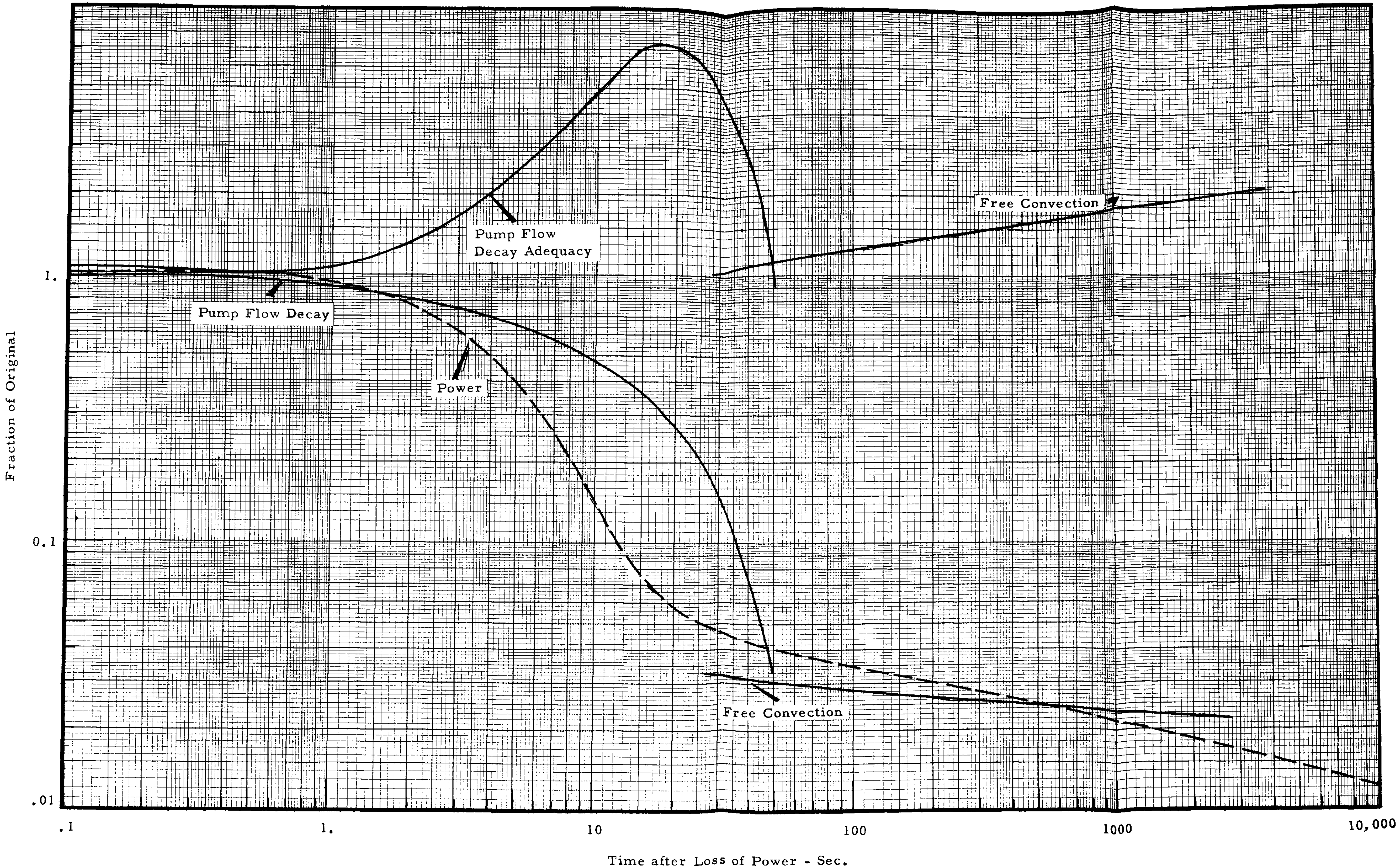

FIGURE 47

Relative Flow, Power, and Adequacy
After Complete Electric Power Failure

Solutions to each problem defined by the postulated break size and its location were obtained by finite difference approximations performed by an IBM-653 digital computer. In the main loop piping, the postulated ruptures considered were:

(1) a complete parting of the 14-inch diameter outlet pipe near the upper ringheader so that coolant would be lost from both broken ends, and

(2) a rupture equivalent to a 12-inch diameter hole in the primary loop piping adjacent to the upper ringheader.

Case 1 is generally considered incredible; however, with the emergency coolant supply system provided, fuel damage is avoided even in this case. Failures in the inlet piping were not analyzed because they represent less severe accidents than outlet piping failures. This is because the venturis in the inlet jumpers will restrict flow when the saturation pressure is developed in the venturi throats. If the failure is in the inlet piping, this choking action will limit the discharge rate through the reactor leg of the system and provide cooling for a longer period of time. Where assumptions as to actual conditions in the reactor were necessary, the values chosen were conservative. Although light water can be injected by the boiler feed pumps when the primary system pressure drops to 500 psig, the calculations were based on use of the 100 psig head backup pump as an added measure of conservatism.

Case 1

For Case 1, defined as the complete parting of the 14-inch outlet pipe, the sequence of events postulated is: a rapid pressure decay in the reactor during which most of the primary coolant would be lost from the system through the two ends of the ruptured pipe (blowdown phase); a period of time following in which the fuel elements would be covered by stagnant steam and no heat would be lost from the elements (adiabatic phase); and, finally, the period subsequent to the time emergency back-up coolant would reach the fuel element (injection phase). It was assumed that reactor scram was initiated within 0. 1 sec after the rupture occurred.

The rate of coolant loss obtained from blowdown calculations performed by Ambrose [23] is shown on Figure 48. Figure 49 shows the fuel temperature transient at the point of maximum temperature. During the blowdown phase all fuel temperatures decrease because of the high coolant flow rate through the process tubes and the low rate of heat generation. The defined adiabatic phase would begin at about 24 seconds after the postulated rupture. At this point, the pressure having decreased to less than 100 psig, emergency coolant could be injected for the first time. Without automatic emergency coolant injection, the critical element is the Pu-Al element; it would reach the melting point of aluminum (1220 F) at 219 seconds after the postulated incident. The UO_2 elements would not reach the melting point of Zirconium (3314 F) before 830 to 1000 seconds. Thus, to prevent any melting of the Pu-Al elements, cooling must be provided before 219 seconds have passed. With automatic injection from the emergency well pump, the boiler feed pumps, or the process water pumps, a minimum of 750 gpm of light water coolant can be injected into the ringheaders at a minimum pump head of 100 psig, beginning at 24 seconds after the rupture. With injection beginning at this time, the fuel elements would be covered with liquid coolant 120 seconds after the incident. The computed temperatures of a UO_2 Mark I fuel element after injection has been initiated are shown in Figure 49. Also shown are the temperatures for injection starting at later times so that the coolant reaches the bottom of the fuel elements at 10 minutes and 15 minutes after the incident. In these two cases, the temperature rises above the adiabatic curve upon injection due to the heat evolved from the reaction of water with Zircaloy-2. Although the effects of the heat evolution are readily apparent, in neither case is an appreciable fraction of the Zircaloy-2 reacted.

Case 2

Case 2 was postulated as that of a single hole in the outlet pipe of an area of 0.80 sq ft, equivalent to a 12-inch diameter hole. The blowdown pressure transient for this Case is shown on Figure 50. The pressure

(23) Ambrose, T. W. Water Loss Rate Following a Piping Failure in the Plutonium Recycle Test Reactor, HW-60654. June 8, 1959.

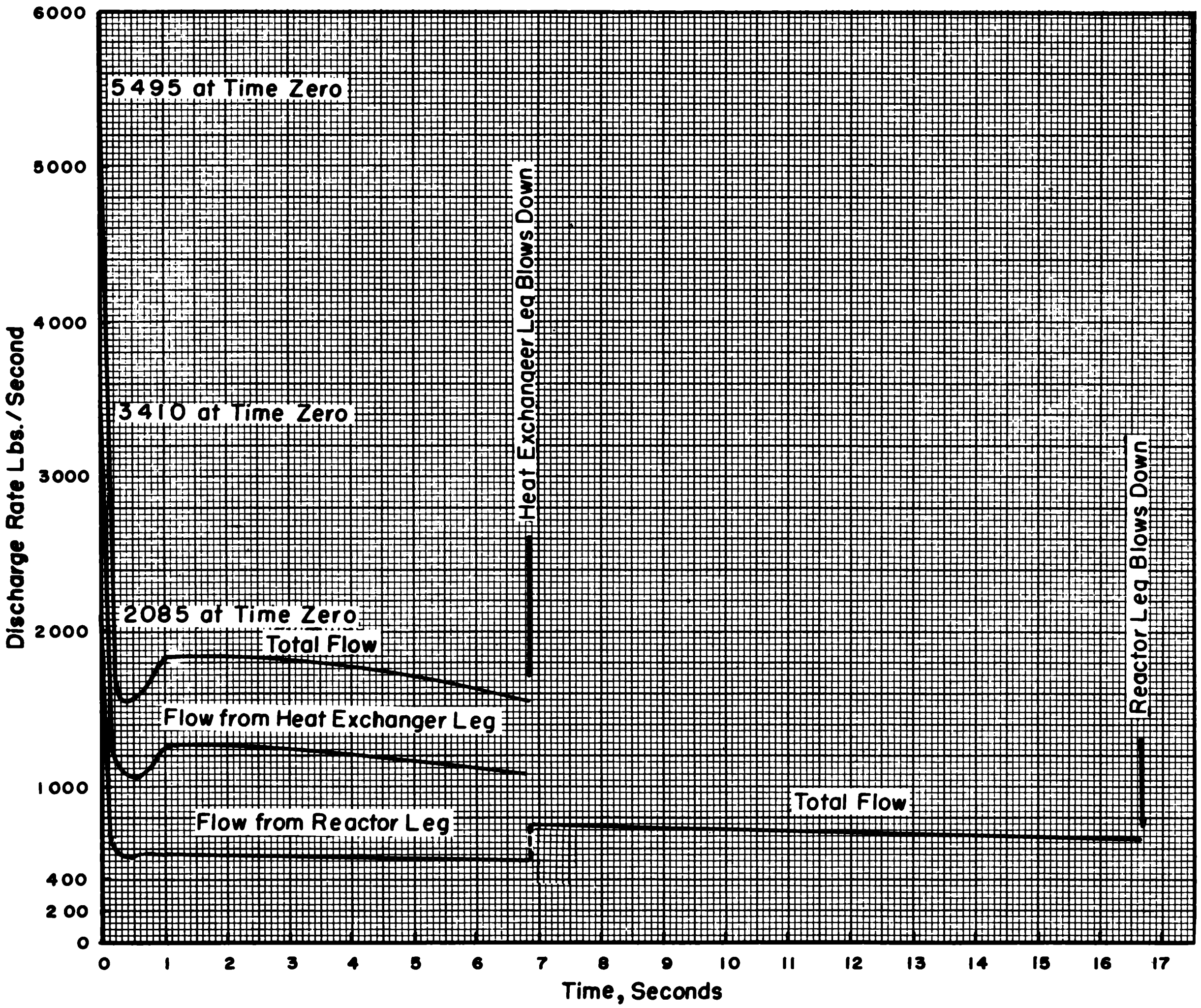

FIGURE 48

Primary Coolant Loss Following Complete Parting of the Top 14-inch Header decay for this case is much slower than for Case 1. Figure 51 shows the fuel temperature of a Mark I UO_2 element during the first part of the blowdown. The adiabatic and injection phases of this case would parallel those of Case 1. However, since the cooling during blowdown is not as effective as in Case 1 for the UO_2 elements, shorter times to melting would be expected. It is estimated that the UO_2 fuel elements would reach the melting point of Zirconium (3314 F) in about 740 seconds for the Mark I UO_2 element. The Pu-Al elements, which have a much higher thermal conductivity and lower heat capacity, would be at the coolant temperature long before the end of the mixed phase portion of the blowdown phase. Times to melting are estimated to be the same for Case 2 as for Case 1 (219 seconds). The delay time from start of injection until the water reached the base of the fuel elements would be approximately the same as in Case 1. The behavior during injection for this case would be the same as for Case 1, except that injection would begin at 36 seconds after the incident rather than 24 seconds.

It is concluded that, even for the case of a complete separation of the 14-inch outlet pipe, the emergency coolant injection system will prevent fuel melting.

b. Process Tube Jumper Rupture

Calculations similar to those previously outlined were carried out for:

(1) the complete parting of one of the small jumpers that connect each process tube to the upper ring header and,

(2) the complete parting of a lower jumper. With these ruptures, discharge would occur through the pipe stub from the ring header and through the stub from the process tube, designated the victim tube.

Figures 52 and 53 show the coolant loss rates for these cases. With the rupture associated with a specific victim tube, both the conditions within the victim tube and the intact tubes in the rest of the reactor must be considered separately in the analysis.

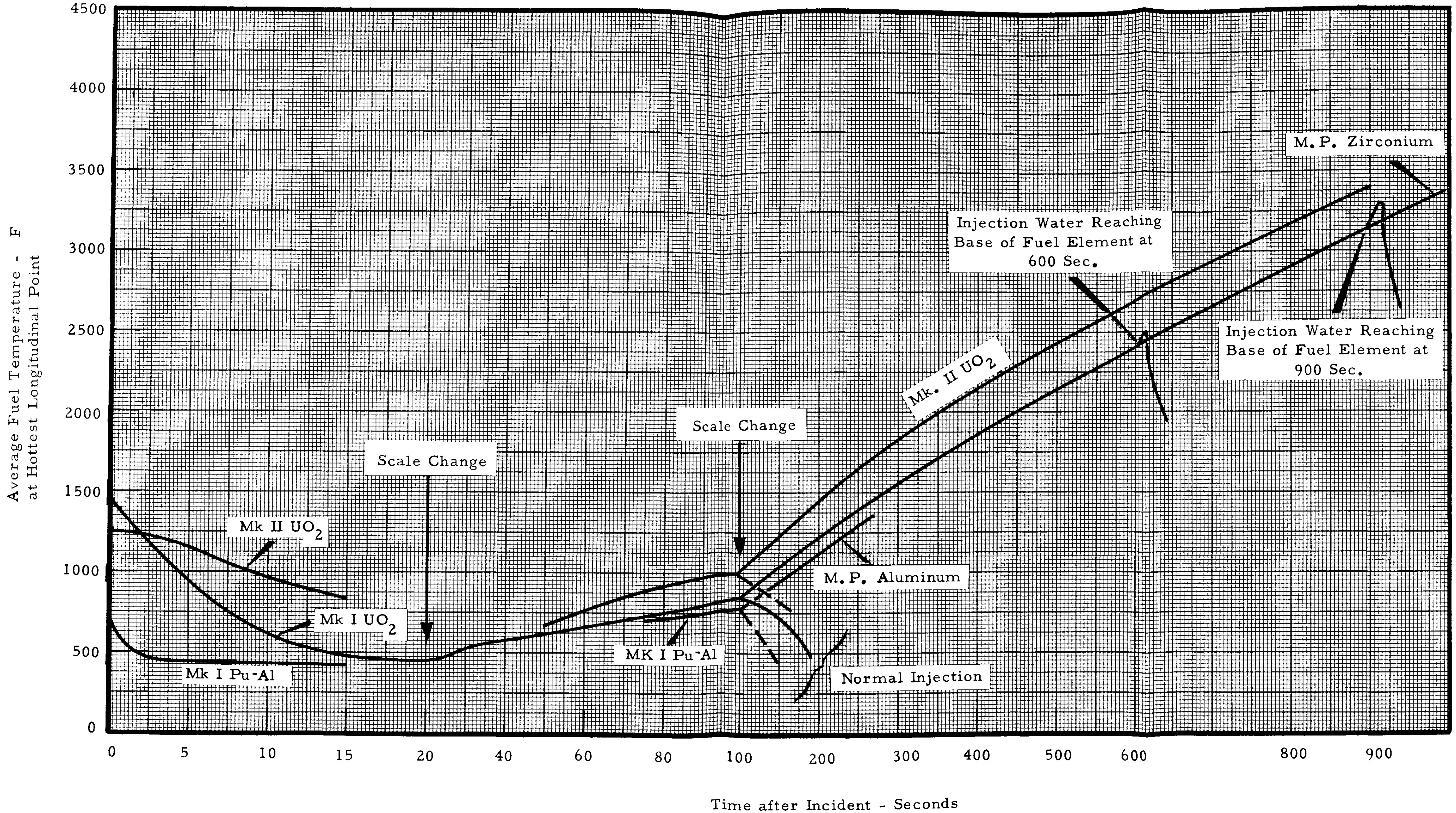

FIGURE 49

Fuel Temperature after Complete Parting
of the top 14-inch Header

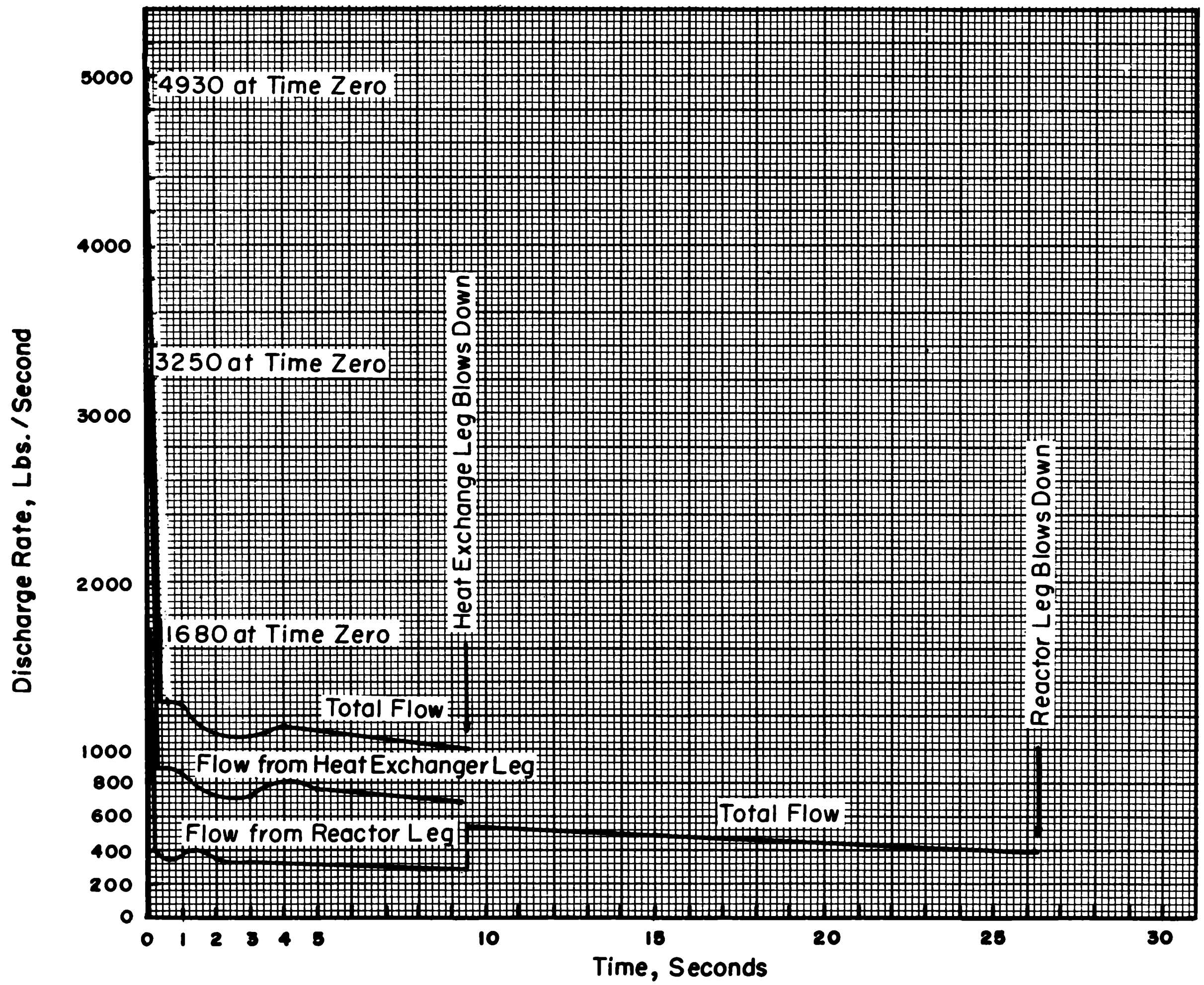

FIGURE 50

Primary Coolant Loss Following Rupture of Top 14-inch Header

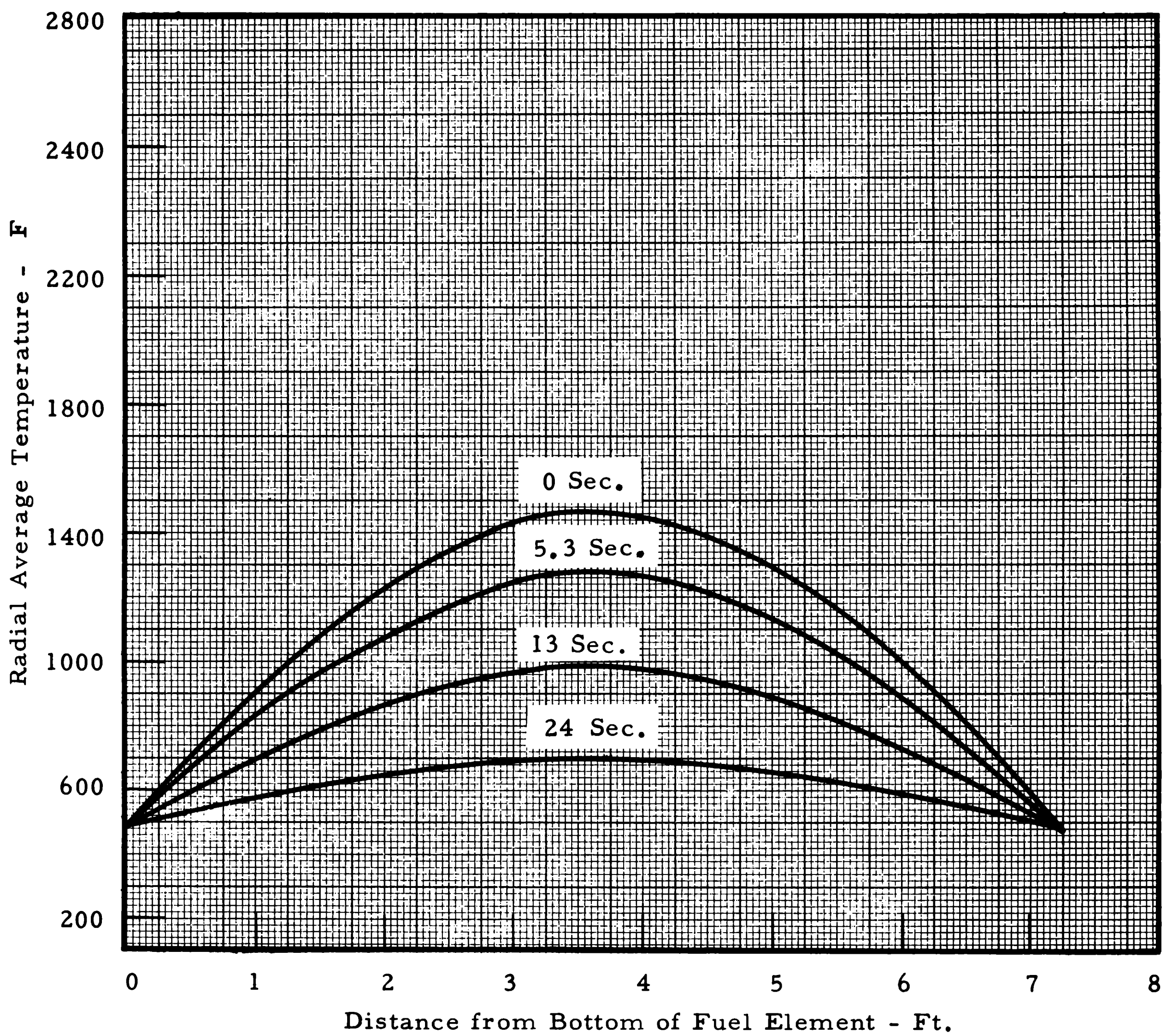

FIGURE 51

Mk I UO_2 Fuel Temperature
after Rupture of Top 14-inch Header

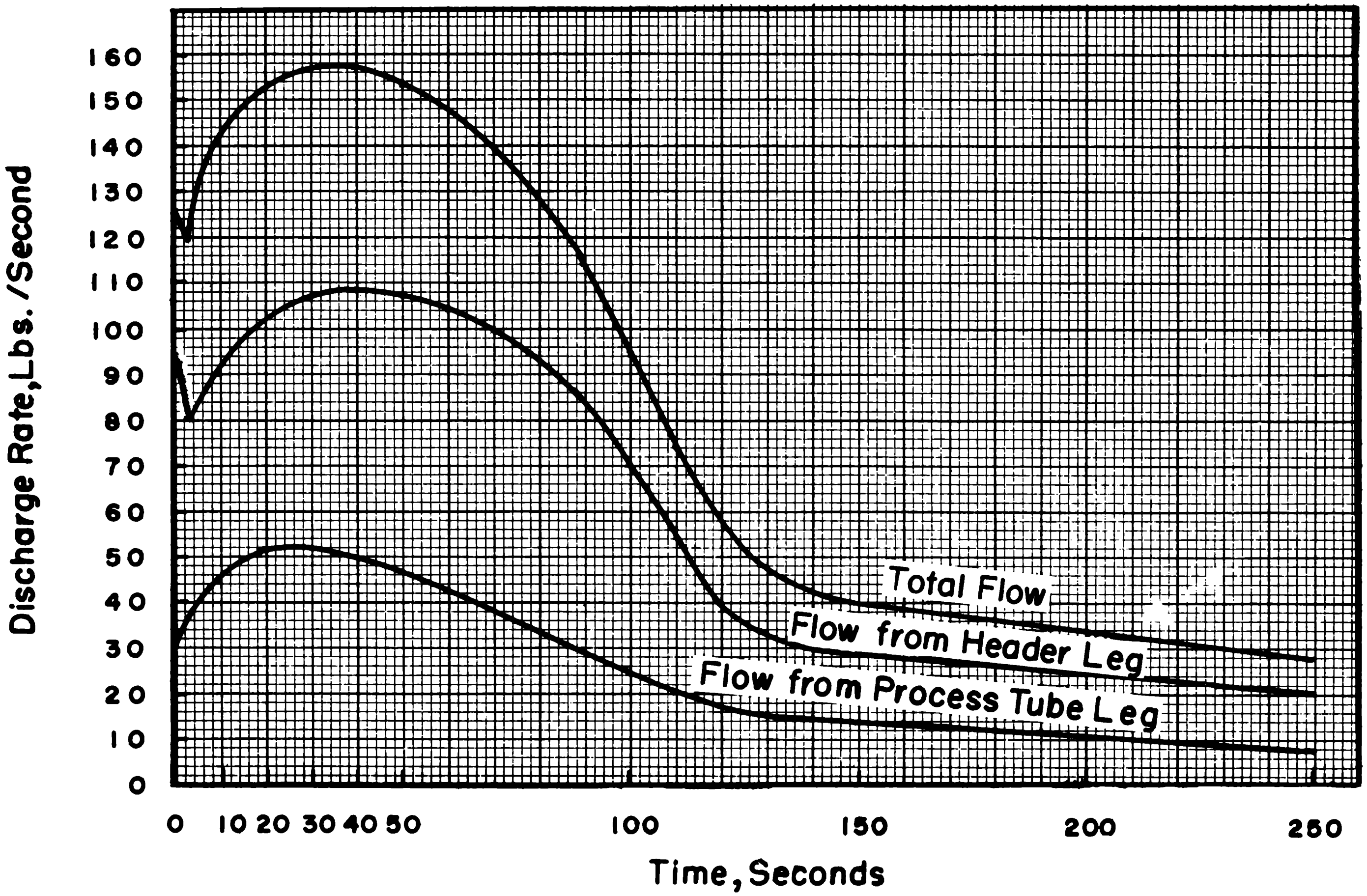

FIGURE 52

Loss of Primary Coolant
Following Parting of a Top Jumper

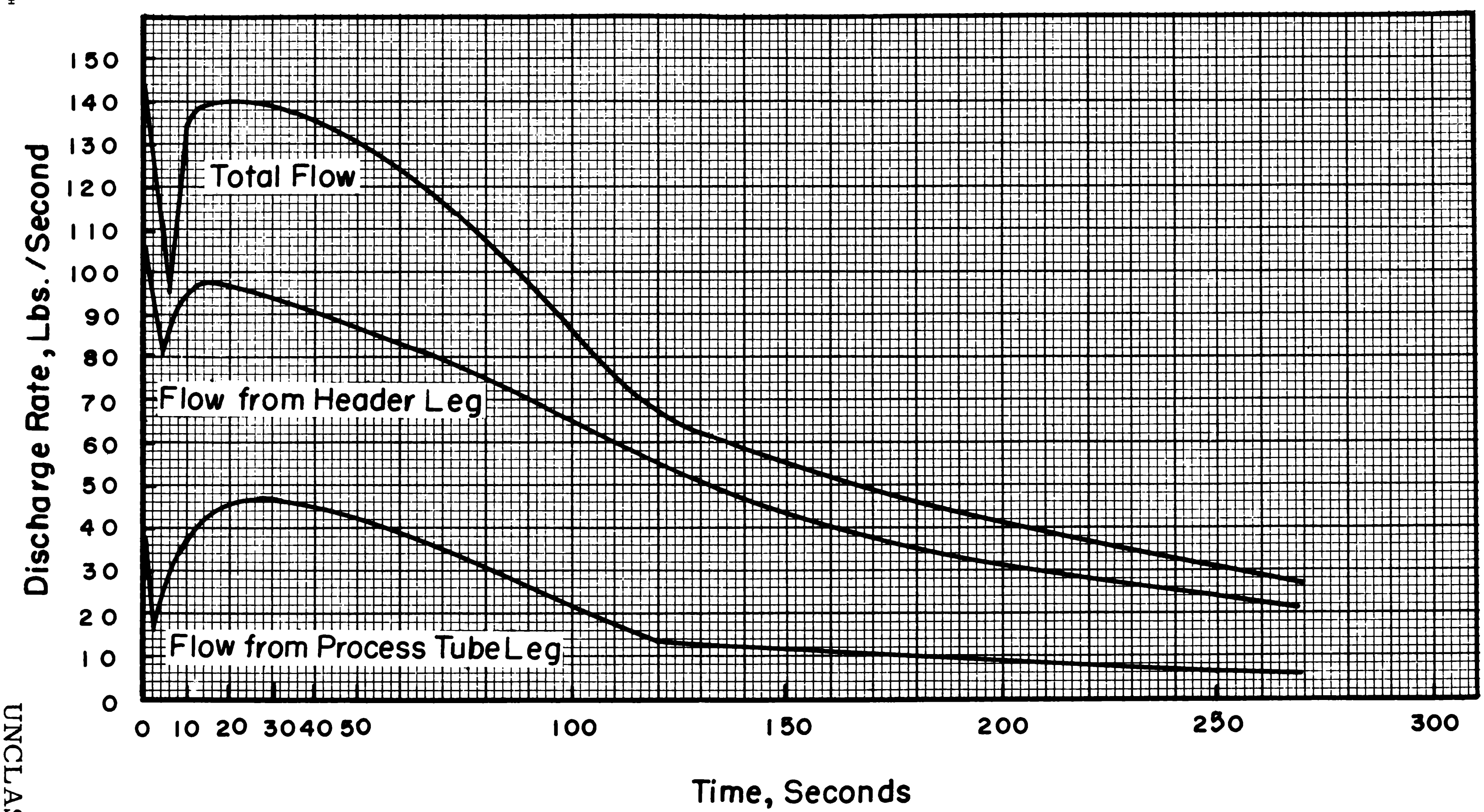

FIGURE 53

Loss of Primary Coolant
Following Parting of a Bottom Jumper

Case 1

In Case 1, defined as complete parting of a top jumper, the pumps would continue to pump for about 52 seconds, at which time they would become vapor locked. No temperature excursion in the intact tubes could occur during this phase since the flow of coolant approximates the normal flow rate and is more than adequate to remove the sensible heat in the elements and the decay heat generated.

For a considerable length of time after pump stalling the victim tube would present no hazard because flow rates would still be higher than normal. In any case, the temperature of the fuel element in the victim tube would be lower than those in the intact tubes due to the higher flow rates at all times through the victim tube.

Temperature profiles of a Mark I UO_2 element in an intact tube are shown in Figure 54 from the time of the incident to 200 sec. The fuel temperature decreases during the time pumping continues; temperatures rise after pumping stops, even when the fuel elements are covered by a stagnant pool of water.

At 202 sec after the incident, the system pressure is 100 psig so that injection of emergency water can begin. The water level at this time would be at the bottom of the fuel elements. Commencing at 202 seconds a flow of 375 gpm of auxiliary coolant is added to both the top and bottom ring headers. This situation is quite similar to that of the larger ruptures previously described in that bottom injection would be added to cover the fuel elements. However, in the jumper rupture case, no more steam can be produced in cooling the fuel elements than can escape through the rupture or be condensed in other portions of the primary system without raising the system pressure. Step-by-step heat and mass balances of the water in both headers were made by computing the enthalpy of the water in the bottom ring header at one second intervals after injection was

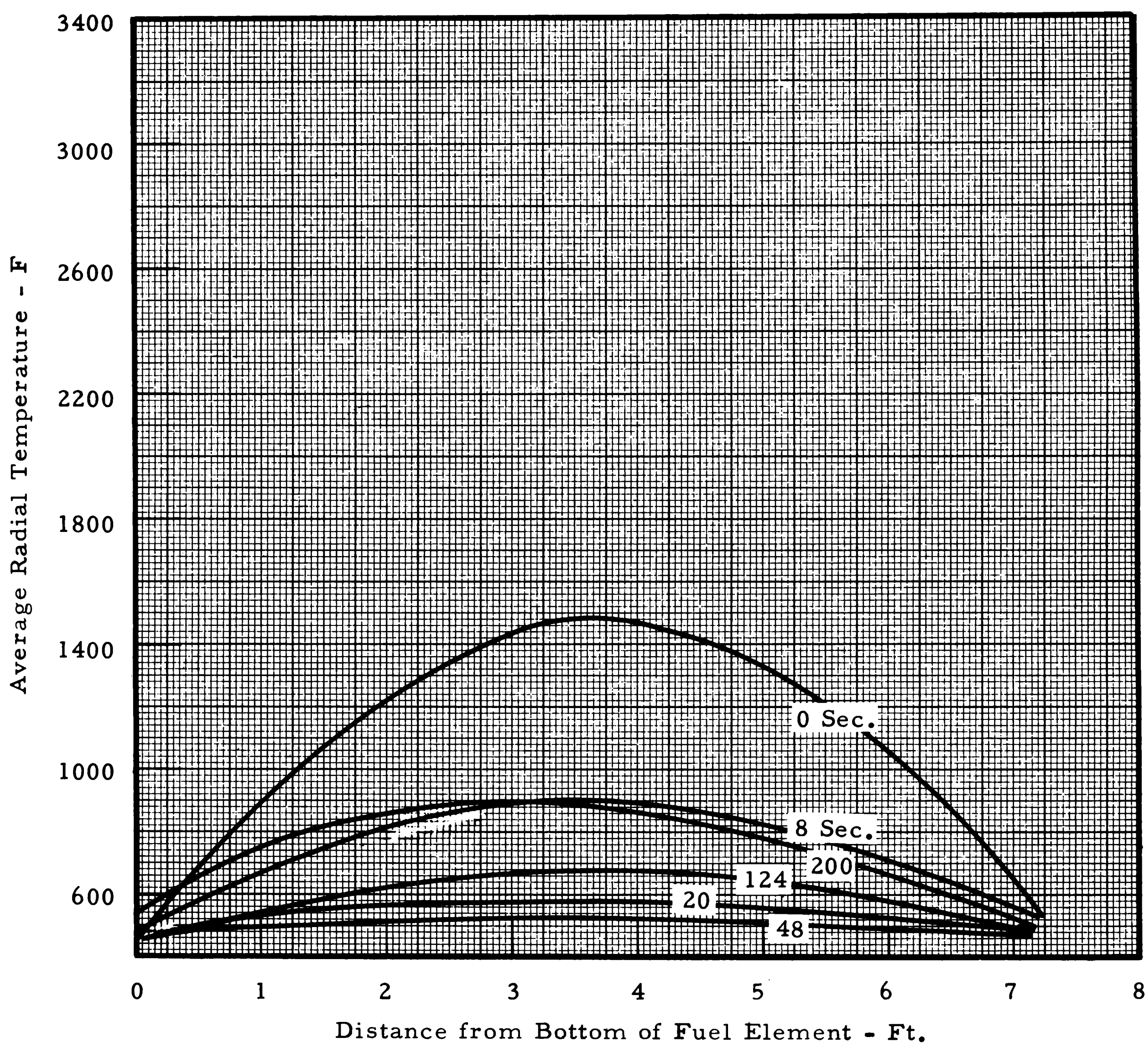

FIGURE 54

Mark I UO_2 Fuel Temperature
after Parting of a Top Jumper

started. It was assumed that fluid would leave the bottom header by loss out the victim tube or by vaporizing and either condensing in the upper ring header or being lost from the header leg of the rupture. A stepwise temperature excursion computation was performed to determine the steam generation rates with the energy and mass balance values. Figure 55 shows the computed water level as a function of time after injection until the upper ring header becomes filled with water.

Once the upper ring header is filled, the water would begin overflowing through the jumpers into the intact process tubes. This would be possible because the water would be subcooled and could condense any steam remaining. At 232 sec after the incident the fuel elements would be covered to 90 per cent of their length and the upper ring header would be filled. The water temperature would be 224 F and the pressure would be about 24 psia.

Critical flow of water out the rupture would be approximately equal to the injection rate when the primary system pressure is 100 psig. As long as the pressure in the reactor remained below 100 psig, the coolant would continue to rise. Computations based on the injection rate and heat transfer rate to the water indicate that the process tubes would be filled with water before sufficient heat had been transferred to the water to raise its temperature to the saturation temperature at 100 psig. Thus it is considered that the reactor could be filled with water. However, very little subsequent circulation would occur because the rate of injection of coolant water would be just sufficient to flow out the rupture, that coming in the top going out the jumper stub. Steam would form in the intact process tubes and drive out the water. After the water is driven out, steam would come in contact with cooler water in the headers, condense, and the tubes would refill with water. Since there are 84 intact tubes with different heat generation rates, this would probably be a cyclical occurrence with different tubes discharging at different times. This could continue for an extremely

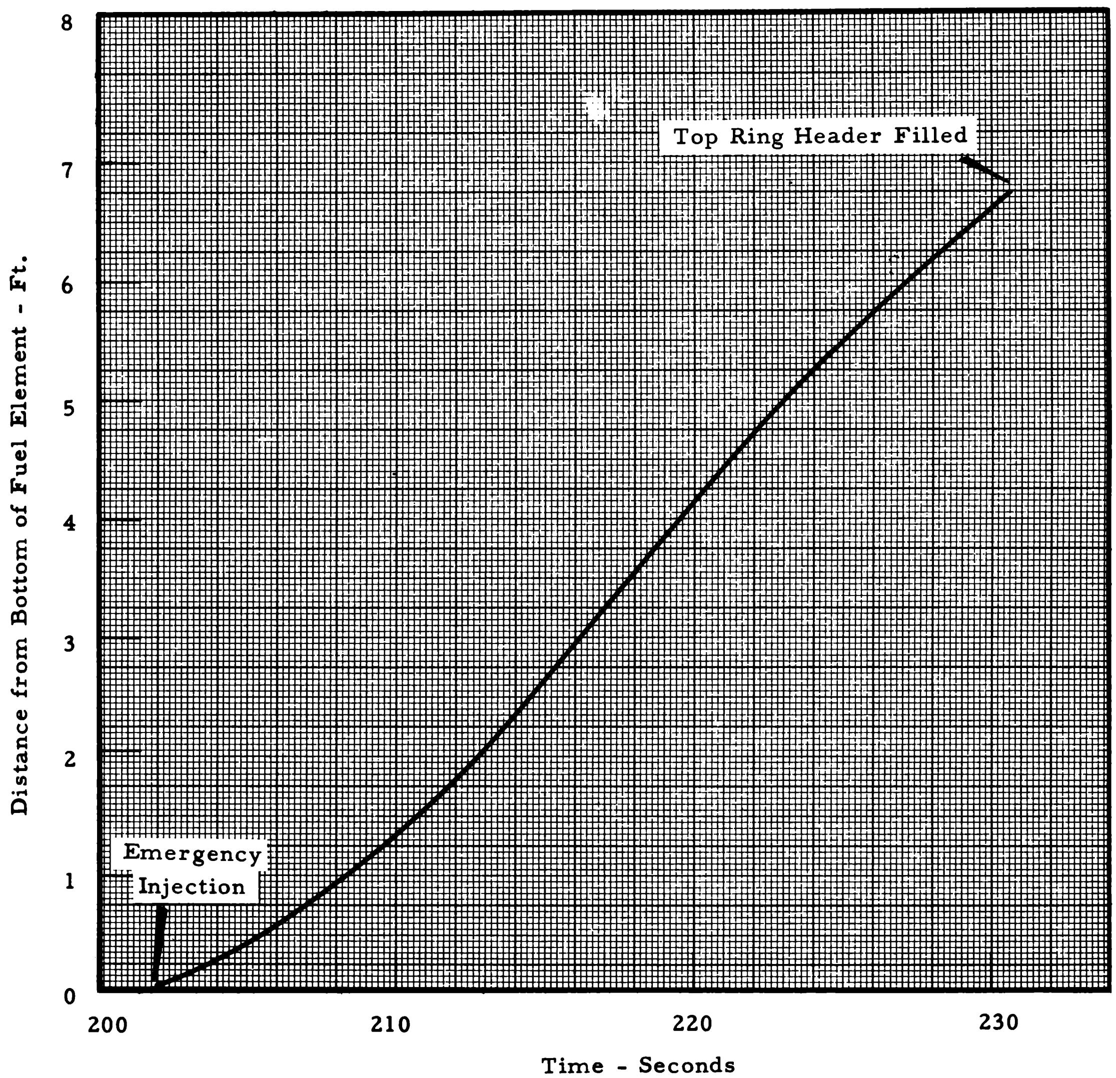

FIGURE 55

Top Jumper Severance,
Back-up Coolant Level in Process Tubes

long time after the incident. In no case, however, is any danger of fuel melting anticipated as long as the injection system remains in operation.

Case 2

Case 2 is postulated as the rupture of a jumper tube connecting a process tube to the bottom ring header with the primary coolant being lost from the victim tube stub and the bottom ring header stub. Again, the pumps are not expected to fail for a considerable period of time after the incident, estimated as 58 sec. Adequate cooling of all tubes would occur during the time of continued pump operation.

After pump stalling, it is again necessary to consider both the victim process tube and the intact process tubes. The flow rate through the victim tube would still be approximately 10 lb per sec at 4 min after the incident. Hence, the victim tube would be in no danger at any time.

Considering the intact tubes, with the pumps having failed at 58 sec, the fuel would remain covered with a stagnant pool until 137 sec after the incident, when the level of primary coolant would fall to the top of the fuel. It was computed that the fuel elements would be completely uncovered by 200 sec after the incident. By the time the fuel elements become completely uncovered, the maximum temperature of the most critical element, Mark I Pu-Al, would rise to 801 F. The system pressure would decay to 100 psig at 270 sec after the incident. There would thus be a considerable period of time during which the fuel elements would be steam blanketed and an adiabatic phase would occur. Computations showed that the maximum temperature of a Pu-Al fuel element would reach 1005 F at 270 sec after the incident.

A computation of the rate of filling the upper ring header and outlet pipe indicated that the emergency coolant starting at 270 sec would overflow the top ring header into the intact tubes at 305 sec after the incident. This was based on the assumption that the valve in the 14-inch outlet pipe to the heat exchanger would be closed so that it would not be necessary to first fill the heat exchanger leg of the primary system. At this time the maximum temperature of a Pu-Al fuel element was computed to be 1100 F. This temperature drops 6 F during the first 5 sec after the top ring header is filled and continues to drop thereafter.

The situation appearing most likely during top injection would be that injection water would flow downward through the tubes and about 65 per cent of the water would vaporize. Since the water in the headers would be at a temperature lower than the saturation temperature of the steam in the tubes, a nonequilibrium state would exist. The steam would condense after contacting the relatively cool water in the headers. Thus the pressure of the steam in the tubes would tend to approach the saturation pressure of the cooler water in the header.

Again computations showed that cooling of all process tubes is adequate so long as emergency water injection, at a minimum of 100 psig, continues.

c. Process Tube Rupture

Failure of a process tube would endanger the fuel elements in that tube and possibly damage the shroud tube and calandria. Other reactor tubes would be threatened by vapor binding if the coolant were discharged through the rupture at a sufficient rate. The reactor would be scrammed by a flow meter trip, which would also identify the leaking tube. Moisture detected by the reactor atmosphere sampling system would indicate that the leak was inside the reactor.

An analysis of the possible reactor damage associated with a process tube leak was performed.[24] The discharge from a process tube leak is vented up the process-shroud tube annulus, around the shroud tube bellows,

(24) Muraoka, J. PRTR Hazard Analysis for Various Mechanical Failures, HW-60963 REV. June 6, 1959.

through a one-inch space between the top tube sheet and the bottom of the top shield, and out three eight-inch vent pipes to atmosphere. The maximum pressure build-up within the shroud-process tube annulus was estimated by determining supply and demand curves for the system and interpreting the intersection of the curves as the equilibrium point. The supply curve was based on the normal system operating conditions of 478 F inlet temperature 530 F outlet temperature, and 1025 psig pressurizer pressure. The demand curve was based on the two phase pressure drop characteristics (using Martinelli factors) of the vent path. These curves are shown on Figure 56. The maximum pressure so obtained was 700 psia at a discharge rate of 100 lb/sec. The system is dynamically capable of maintaining this discharge rate for about 100 seconds. The estimated maximum pressure was deduced on the basis that the discharge rate was limited by the pressure drop in the process-shroud tube annulus, not by the size of hole in the process tube.

Taking into account critical flow effects through the process tube rupture in the discharging steam-water mixture yielded the following numerical results:

Process Tube Hole Diameter (Circular hole)-inches	Discharge Rate lb/sec	Pressure in Shroud-Process Tube Annulus psia
>1.32	100	700
<1.32	<100	<700
0.806	18.75	180 (approximate pressure of shroud tube)

It is apparent that the shroud tube would be ruptured by a process tube failure of moderate proportions.

Possible damage to the calandria and adjacent shroud tubes was considered in the event of shroud tube rupture. The calandria is vented by connecting lines between the moderator storage tank and calandria, which are equipped with eight-inch rupture discs relieving to atmosphere at a pressure

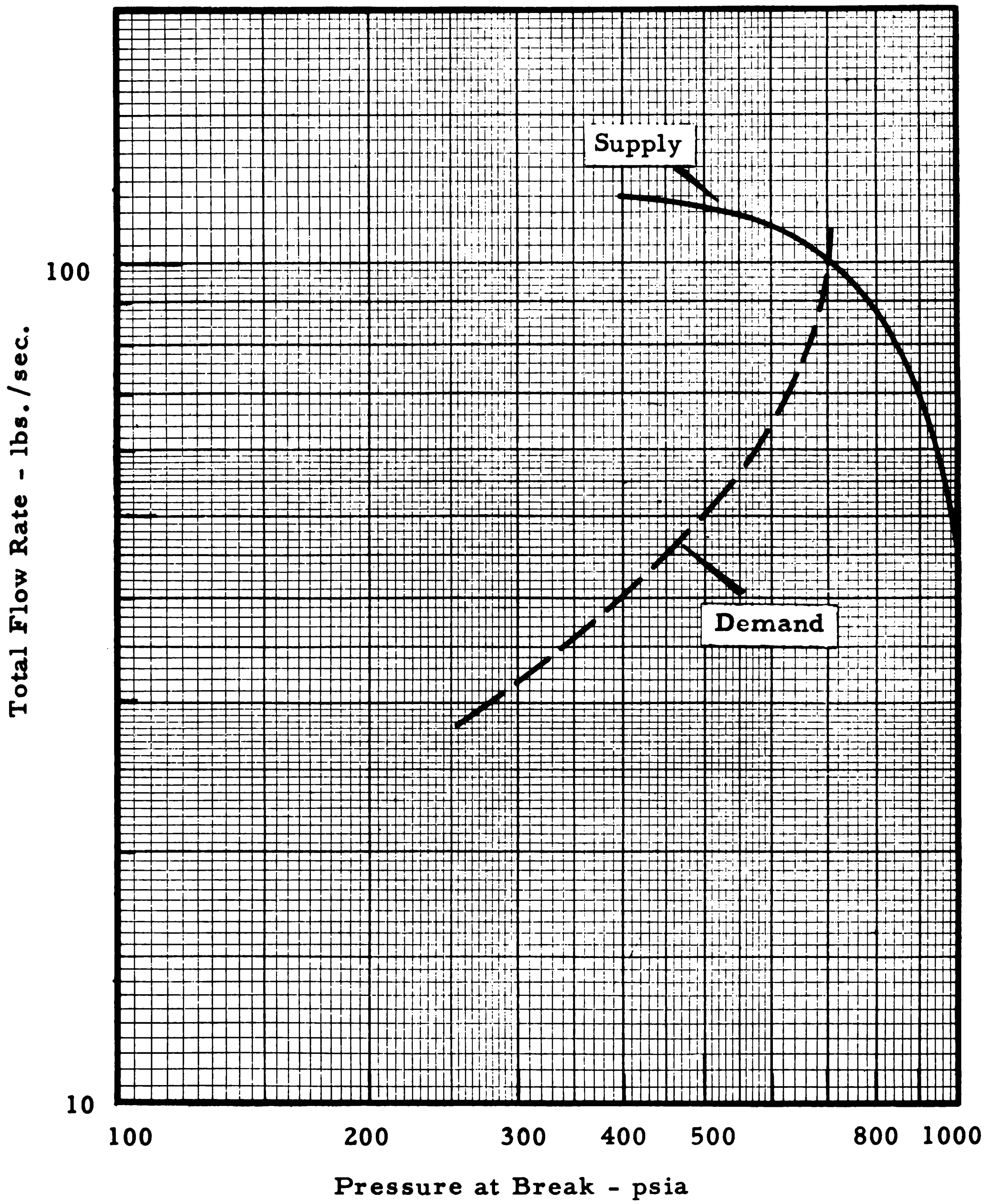

FIGURE 56

Process Tube Rupture,
Supply and Demand Curves

of 7 to 8 psig. The calandria can withstand a pressure of this magnitude for a limited period of time with little or no damage (calandria design pressure 5 psig). Shroud tube collapsing pressures range from 55 to 60 psig at 500 F.[25]

Each of the 8 lines can discharge 22.5 lb/sec at a calandria pressure of 7 psig, making a total of 180 lb/sec total relief capacity. Immediately following the scram, while moderator is being drained from the calandria, the four lower lines would be filled with water and their relief capacity would be slight. However in less than 5 seconds the full relief capacity would be available. Since the maximum conceivable discharge rate is some 160 lb/sec (refer to Section (b) on jumper break) with complete severance of a process tube, a very severe process tube leak can be tolerated with little or no damage to the adjacent shroud tubes.

An analysis of the effects of a process tube failure on fuel temperatures was performed. The analysis paralleled those described in the previous sections which dealt with exterior piping ruptures. The postulated sequence of events is the same as for exterior piping ruptures with blowdown, adiabatic, and injection phases occurring. Coolant flow rates through the victim tube would always be greater than those in the intact tubes; this analysis deals only with intact tube conditions since fuel temperatures in the victim tube would be no higher than those in the intact tubes.

On loss of coolant, emergency light water injection can be supplied by the boiler feedwater pumps (at 500 psig head). A low capacity boiler feedwater pump is supplied with emergency generator power; therefore the analysis is valid for two simultaneous failures - process tube failure and normal pump power supply failure.

Since the severity of a process tube failure can vary widely it was necessary to consider several rupture sizes. For all cases it was assumed

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(25) Gruver, R. L. Final Report - Design Test PR-25, Shroud Tube Collapsing Pressures and Installation Methods. HW-57089. August 12, 1958.

that the leak occurred near the inlet of the tube because this would cause the most rapid drainage of the reactor and the most pessimistic fuel temperatures.

Very large leaks would parallel the process tube jumper rupture cases considered in the previous section; complete severance of a process tube at the inlet would be the same as Case 2 in which emergency injection at a head of only 100 psig is adequate to prevent fuel melting. The analysis largely considered failures of lesser magnitude.

The rate of coolant loss (blowdown phase) was computed by critical flow relationships and by taking the system pressure as the sum of the heavy water vapor pressure and that provided by the helium pressurization system. It was assumed that the primary circulating pumps vapor lock at the time when the heavy water is drained from the pressurizer, steam generator, and connecting pipes (about 200 cu ft). After pump vapor locking, the system pressure would continue to decrease until the injection system pressure is attained. During this period the fuel elements are partially uncovered; an "adiabatic" phase would occur. Upon the initiation of injection the reactor would again refill at a rate computed from the injection pump characteristics, energy and mass balances in the ring headers, and steam generation rates. During injection, an oscillatory "perking" action similar to that described in section (b) is postulated. The peak Pu-Al fuel (most critical) temperatures attained after the adiabatic phase, and before injection decreases the fuel temperatures are given in Table VI.

TABLE VI

Peak Temperatures After Process Tube Failure

Approximate Diameter of Hole (inch)	Maximum Temperature Attained by Pu-Al Fuel (F)	Time after Rupture when Maximum Temperature Occurs (minute)
0.32	>1200	35 (at 1200 F)
0.4	1200	25
0.56	900	11
0.75	800	7

As is seen, as the hole size decreases the peak temperature attained is increased. This is largely due to the effect of the smaller holes greatly lengthening the time after the incident at which injection can begin while not appreciably affecting the amount of coolant in the tubes when injection begins. Process tube holes of about 0.4 in. diameter lead to temperatures in the Pu-Al fuel melting range.

Upon indication of a process tube leak the reactor will be shutdown and depressurized (which would permit emergency injection at an earlier time). At least 25 min would be available in which to do this before the minimum fuel (Pu-Al) melting point is reached.

It is concluded that the hazard of fuel melting subsequent to process tube failure is minimal; excessive fuel temperatures after small leaks (0.4 in. diameter hole or less) are avoided by depressurization of the primary loop over a period of at least 25 min followed by automatic injection of emergency coolant; the automatic emergency injection system functions to adequately limit fuel temperatures after larger leaks with no manual action required.

To minimize the probability of a serious process tube leak or failure, the zircaloy process tubes will be thoroughly tested and examined prior to installation as follows:

(1) Wall thickness will be measured by a Vidigage ultrasonic instrument.

(2) All flanges and tapers will be radiographed.

(3) The inside and outside surfaces will be fluorescent dye checked.

(4) The small and large diameter portions of each tube will be eddy current tested.

(5) The small and large diameter portions of each tube will be sonic tested.

(6) Suspect areas indicated by dye penetrant, eddy current, or sonic tester will be radiographed. Removal of defects will be done with vapor blasting, a fine hand file or emery paper. Suspect indications which

have not been definitely identified as defects will be conditioned only after all tests and in no case will the wall thickness be reduced below 0. 150 in. before pickling and autoclaving.

(7) After autoclaving, the wall thickness of all tubes will be checked with a Vidigage.

Process tubes will be routinely removed from the reactor for monitoring purposes at the rate of one tube per one to three months. These tubes will be measured and tested and the measurements compared with data obtained prior to installation to detect evidences of corrosion, erosion, mechanical damage and creep. The irradiated sample tubes will also be metallurgically examined to detect evidences of undesirable physical property changes during use. Upon any indication of such undesirable effects appropriate steps will be taken to lessen the probability of a process tube rupture.

d. Pump Shaft Seizure or Shaft Failure

Loss of one of the two operating primary circulation pumps by shaft seizure or fracture would cause an immediate reduction in flow and head. Because of the pressurizer, the system pressure would remain essentially constant. The reactor would be scrammed by a low flow signal from the tube flow monitors. The potential hazard is that vapor binding could occur in the reactor due to the increased coolant temperature and result in fuel element burnout.

The initial and final calculated equilibrium conditions, assuming no scram, are:

	Initial	Final
Bulk Outlet Temperature, F	530	540
Inlet Temperature, F	478	475
Total Flow, gpm	8400	6650
Hot Tube Flow, gpm	123	92.2
Hot Tube Outlet Temperature, F	542	547*
Hot Tube Power, KW	1200	1200
Outlet Jumper Pressure, psia	1055	1047

* Saturation temperature at 1047 psia.

The transient flow decrease would stabilize very rapidly after the pump failure (within 0.2 second). The temperatures would require a somewhat longer time to stabilize because of the coolant heat capacity and recirculation cycle time. As indicated in the table, the hot tube outlet temperature corresponds to a vapor pressure of 1047 psia, the discharge header pressure. Bulk boiling in the hot tube would occur on the last 2 feet of fuel element with a maximum heat flux in the boiling region of 250,000 Btu/(hr) (sq ft). Exit steam quality is 3 per cent. Laboratory experiments with an electrically heated prototype test section under conditions much more severe than these did not result in excessive temperatures or boiling burnout. These conditions were evaluated in terms of applicable flux correlations. The correlations of De Bortali,[26] et al, and Bernath[27] were used and found to yield burnout heat fluxes 11 and 1.6 times that calculated to occur after the failure, respectively.

As previously mentioned, the reduction in flow would cause a reactor scram, within 0.5 sec; the bulk boiling condition would not persist for more than one second at the most. This, together with the burnout safety margin constitutes sizeable protection against fuel burnout.

It is concluded that the failure of one primary pump by shaft seizure or shaft fracture would not cause fuel element burnout.

e. Steam Leak

The heat sink for dissipating the reactor energy consists of a heat exchanger with boiling water on the shell side, appropriate feed water and effluent piping, and the necessary auxiliary equipment. The primary coolant passes through 1/2 in. OD horizontal U tubes. Complete loss of the steam generator system would cause rising primary coolant temperatures and pressures until the reactor was manually shutdown or scrammed by primary or

(26) De Bortali, R. A. Forced Convection Heat Transfer Studies for Water in Rectangular Channels and Round Tubes at Pressures Above 500 psia, WAPD-188. October, 1958.

(27) Bernath, L. Prediction of Heat Transfer Burnout, Presented at the National Heat Transfer Symposium, National Meeting, Louisville, Kentucky, AICHE, 1955. (Preprint No. 8).

secondary system safety circuit trips, and the primary system over-pressure relief valves opened. In this section, an analysis of the hazards of large ruptures in the connecting steam generator steam piping is described.

The heat exchanger is a horizontal cylindrical tank containing 488 cu ft of liquid water (610 cu ft including steam voids) with horizontal U tubes normally submerged to a depth of 15 in. Upon depressurization of the steam generator, as after a connecting pipe rupture, the shell side temperature will be reduced and the primary coolant temperature will decrease provided sufficient cooling tubes are covered with secondary coolant to provide adequate heat transfer area. The possible hazard, then, is that rupture of the steam piping would cause the water level in the steam generator to decrease to the point where there is insufficient heat transfer area to handle the primary coolant heat load without excessive temperature drop.

Analyses were performed for the cases of rupture of the 26-in.main steam header on the top of the steam generator and rupture of a 4-inch blow-down line on the bottom of the steam generator.[28] In either case, since the normal set point of the steam pressure control valve is 425 psia, the pressure control valve would tend to close because of the pressure decrease within the steam generator. It was assumed that the steam generator feed system remained normal, feeding water into the steam generator and following the normal pump characteristic curves. The results of the analyses are shown on Figures 57 and 58. Discharge rates from the steam generator were limited only by critical flow of the steam-water mixture.

In the 4-inch line rupture case a reactor scram by the steam generator low liquid level trip occurred 32 sec after the rupture. A maximum of 16.5 per cent of the heat transfer tubes were uncovered approximately 120 sec after the rupture. At this time the steam generator water temperature

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(28) Muraoka, J. Ibid., HW-30963. page 186.

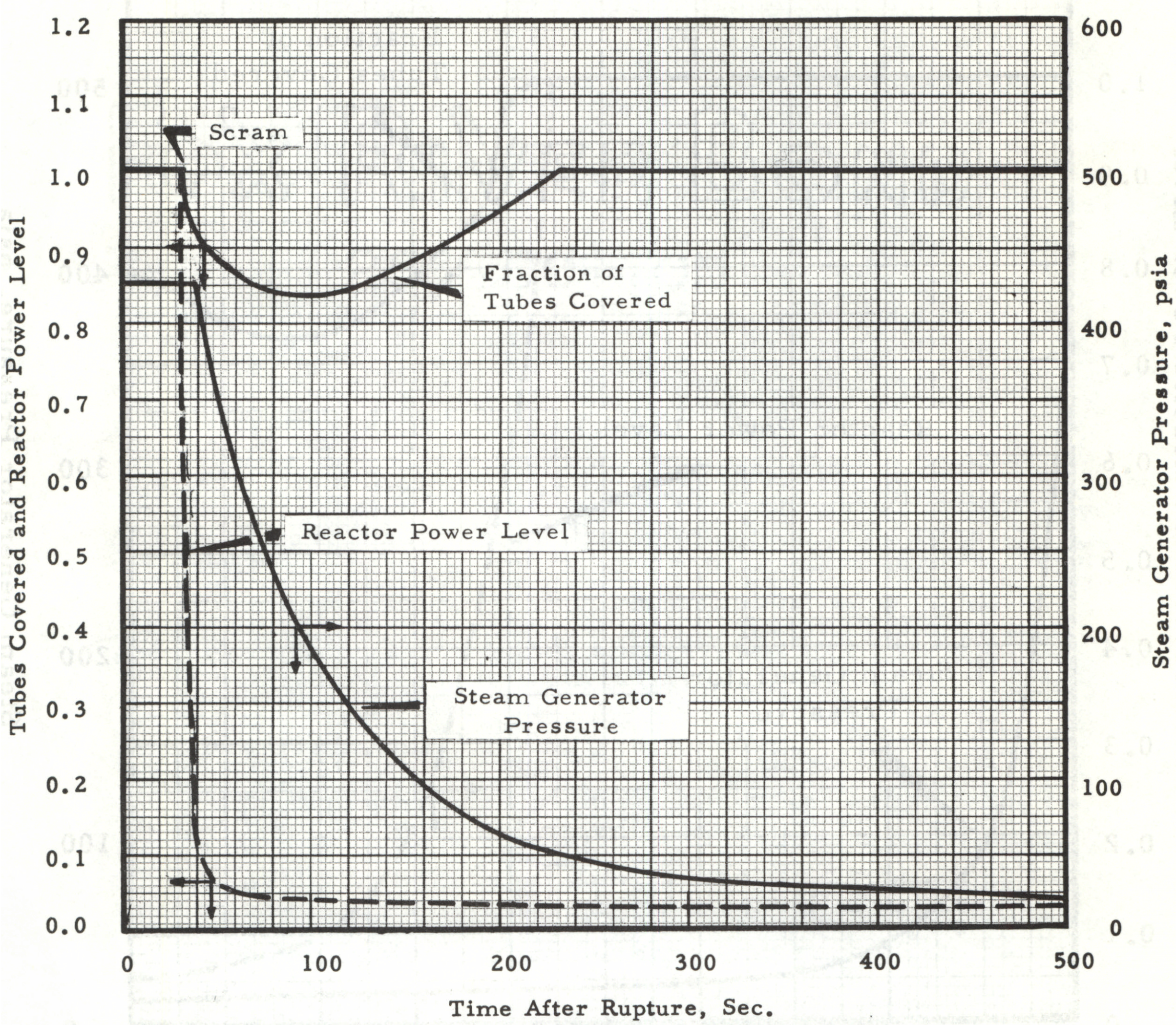

FIGURE 57

Steam Generator Conditions
after Rupture of 4-inch Drain Line

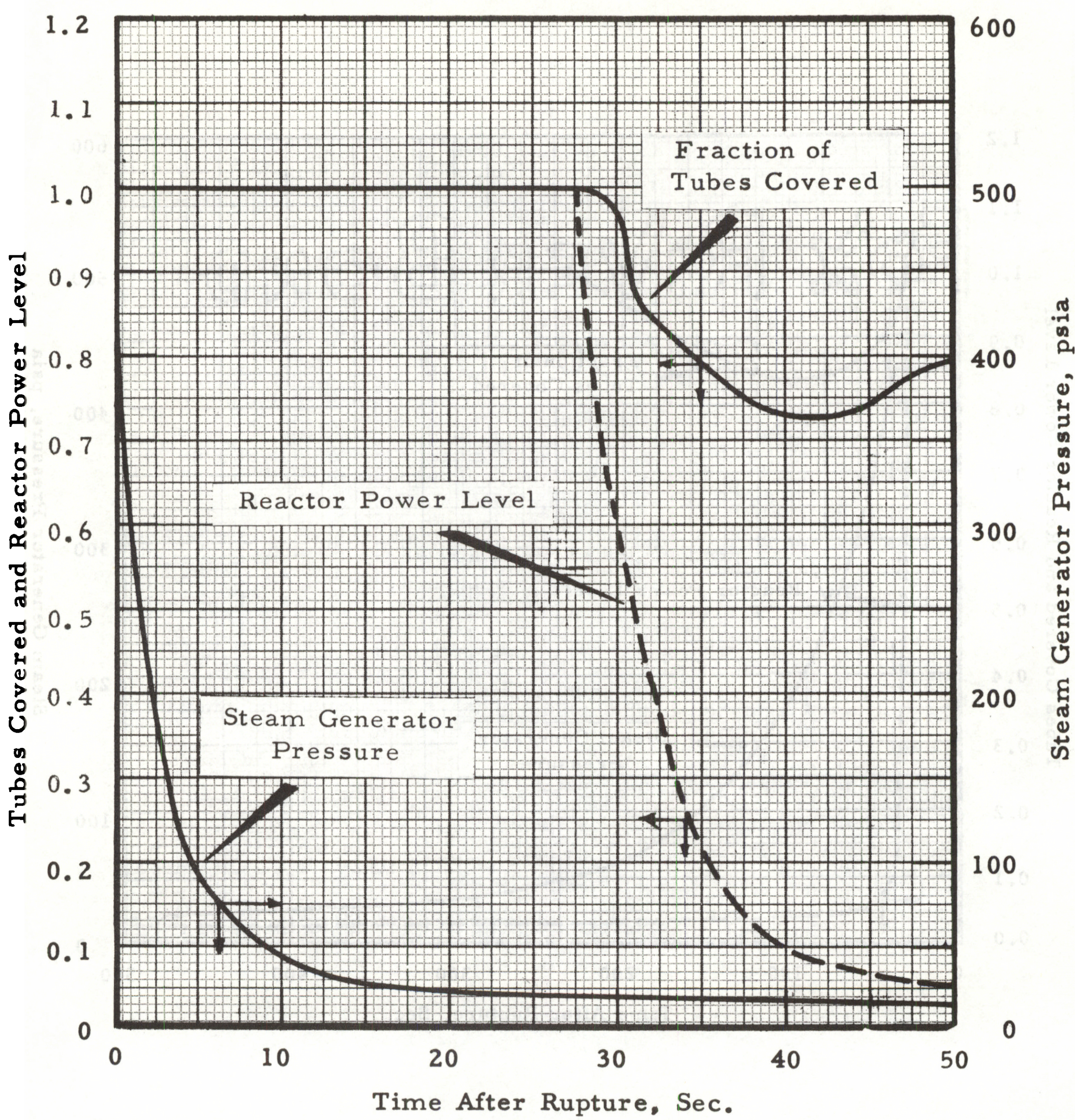

FIGURE 58
Steam Generator Conditions
after Rupture of 26-inch Steam Line

had decreased approximately 120 F and the reactor heat generation had decreased to more than compensate for the reduction in heat transfer area.

In the 26-in.main header rupture case the reactor trip point is reached 27 sec after the rupture, and the minimum water level occurs after 41 sec, uncovering 28 per cent of the tubes. The apparent liquid volume at the time of rupture is greater than in the previous case because of the large initial increase in steam flow rate and the attendant increase in void volume. At the time when the first tubes are uncovered the steam generator water temperature has decreased by 138 F. This, and the reduced heat load following the reactor scram more than compensate for the reduction in heat transfer area. During the transient, the primary coolant temperature would be reduced.

In either of the above cases the boiler feed water pumps would continue to supply water to the steam generator and would recover the tubes within a short time. It is concluded that rupture of the main steam discharge line or of the liquid blowdown line would cause neither a rise in primary coolant temperatures nor melting of fuel elements.

f. Steam Generator Feedwater Failure

As indicated in the previous sections, loss of the primary coolant heat sink could cause an increase in the primary coolant temperature and possibly lead to fuel element melting. Backup systems are provided to maintain a steam generator feedwater supply in the event of a power failure or feed line piping rupture.

(1) Power Failure

Upon loss of the normal electrical power source, the boiler feedwater pumps would stop. The power loss would also result in an immediate reactor scram. The secondary cooling system has been provided with back-up systems to prevent loss of cooling during a power outage.

These systems are described below:

(a) A 50 gpm capacity, 500 psig head, boiler feed pump connected to the steam generator by an independent line is supplied by the emergency electrical power system and can be manually started from the control room.

(b) A process water pump is started automatically about 5 sec after the emergency generator has reached full voltage and frequency. This is a low head pump which would supply boiler feed water in the event that the high head pump is not available. It would be necessary to depressurize the steam generator to approximately 100 psig before water from the low head pump could be admitted.

(c) Should both the normal and emergency power systems fail, the steam generator could be supplied with water at a head of 100 psig from the emergency diesel-driven pump. Use of this pump would also require depressurization of the steam generator.

A period of 7 min could be allowed for starting the high pressure backup pump without uncovering any of the steam generator tube bundle. If it were necessary to use one of the low head pumps, some tubes would be uncovered during the blowdown. However, even with the blowdown occurring 30 min after the power loss, less than 30 per cent of the tubes would be uncovered. The shutdown heat generated in the reactor at this time would be about 2 per cent of the normal heat load, and the steam generator shell side temperature would be reduced about 115 F during blowdown. Thus, the primary coolant temperature would remain well below the normal operating temperature. Manual starting of pumps and/or depressurizing of the steam generator would be accomplished long before the loss of heat transfer area would be sufficient to raise the primary coolant temperature above normal.

(2) Feed Line Piping Rupture

The 6-in steam generator feed line branches into two 4-in lines which pass through the bottom of the steam generator shell to supply the coolant distribution headers. Each of the 4-in lines contains a check valve located within a few inches of the shell to prevent expulsion of water from the steam generator in the event of a feed line rupture.

It is possible, however, for a rupture to occur in the short section of a 4-in. line between the check valve and the steam generator shell. In such a case, assuming an off-set break, the boiler feed water would be pumped out of the pump side of the break and a steam-water mixture would be expelled from the steam generator side. About 33 sec after occurrence of the rupture the reactor would be scrammed by the steam generator low liquid level trip. About 2 sec later uncovering of the heat exchanger tubes would begin. The decrease in reactor power would be much more rapid than the loss of heat transfer area, as shown on Figure 59. Up to 110 sec after the rupture the decrease in heat transfer area is less than the decrease in reactor power. With the low reactor power, even after complete loss of heat transfer area a period in excess of 5 min would be required to raise the reactor outlet temperature to the normal operating point.

A manually started emergency pump can be used to supply water through a separate feed line connected to the top of the steam generator. This water would be distributed by the demister to cascade over the tube bundle and would cool the primary coolant by evaporation. Vaporization of 19 gpm of water would be required to remove the heat generated in the reactor. The emergency pump is capable of supplying 50 gpm at 500 psig. This pump can be operated either on BPA or emergency generator power.

Consequences of a feed line rupture upstream of the check valves would be similar to those for a loss of pumping power. No blowdown would occur through the break. However, the loss of the feed line would prevent the use of certain back-up pumping systems. It would be necessary to supply water through the system described above under the case of feed line rupture downstream of the check valve.

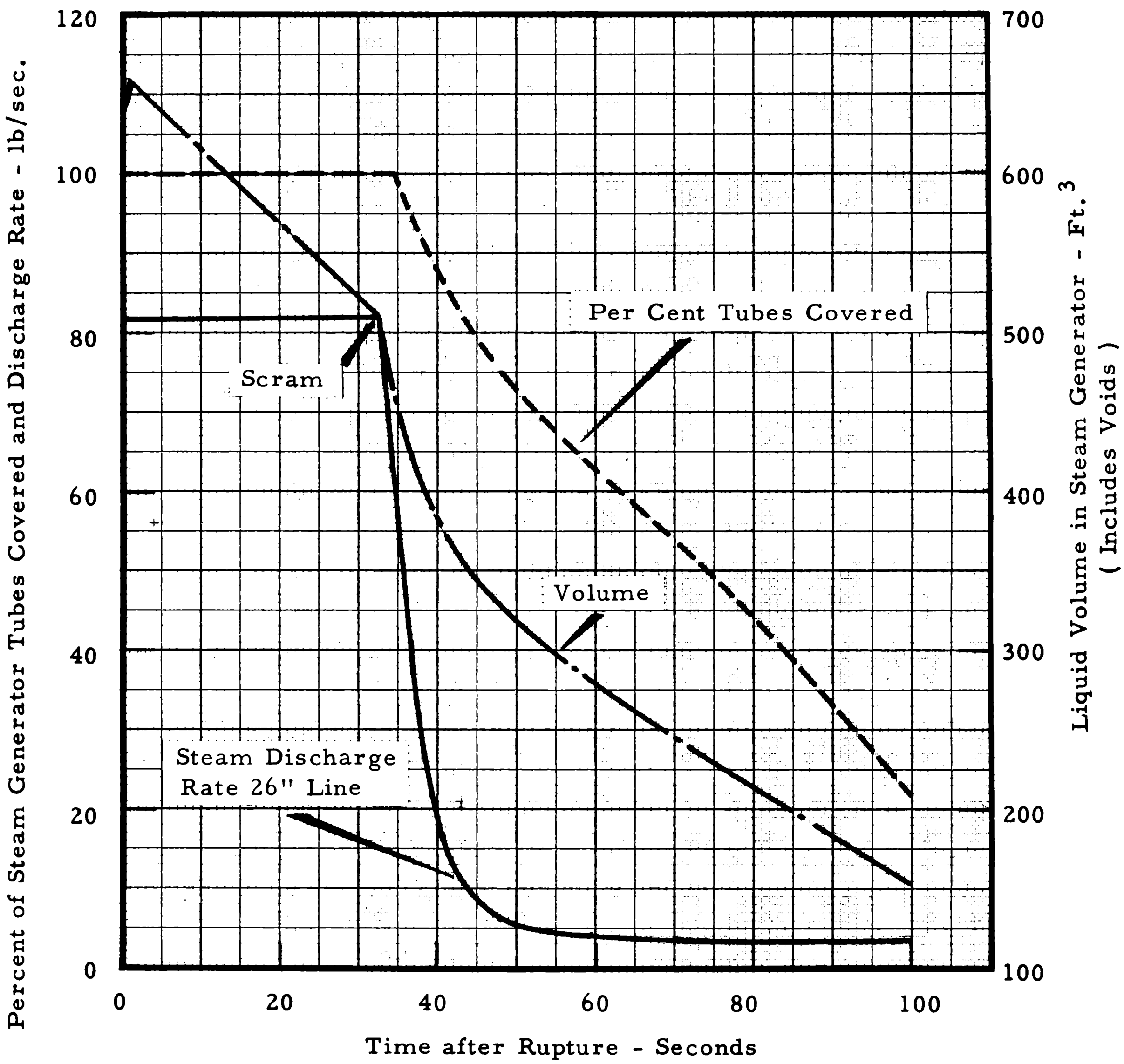

FIGURE 59

Steam Generator Conditions
after Rupture of Feed Water Line

g. Valve Malfunctions

Possible hazards to the PRTR as the result of malfunctioning automatic control valves in the primary and secondary coolant systems were evaluated. The analyses were generally performed on the basis of pessimistic assumptions and are briefly described in the following paragraphs. In no case of valve malfunctioning is the reactor exposed to serious hazard.

(1) Steam Generator Pressure Control Valve

A 26-in.diaphragm operated valve in the 26-in.discharge steam header controls the steam generator pressure. The pressure is sensed by a pressure tap in the steam void space of the steam generator. If the valve or control system malfunctions, the pressure control valve could either fully open or fully close. If the valve opened, the pressure would decrease to near atmospheric, the steam generator temperature would decrease, and the primary coolant temperature would be decreased. When the valve opened, the water level in the steam generator would decrease because of the hot water flashing; however, the boiler feed water rate would automatically increase and return the water level to normal.

If the pressure control valve closed, the steam generator pressure would immediately increase. The reactor would be scrammed by the steam generator high pressure trip circuit at a pressure of 450 psig. At 450 psig, the saturation temperature in the steam generator is 459. 6 F (normally 450 F); the primary coolant temperature would increase. At the time of scram a reactor inlet temperature of 487. 6 F, an outlet temperature of 539. 6 F, and a "hot tube" outlet temperature of 548 F could be reached. Since the saturation temperature at the reactor outlet remains at 548 F, bulk boiling would occur in the hot tubes. Bulk boiling would occur over the top 2 feet of these fuel elements. Following the scram these temperatures would decrease. The steam generator safety relief valve will be set to prevent overpressure damage and consequent loss of the heat sink. It is concluded that malfunction of the steam generator pressure control valve would not threaten fuel burnout.

(2) Process Water Surplus Disposal Valve

The normal boiler feed water stream flows to the deaerator via heat exchangers for the reflector, moderator, shield coolant, and fuel rupture detection systems. The feed water stream originates at the process water pumps and serves as the coolant in these heat exchangers. Surplus process water is diverted to the process sewer by a diaphragm operated valve which is pilot operated to maintain a constant pressure upstream of the dearator inlet valve. Should this valve, or its control mechanism malfunction, it could either open or close fully. If it opened, the deaerator supply would be stopped and the normal boiler feed water supply would stop. The water level in the steam generator would begin to decrease and would cause a reactor scram. Boiler feed would be restored by opening a motor operated valve in the process water by-pass line which would permit steam generator supply via the boiler feed water pumps directly from the process water pumps.

If the process water sewer valve closed, all of the process water would be routed to the deaerator - boiler feed system. The water level in the deaerator is controlled by a valve in the supply line which is mechanically controlled by the deaerator water level. The boiler feed system would not, then, be affected adversely by the valve failure. However, closure of the sewer valve would reduce the coolant flow through the upstream heat exchangers which cool the moderator, reflector, shield coolant, repurification, and fuel rupture detection streams. Above normal temperatures in these systems, as well as the lowered secondary coolant flow rate, would cause alarms to sound in ample time to take appropriate remedial action.

(3) Steam Generator Feedwater Control Valve

The steam generator feedwater control valve is a 6-in. diaphragm operated valve controlled by the liquid level in the steam generator. Should the valve actuating mechanism or control mechanism seriously malfunction, the control valve might either completely open or close.

On closure, the normal supply of feedwater to the steam generator would be stopped and the water level would decrease. This condition would be indicated by the low steam generator feedwater flow alarm. The lowering of water level in the steam generator would continue until the low level steam generator reactor scram point is reached. At this time uncovering of about 9 per cent of the cooling tubes would result from the collapse of steam voids. However, the rapid decrease of reactor heat output following the scram would more than compensate for the loss of heat transfer area. The volume of water in the steam generator would be sufficient to prevent above normal primary coolant temperatures for several hours (e.g. after 3 hr with the reactor power output about 1.2 per cent normal operating level, 50 per cent of the heat transfer area would be covered). Within a few minutes after the valve failure action would be taken either to manually open the valve or to supply water from the 50 gpm pump which feeds the line entering the top of the steam generator.

If the control valve were to fully open, the steam generator feedwater flow rate would increase by approximately 50 gpm. This would cause the water level in the steam generator to increase until, after some 18 min, it would be completely filled.

At about this point, the demister, in the top of the steam generator would be covered with water and the pressure drop across it would increase. The steam pressure control valve is controlled by the steam pressure in the boiler; it would tend to open to maintain a constant set pressure (~425 psia). The pressure drop across the demister could increase by as much as 385 psi before the pressure within the steam generator could rise. Pressure drops across the demister of this magnitude would probably collapse the demister and eliminate it as a flow restriction. The system would then continuously discharge about 50 gpm of liquid water out the steam header to the contact condenser. This situation would not be harmful and the steam generator pressure would remain at the pressure controller set point. It is concluded that no increase in primary system temperature would occur.

(4) Steam Generator By-Pass Valve

A limited capacity by-pass line around the primary heat exchanger is provided to by-pass a fraction of the primary coolant for fine control of the reactor inlet temperature.

An 8-in. diaphragm operated valve in this line is controlled by the reactor inlet temperature. During normal operation it is expected that a portion of the primary coolant will by-pass the heat exchanger. This valve or its control system could malfunction and cause it to either fully open or close. If the valve were to close, all of the primary coolant would be routed through the heat exchanger; the bulk primary coolant temperature would be reduced. If the valve opened, the fraction of primary coolant flowing through the heat exchanger would be reduced; the bulk primary coolant temperature would rise. The latter case was analyzed to determine the possible extent of the temperature rise and any resulting undesirable effects.

For purposes of the analysis a pessimistic assumption was chosen; steam generator fouling necessitates that the by-pass valve be completely closed to route all primary coolant through the heat exchanger in order to maintain the desired primary coolant temperature. When the by-pass valve is fully opened some 20 per cent of the primary coolant by-passes the heat exchanger; the over-all heat exchanger coefficient decreases from 681 to 652 Btu/(hr)(sq ft)(F) due to the reduced velocity. The system temperatures would be increased by $\sim$10 F when steady-state conditions are reached in about 60 sec.

The reactor inlet temperature would be 488 F, the bulk reactor outlet temperature 540 F, and the outlet temperature of the "hot tube" at the saturation temperature of 548 F, all based on 70 MW reactor power level.

The saturation temperature in the outlet ringheader is 548 F (1055 psia). When the reactor inlet temperature or any tube outlet temperatures exceeded their normal values, annunciator alarms would sound and the reactor would be manually shutdown or the steam generator steam discharge pressure control valve would be adjusted to decrease the primary coolant temperatures. These steps would be taken within minutes of the alarm. Prior to shutdown boiling would occur over the last 1.5 ft of fuel elements in the hottest tubes, with an outlet steam quality of 2 per cent. The maximum fuel element heat flux in the boiling section of the tubes would be 250,000 Btu/(hr) (sq ft). Laboratory experimentation with an electrically heated prototype test section under conditions much more severe than these did not result in excessive temperatures or boiling burnout.

It is concluded that, after a malfunction of the heat exchanger by-pass valve, fuel element burnout conditions are not approached Bulk boiling would not occur in the outlet header. No significant hazard to the reactor exists.

(5) Deaerator Steam Valve

If the control valves in the feedwater or steam deaerator supply lines fail, the worst that could happen is that the normal boiler feed supply would be stopped. As indicated elsewhere, this condition would be known when alarms actuated by low boiler feedwater flow and low steam generator feedwater level (also a reactor scram) occur. Ample time is available after these alarms to provide steam generator feedwater via the process water by-pass system or the emergency well pump system.

(6) Helium Pressurizer Valves

The primary coolant system is pressurized by helium supplied at 2200 psig to two valves; one having a capacity of 50 scfm and the other 3000 scfm. These valves are diaphragm operated and controlled by the pressure in the pressurizer. Two safety relief valves on the pressurizer are provided which can each relieve 40,500 lb/hr of gas at the primary loop design pressure (1250 psig). At a pressurizer pressure of 1130 psig the reactor would be scrammed either by the high pressure trip in the normal safety circuit or by a pressure switch with a contact in series with the dump valve solenoids. The sizing of the relief valves was based on the criterion that the relief valves would function to limit the primary loop piping pressure to 110 per cent of the design pressure in the event that a pump power failure occurred and the reactor was not scrammed (not considered credible). Since the relief capacity greatly exceeds the maximum rate at which helium can be added, failure of the valves in the connecting lines, cannot, in itself, cause serious over-pressurization of the primary loop.

If the helium addition valves fail by closing, the primary system pressure would gradually decrease until the low pressurizer pressure trip point is reached (~ 1020 psia) and the reactor is scrammed. At this time the primary coolant temperature would begin to decrease due to the reduced power. Figure 60 shows the transient pressures after the valve failure. A minimum margin of 30 psi would exist between the pressure in the discharge header and the vapor pressure at the "hot tube" (542 F) outlet. Therefore, bulk boiling is not indicated.

3. Procedural Errors

Manually actuated valves in the primary coolant system are under direct control of operations personnel. If these valves are erroneously manipulated severe hazard to the reactor components can result. The most sensitive valves are those in the inlet jumpers of each process channel and the 14-in. valve in the primary coolant loop at the entrance of the heat exchanger. The jumper valves, in the lower access space, are hand operated and thus are not physically accessible during operation because of high radiation levels and temperatures. If one of the valves were left closed or partially closed during an outage the tube coolant flow monitor

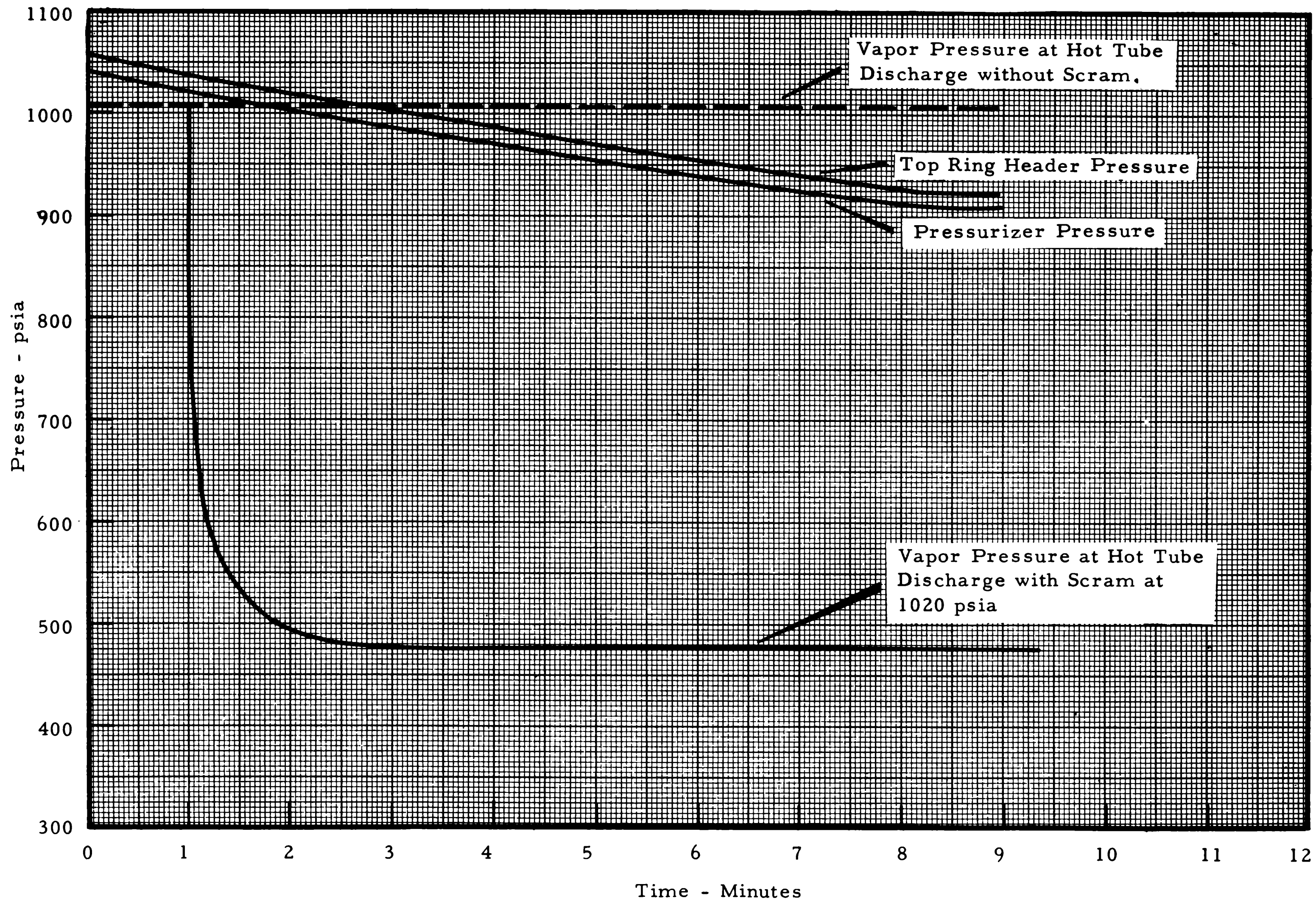

FIGURE 60

Primary Coolant Pressure Transient
after Helium Valve Failure

safety circuit trip would have to be by-passed to permit reactor startup. This is not permitted by procedure and is a condition immediately detected by the alarm circuitry.

Closure of the 14-in. primary loop motor-operated valve during operation would result in an immediate reactor scram from the low flow trips. Two 300 gpm capacity liquid relief valves located on the reactor side of the 14-in. valve would limit the pressure in the reactor side piping to 110 per cent of the design value, and would prevent overpressure damage to the system. However, the loss of coolant flow would probably result in fuel melting within a short time. Interlocks are provided with the reactor safety circuit to de-energize the valve operator when the safety circuit is made up.

E. Metal-Water Reactions

1. General

The reaction between solid zirconium and water follows the classic corrosion rate equation:

$$\frac{\Delta W}{A} = K\, t^{\frac{1}{2}} \exp\left(-\frac{Q}{RT}\right);$$

except for a very short initial period. Reaction rates for the molten metals with steam would be expected to be higher than rates predicted by an extrapolation of the equation above. A study of zirconium-steam reactions at Battelle [29] showed that reaction rates of solid Zircaloy-2 with steam correlated with a form of this equation. Also, it was found that molten Zircaloy-2 and steam reacted at rates greater than the extrapolation of this equation. Generally, violent reaction of zirconium or aluminum with water does not take place unless the metal temperature is at or above the melting point.

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(29) Lemmon, Alexis W., Jr. Studies Relating to the Reaction Between Zirconium and Water at High Temperatures, BMI-1154. January 3, 1957. page C-1.

Spontaneous accidents which can cause melting of fuel element jackets and/or cores are:

a. Nuclear excursion, and

b. Loss of coolant.

It has been shown that a high degree of completion of reaction and extremely rapid reaction can be achieved only when the metal is finely dispersed. Higgins and co-workers correlated the violence and degree of completion of molten metal-water reactions with the droplet size of the molten metal.[30] The greatest uncertainty in the analysis of metal-water reactions in reactors involves the postulation of a mechanism of dispersal of the molten metal and prediction of the droplet size.

UO_2 and water will react under certain conditions, but the reaction is not very rapid. Pressed and sintered UO_2 pellets were exposed to 650 F and 750 F degassed water for about 300 days with little or no weight change of the specimens. Several specimens were exposed to 650 F water containing 1 to 3 cc O_2/kg. The compacts lost approximately 15 per cent of their weight in 8 days.[31] In other tests in which UO_2 was melted in pressurized water it was found that there was no chemical reaction.[32] Since UO_2 either reacts slowly with water or does not react, no further analysis of UO_2-water reactions was made in estimating the effects of PRTR accidents.

An analysis of the potential for metal-water reaction in the PRTR under conditions of credible nuclear excursions and loss of coolant follows.

2. Nuclear Excursion

In Section VI-C, Nuclear Excursions, credible nuclear excursions in the PRTR are analyzed. In none of these excursions did the core or surface temperature reach the melting point of zirconium (see Figure 41, 42, 43, 44, 45, and 46). In only one of the excursions, Figure 42, a startup

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(30) Higgins, H. M. and R. D. Schultz. The Reaction of Metals with Water and Oxidizing Gases at High Temperatures, IDO-28000. April 30, 1957.

(31) Belle, J. and B. Lustman. Properties of UO_2, WAPD-184. September, 1955.

(32) Bostrom, W. A. and Alan B. Rothman. Chemical Stability of Uranium Dioxide in PWR Water, WAPD-MDM-3. April 3, 1954.

accident in which the moderator level is raised at the maximum rate after all shim rods have been removed and the scram trip is set at 1.3 P, did the core temperature of a Pu-Al element exceed the melting point of aluminum, 1220 F.

The startup accident illustrated on Figure 42 with scram at 1.3 P is considered incredible, since the accident can occur only in the event of four simultaneous equipment failures and operating errors.

These failures and errors are:

a. The controller malfunctions and raises the moderator level at the maximum rate;

b. The period scram trip fails to scram the reactor;

c. All of the shim rods are removed from the core; and

d. The power level scram trip is not set for startup, 0.15 P.

Even in this case the temperature of the Pu-Al cores is above the melting point of aluminum for only 1.5 sec. In this short time the molten Pu-Al would not diffuse through the Zircaloy jackets. Since none of the credible nuclear excursions resulted in melting the Zircaloy jackets no metal water reaction should result from nuclear excursions in the PRTR.

3. Loss of Coolant

A study of possible loss-of-coolant accidents in the PRTR concluded that none of the accidents, ranging from large to relatively small coolant system ruptures, would result in fuel element melting if the automatically actuated light water coolant injection system operated.[33] Loss of the primary coolant in the PRTR could result in fuel element melting only if the back-up light water injection system also failed.

Loss of the primary coolant accompanied by the simultaneous failure of the back-up light water injection system is analyzed as the maximum credible accident in Section VII. The potential for metal-water reaction is analyzed there.

(33) Lemmon, A. W. et al. Ibid. BMI-1356 page 168.

F. Tritium Hazard

Tritium will be formed in appreciable quantities in the heavy water in the PRTR. In the moderator, reflector, and primary coolant tritium will be formed by n, γ reactions of deuterium. In the primary coolant tritium will also be formed by n, α reactions of Li^6 in the LiOH used to control the pH.

A biological hazard does not exist from tritium, as tritium oxide, contained in the reactor systems, since the piping and vessel walls shield personnel from the soft beta radiation. Absorption through the skin, inhalation, or ingestion of tritium does pose a serious biological hazard. Use of protective clothing and adherence to radiation zone work procedures will protect personnel against absorption of tritium through the skin and inhalation of tritium. An analysis of the tritium concentration in the heavy water, potential heavy water leaks, process area ventilation, and personnel access to process areas is necessary to place the tritium hazard in proper perspective.

Leakage of primary coolant into the secondary coolant could be a potential source of river contamination. Analysis of this problem is necessary to insure that leakage detection provisions are adequate to protect the public.

1. Tritium Concentration in Reactor Systems

The concentration of tritium in the moderator, reflector, and primary coolant has been calculated.[34] It was assumed that the annual leakage rate would be 500 lb/yr from the moderator and 4500 lb/yr from the primary coolant. Replenishment of the heavy water lost from these systems was assumed to be accomplished by transferring 4500 lb/yr of heavy water from the moderator to the primary coolant and adding 5000 lb/yr of unirradiated heavy water to the moderator. The specific tritium activities

(34) Regimbal, J. J. and W. H. Hayward. Tritium Generation in D_2O Systems of PRTR, HW-61151. July 20, 1959.

in the primary coolant, moderator, and reflector are shown on Figure 61. The reflector specific tritium activity is greatest because none of the make-up unirradiated heavy water is added to the reflector system.

2. Coolant System Leaks

Two of the three systems in which heavy water is used, moderator and reflector systems, are operated at low temperatures and pressure, 149 and 160 F and essentially atmospheric pressure. Even though the specific tritium activities of the moderator and reflector are higher than the activity of the primary coolant it is estimated that these systems will contribute little to air borne tritium contamination, since the operating temperatures and pressures are so low.

In a study of the primary coolant system leak problem 1378 joints were recognized as potential sources of leakage.[35] These potential leak points are concentrated in three general locations:

(a) The upper reactor access space (340),

(b) The lower reactor access room (425), and

(c) The instrument cell. (596)

The most frequently opened and most difficult to tighten seals, the nozzle caps and process tube to nozzle connection, are in the upper reactor access space, hence leaks are most likely to occur here. The next most likely area for leaks is the lower access room. Although the instrument cell has a large number of primary coolant connections, they are largely 3/16 in. tubing fittings exposed to a low temperature and should be less prone to leak than the high temperature seals.

Foley and Fox estimated that leakage from the primary coolant system can be 1 to 5 gal/hr. Since a leakage rate of this magnitude is intolerable for health and economic reasons, a D_2O recovery system is provided to recover primary coolant leaking into the upper and lower reactor access spaces.

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(35) Foley, D. J. and J. C. Fox. PRTR Primary Coolant Leak Study, Leaks Smaller than Injection Pump Capacity (30 gpm). HW-55738. April 18, 1958.

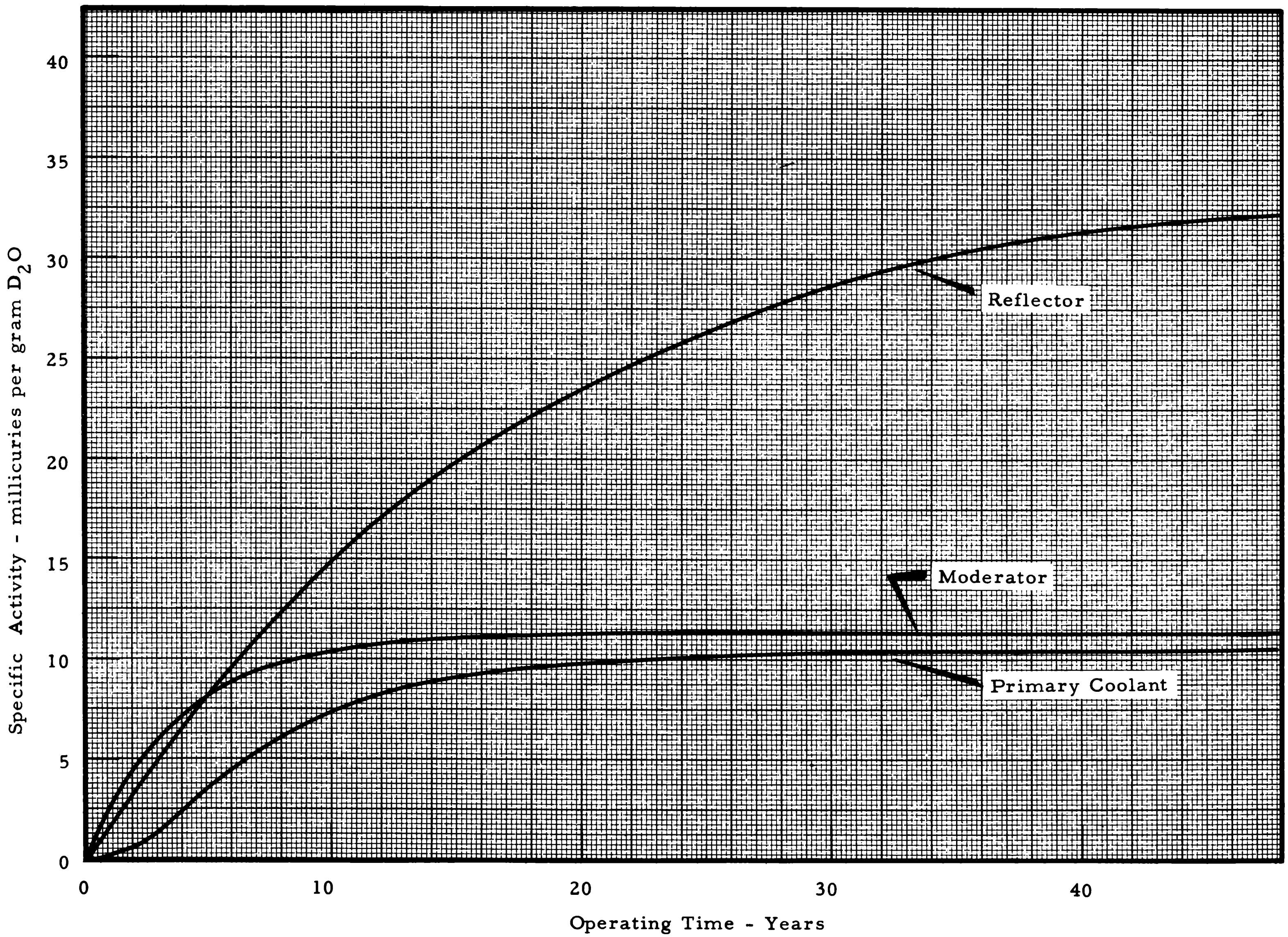

FIGURE 61

Specific Tritium Activity in Reactor Systems

Both the upper and lower access spaces are sealed from the surrounding areas to maintain a closed recirculating air system. The upper access space is effectively enclosed by the side biological shield, top shield, and the rotating disc shield. A rupture diaphragm seals the opening in the side biological shield. The lower access space is enclosed by the thermal barrier, side biological shield, and bottom shield.

Recovery of D_2O leakage in the upper and lower access spaces is accomplished by continuously recirculating air through the spaces. The moisture picked up is removed in a cooler-condenser and drained to a storage tank. Air is recirculated at a rate of 1200 cfm, 600 cfm through each access space. The recovery system is designed to limit the unrecoverable D_2O loss from the primary coolant to 4500 lb/yr.

3. Ventilation System

Ventilation air flow through the containment building will be at a rate of 5000 cfm. Fresh air enters the building in the reactor hall, flows down through the instrument cells and hot shop, and into the process and test cells where it is exhausted to the stack. The direction of air flow is from areas of least contamination to areas of potentially higher contamination.

4. Tritium Monitoring

Monitoring of the air in the containment building for tritium is accomplished by a Kanne chamber which draws samples from six points. Air sampling points are located as follows:

Process Cell	Upper reactor access space
	Lower reactor access space
	Near the ceiling
Test Cell	At a point about 7 ft above floor
Instrument Cell	Near ceiling in upper level
	Near ceiling in lower level.

A sample is drawn from each of the six points every three minutes.

5. Concentration in Building Air

Figure 61 shows that the tritium concentration in the primary coolant will be 0.0075 curies/gm D_2O after 10 years of operation. A 0.1 gal/hr leak would release 2.130 curies/hr of tritium. The D_2O vapor content of the building air between the leak point and the ventilation exhaust line will increase until this amount of tritium can be carried away by the ventilation system. With the 5000 cfm ventilation air flow rate the tritium concentration in the air would reach 2.5×10^{-4} $\mu c/cc$. This is about 13 times the maximum permissible limit, 2×10^{-5} $\mu c/cc$ for breathing air. When tritium is detected in the air, personnel who must enter the contaminated zone will wear fresh air masks and protective clothing. Atmospheric dilution of tritium releases of this magnitude in the exhaust ventilation air will reduce the concentration well below the maximum permissible concentration. Therefore there should be no overexposure of off-site personnel.

6. River Contamination

The possibility of leakage of primary coolant into the secondary coolant in the steam generator and other heat exchangers was recognized. The secondary coolant activity monitor, provided to detect such leakage, will detect about 1 ppm D_2O in the secondary coolant. This is equivalent to a leak rate of $\sim$1 gal/day.

If 1 gal/day of primary coolant were discharge to the river when the river flow rate was at the seasonal low, 25,000 cfs, the tritium concentration of the river water after uniform mixing would be 3.4×10^{-7} $\mu c/cc$. The maximum allowable tritium concentration in drinking water for non-occupational use is 3×10^{-3} $\mu c/cc$. The secondary coolant activity monitor will be set to alarm when the D_2O leakage rate into the secondary coolant reaches 10 gal/day. The reactor will be shut down manually if the secondary coolant activity monitor alarms.

G. Critical Mass Considerations

Fuel element storage facilities were designed to prevent the attainment of a critical array of fuel elements. Analyses have verified the nuclear safety of the fuel element storage design.[36]

1. Unirradiated Fuel Storage Pit

In the unirradiated fuel storage pit a 9 x 5 array of fuel elements can be stored with a center-to-center distance of 8 in. These 45 fuel elements are subcritical in air and subcritical even if the storage pit were filled with light water.

2. Fuel Transfer Pit

In the fuel transfer pit 10 irradiated fuel elements can be stored under water on two rows of supports. The rows are spaced on 33-inch centers. Fuel elements in a row are spaced on 9-in. centers. One fuel element can be in motion between the rows at the center line between rows. Under these conditions the array will be subcritical.

3. Storage Basin

Fuel elements are stored under water in the storage basin in 17 rows of 12 elements per row. The center-to-center distance between rows is 10 in. and between elements within a row is 8 in. This array will be subcritical.

H. Disposal of Wastes

1. Aqueous Wastes

A process waste disposal system is provided to handle radioactive wastes. The system consists of the following facilities:

a. Three hot waste holdup tanks complete with transfer pumps, sampling and monitoring facilities.

b. A valve box structure for diversion of the wastes to the river or truck out station.

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(36) Ketzlach, N. PRTR Fuel Element Nuclear Safety, HW-59786. March 30, 1959.

Three 5000-gal hot waste holdup tanks of stainless steel construction, buried under 6 ft of earth for radiation protection, are provided. Each tank has an 18-in. diameter pump well fitted with an electric motor driven vertical centrifugal deep well pump. The capacity of two of these pumps is 75 gpm at 45 ft TDH and for the third pump 34 gpm at 72 ft TDH. The pump discharge line will be provided with sample cock and flushing connections. Each tank is provided with a 4-in. diameter sample riser and vent, complete with vent filter. An 8-in. schedule 40 pipe blind well is provided adjacent to each tank for future radiation monitoring facilities.

A valve box structure, adjacent to the waste holdup tanks, is provided to route wastes as follows:

a. to the river,
b. to the truck out station,
c. to another holdup tank, and
d. to recirculate.

The process sewer is a 12 in., schedule 20, carbon steel line which discharges into the barometric condenser effluent line.

2. Gaseous Wastes

The ventilation air flow in the containment vessel is directed to pass from areas least likely to become contaminated to those progressively more likely to contain radioactive gases or particles. The ventilation air is exhausted through a 30-in. diameter line to a 150-ft high stack. A 12-in. process vent line connects B cell, in which special test loops will be located, to the stack. Both lines are provided with containment valves. Normal flow through the stack will be 5000 cfm, with up to 10,000 cfm capacity available for purging of the containment vessel.

An HM chamber monitors the exhaust air activity. An activity level of $4.5 \times 10^{-8} \mu c\ \gamma/cc$ will trip the ventilation containment circuit, and will also result in a reactor scram.

I. Sabotage and Non-Nuclear Incidents

1. Sabotage

The facility will be manned with operating personnel during all operating periods, and except for emergency evacuations, the building will be attended at all times until the reactor is removed from service. The doors to the service building as well as the emergency air lock are equipped with locks to prevent unauthorized entry.

The main air lock is equipped with indicating lights in the control room to warn the operators of its use and the air lock door is provided with a key lock.

The process and experimental cells are equipped with key locks and, in addition, have electric locks which may be opened from the control room only.

All critical process switches and valves are equipped with suitable lock switches and guards. Keys for all locks will be controlled by the operating staff.

2. Bombing

No protection against bomb and external missiles is incorporated in the design of the PRTR building.

3. Earthquake

The PRTR building and equipment are designed to resist forces associated with earthquake conditions for Zone 2 as defined in the Uniform Building Code.

4. Windstorm

The containment vessel, service building, and other structures are designed and constructed to resist wind pressures in accordance with the Uniform Building Code section 2307.

5. Floods

The reactor hall floor is 49 ft above the estimated average flood stage of the Columbia River. In fact, the reactor hall floor is 35 ft above the estimated 100-year maximum flood stage. The site, therefore, offers safety from natural Columbia River floods.

J. Interaction of PRTR and PFPP

The PFPP reflects the Hanford experience of over 12 years of continuous safe handling of large quantities of plutonium. It is believed that in some respects the building and equipment offer improvements in safety over those with which HAPO's record has been set.

It must be recognized that the pilot plant operations will bring a contamination control problem which has previously been isolated in the outer areas to the 300 Area. This problem is the presence of plutonium in quantity. Quantities of plutonium handled in the 300 Area previously have been so small that spread of uranium contamination could be tolerated on occasion without worry that it contained plutonium.

The PFPP is sufficiently distant from existing 300 Area buildings and the PRTR that no damage to the PFPP from accidents arising elsewhere is expected. Also, it is expected that an incident in the PFPP will in no way cause incidents in or damage to the other buildings. Because of the distance between the PFPP and the PRTR (about 100 yards), as well as the shielding of the reactor, no neutron interaction between these two facilities can be possible.

VII. MAXIMUM CREDIBLE ACCIDENT

VII. MAXIMUM CREDIBLE ACCIDENT

A. General

In Section VI, Safety Analysis, possible nuclear excursions and equipment malfunctions, including loss-of-coolant accidents, were analyzed. In no credible case did a nuclear excursion lead to melting of the core with consequent release of fission products to the containment vessel. For the loss-of-coolant accidents fuel element melting was prevented by the automatic injection of backup light water coolant.

A major accident, one resulting in a large release of fission products to the containment vessel, could occur only if a number of simultaneous failures were compounded. The most damage to the reactor would be caused by non-nuclear accidents such as loss of the primary coolant accompanied by loss of the backup light water coolant.

B. Course of Maximum Credible Accident

1. Events Leading to Accident

The maximum credible accident* in the PRTR is the rupture of the top 14-inch header accompanied by the simultaneous loss of backup coolant. This accident could occur as a result of the following chain of events:

a. A ductile failure of the top 14-inch header ruptures the pipe to give a hole of 12 inches equivalent diameter (0.80 sq ft). The failure could be the result of an undetected flaw, possibly caused by thermal stress.

b. The backup light water coolant system fails as a result of:

(1) The boiler feed pump, process water pump, and deisel driven well pump fail, any of which can supply injection water;

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* The maximum credible accident is defined as a foreseeable accident based on a chain of events.

(2) The light water injection valves in the two injection lines fail to open; or

(3) Both light water injection lines are ruptured.

It is emphasized that this accident can occur only if three or four simultaneous failures occur in independent systems, depending on the failure mechanism of the backup coolant system.

2. Blowdown of Primary Coolant

Blowdown of the primary coolant through an 0.80 sq ft rupture in the top 14-inch header has been calculated.[37] The blowdown flow rate transient is shown on Figure 50, Page 177.

The initial discharge rate is about 5000 lb/sec decreasing rapidly to 1150 lb/sec in 0.5 sec. The discharge rate then decreases slowly to 400 lb/sec at 26.5 sec when the blowdown is virtually complete. Steam would continue to flow until 42 seconds after the incident. At 42 seconds the pressure in the reactor would reach 24 psia.

If the rupture should occur in the top reactor access space, a differential pressure would exist across the top rotating shield. The vent from the top reactor access space to the process cell varies in cross-sectional area from 42.8 to 47.85 sq ft over a 4.5 ft length. Assuming steady-state conditions at the maximum discharge rate, the pressure in the access space builds up to 13.7 psig.[38] Since the hold-down devices for the rotating shield are designed to withstand an 18 psi differential pressure across the shield, the pressure buildup will not move the shield.

(37) Ambrose, T. W. Ibid HW-60654, page 172.
(38) Muraoka, J. Ibid, HW-60963, page 186.

3. Meltdown of Fuel Elements

Drastic changes in the primary coolant system pressures and flows would trip the safety circuit and shut down the reactor. The fuel temperature excursion following the loss of coolant is shown on Figure 62. Those Pu-Al elements which have been operating at 1200 KW will reach the melting point of aluminum in about 219 seconds after the incident. The 1200 KW UO_2 elements will reach the melting point of zirconium in 740 to 900 seconds.(39)

Meltdown of the reactor core will be progressive, starting at the center of the reactor and gradually extending to the fringes. It has been estimated that the core melting would start at about 3.5 minutes and continue until about 90 minutes after the accident.

4. Metal-Water Reaction

After the blowdown of the coolant is complete, very little water will remain in the primary coolant system. A pool of water will stand in the lower portions of the system with the water level at about the mid-plane of the lower ring header. Fuel elements in the process tubes will be in a stagnant atmosphere of 24 psia steam.

The amount of stagnant steam in the reactor core is sufficient to react with less than 1.0 per cent of the zirconium in the fuel element jackets. As the meltdown of the fuel elements progresses the process tube and shroud tube temperatures will increase until they also will melt. At the bottom of the calandria is a pool of moderator into which the melting process and shroud tubes can fall or slump.

Experimental studies of molten-metal-water reactions by Higgins showed that the percentage completion of these reactions was dependent

(39) Lemmon, A. W., Jr., et al, Ibid, BMI-1356, page 168.

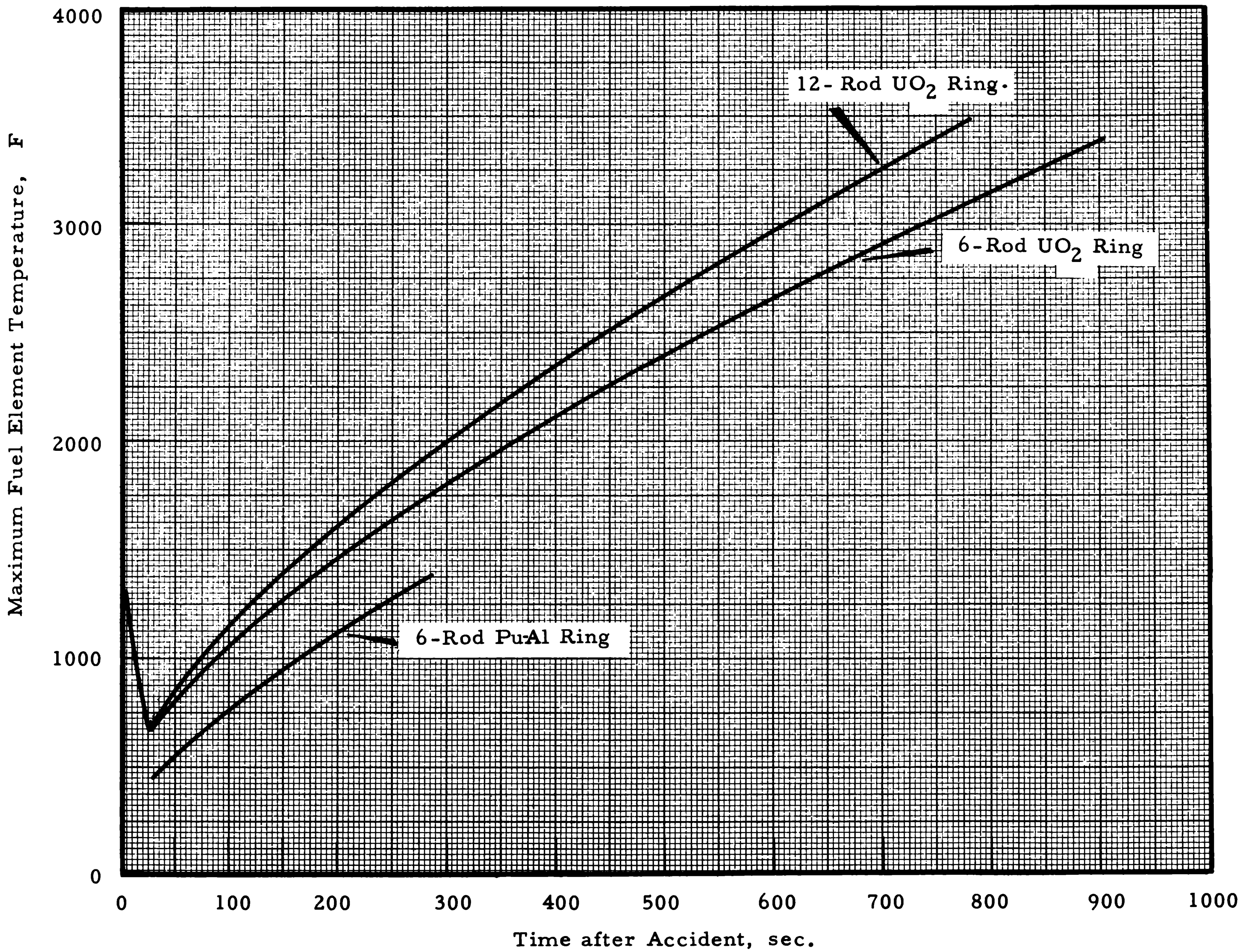

FIGURE 62

Fuel Temperature Rise Following
Maximum Credible Accident
(1200 KW Elements)

upon the particle size and the metal temperature.[40] It was found that the reactivity of molten aluminum was nil at temperatures up to 1170 C, under the conditions of the Aerojet work.

No mechanism for dispersion of the molten metal into small droplets as it runs or falls into the pool of moderator has been postulated. However, for purposes of conservatism, it is assumed that molten zirconium contacting the pool of moderator will react to 5 per cent completion. Since the molten aluminum from the shroud tubes will be at about 660 C there should be no reaction of this material. The Pu-Al cores of the spike enrichment elements will flow out of the Zircaloy-2 jackets when the jackets fail. This molten Pu-Al will run or fall down the empty process tube to an elevation below the bottom of the calandria. It is concluded that the extent of metal-water reaction is as follows:

Zircaloy-2 Fuel Element Jackets	1.0%
Pu-Al Cores	5.0%
Zircaloy-2 Process Tubes	5.0%
Aluminum Shroud Tubes	0.0%

Based on the above percentage reaction, the amount of metal reacting is:

Zircaloy-2	175 kg
Pu-Al	25 kg

When either zirconium or aluminum reacts with water hydrogen is formed as illustrated by the following reactions:

$$Zr + 2\,H_2O \rightarrow Zr\,O_2 + 2\,H_2$$

$$2\,Al + 3\,H_2O \rightarrow Al_2\,O_3 + 3\,H_2$$

(40) Higgins, H. M., Ibid., IDO-28000, page 209.

Since the metal-water reactions would take place in the pool of moderator at the bottom of the calandria, the sensible heat of the metal and the fuel element cores and the heat of formation of the metal oxides would vaporize water. The steam-hydrogen mixture would discharge from the reactor to the building. As the steam is condensed by contacting cool surfaces or because of the water fog spray the steam in the steam-hydrogen mixture would be replaced by air giving a hydrogen concentration in air no greater than the concentration in the steam mixture.

For conservatism, if only 50 per cent of the sensible heat of the reactor core is assumed to vaporize the water in the bottom of the calandria, enough steam would be produced to give a hydrogen concentration of 5 per cent by volume in the steam discharging from the reactor. As the steam is condensed, the resulting hydrogen concentration in the building air would be no greater than 5 per cent by volume.

The lower limit of flamability of hydrogen is 4.0 per cent, but for downward propagation of flame the hydrogen concentration must be 9.0 per cent.(41) For deuterium the corresponding limits are 4.9 and 10.2 per cent. An explosion would not develop in a hydrogen-air mixture when the concentration is less than the downward propagation limit. It is concluded that the hydrogen produced in this incident could conceivably burn, but it would not detonate.

From Coughlin's data, the following heats of reaction were developed:(42)

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2;\ \Delta H = -137,180 \text{ cal/gm mol}$$

$$2Al + 3H_2O \rightarrow Al_2O_3 + 3H_2;\ \Delta H = -219,320 \text{ cal/gm mol}$$

$$2H_2 + O_2 \rightarrow 2H_2O;\ \Delta H = -58,700 \text{ cal/gm mol}$$

(41) Coward, H. F. and G. W. Jones. Limits of Flamability of Gases and Vapors, Bureau of Mines Bulletin 503, 1952.

(42) Coughlin, J. P. Contributions to the Data on Theoretical Metallurgy: XII, Heats and Free Energies of Formation of Inorganic Oxides, Bureau of Mines Bulletin 542, 1954.

The excess energy contribution from the chemical reactions would be:

Zirconium-water reaction		1,000,000 Btu
Plutonium-Aluminum-water reaction		400,000 Btu
H_2 combustion		1,200,000 Btu
	Total	2,600,000 Btu

5. Pressure Transient in the Containment Building

Depressurization of the primary coolant system through the rupture in the top 14-inch header will raise the pressure in the containment building to 9 psig in about 26 seconds. The method of calculating the pressure in the containment building is outlined in Appendix "D".

Following blowdown of the coolant, fission product decay energy and energy from chemical reactions will be released from the reactor core. The energy from metal-water reactions and hydrogen combustion would be released over a period of about 1 1/2 hours. Although the energy from chemical reactions would give a building pressure of 11.7 psig if released in a short time, the pressure will actually be lower because the water fog spray system will cool the atmosphere in the building faster than the energy is generated. The cooling power of the water fog spray system is 37,000,000 Btu/hr. The pressure transient in the containment building following a loss-of-coolant accident is shown in Figure 63.

Since the pressure in the containment vessel is lower than the design pressure, escape of fission products to the atmosphere would be limited to the amount carried by the specified maximum leakage of 1000 cu ft per 24 hours.

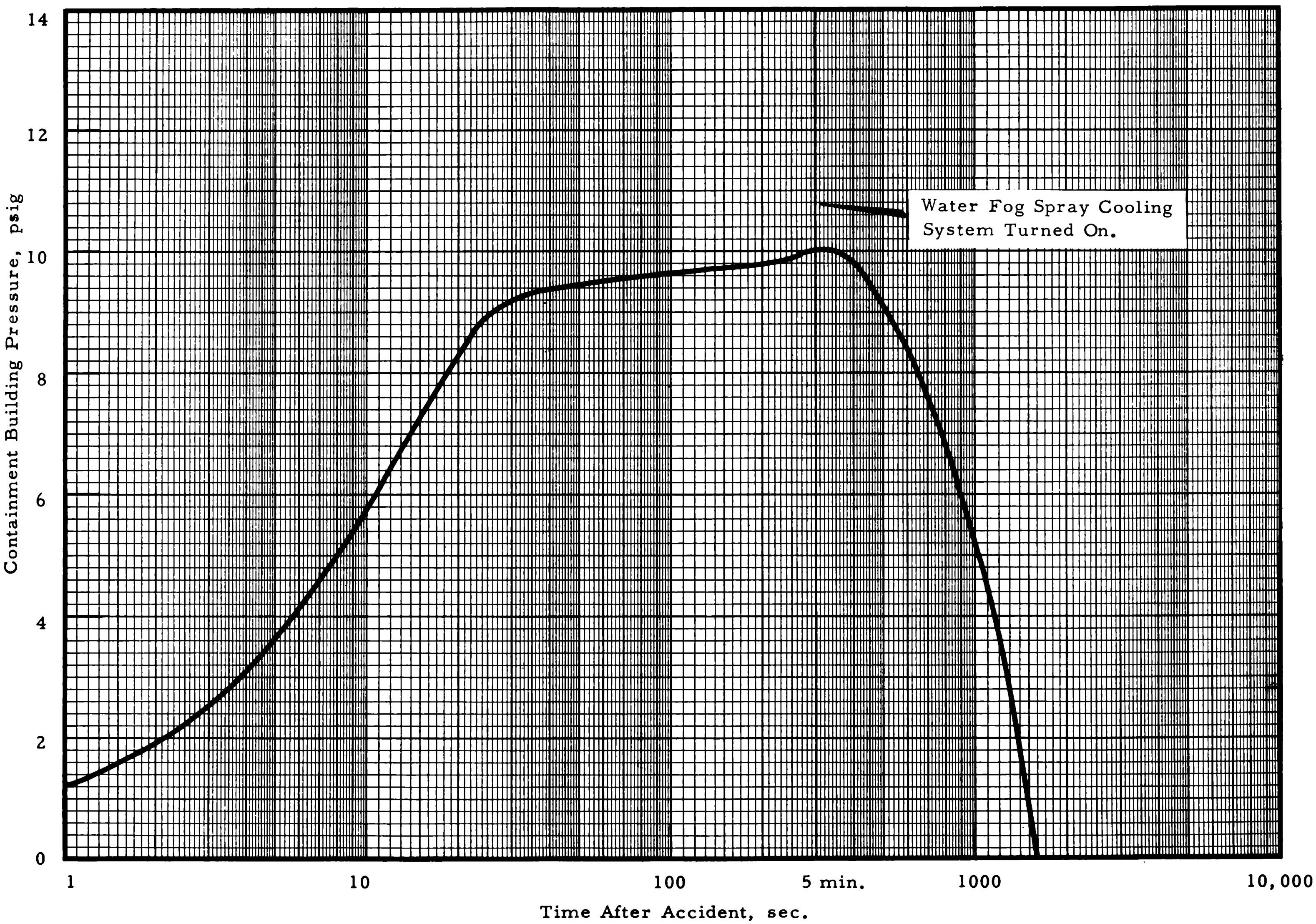

FIGURE 63

Building Pressure Transient after Maximum Credible Accident

VIII. RADIOLOGICAL CONSEQUENCES OF MAXIMUM CREDIBLE ACCIDENT

VIII. RADIOLOGICAL CONSEQUENCES OF MAXIMUM CREDIBLE ACCIDENT

The accident chosen for analysis of the radiological effects consisted of a rapid meltdown with release of a portion of the fission products to the containment shell. Allowance was made for the removal of all except noble gases by the fog spray system. Under these conditions the effects are localized with the bulk of the hazard resulting from the direct gamma radiation from the contained fission products.

A. Inventory of Radioisotopes

Radioisotopes considered to be of importance are the fission products, Np^{239} and the isotopes of plutonium. Previous calculations have indicated that the induced isotopes in the coolant and moderator are negligible in terms of the dose to individuals in the environs even if the entire quantity were released.[(43)]

The quantities of fission products present were assumed to be similar to those produced by the thermal neutron fission of U^{235} with a residence time in the reactor of 500 days. This irradiation time is a maximum estimate and was used because of the developmental aspects of the program which do not allow a firm estimate of the actual operating cycle at this time. The quantities of fission products in each of four categories were obtained by summation of previous computations[(44)] in order to allow estimation of the quantities released. These categories are:

(1) Noble gases - Kr and Xe

(2) Halogens - I

(3) Volatile solids - Cs, Te, Se, Ru

(4) Non-volatile solids - all others

(43) Healy, J. W., to P. C. Walkup. Coolant Loss - PRTR, March 6, 1959.
(44) Healy, J. W., G. E. Pilcher and C. E. Thompson. Computed Fission Product Decay, HW-33414. December 1, 1954.

The amount of Np^{239} was estimated assuming that all of the power was generated in the uranium with a conversion ratio of 0.8 and an equilibrium quantity of Np^{239} present. The quantity of the various isotopes of plutonium is dependent upon the method of recycle as well as the irradiation level. For purposes of calculation and conservatism it was assumed that the reactor contains 35 kg of plutonium, half of which is in the form of plutonium "spike" fuel elements with the other half as enrichment in uranium. All of the plutonium has then been irradiated to 10,000 MWD per adjacent ton. The relative quantities of each isotope were obtained from the calculations of Merrill.[45]

The estimated quantities of radioisotopes present are given in Table VII.

B. Volatilization of Radioisotopes

The release of fission products from heated fuel has been studied in the laboratory at ORNL[46], MSA[47], and Hanford.[48] The results of many of these tests are spotty with inadequate experiments, as yet, to completely characterize the variables involved.

Table VIII summarizes some of the information available from these experiments giving the range of values noted in the release of selected isotopes from uranium fuel elements, stainless steel clad elements and zirconium elements. Data for air and steam atmospheres and for high or low burnup are included.

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(45) Merrill, E. T. Dose Rates from Highly Irradiated Plutonium, informal report.

(46) Creek, G. E., W. J. Martin and G. W. Parker. Experiments on the Release of Fission Products from Molten Reactor Fuels, ORNL-2616, December, 1958.

(47) Rodgers, S. J. and G. E. Kennedy. Fission Product Release During a Simulated Meltdown of a PWR Type Core, Technical Report 63, MSA, October 20, 1958.

(48) Linderoth, C. E. Interim Report - Fission Product Volatilization Studies, HW-59431-A, March 6, 1959 (HW-59431-B, March 25, 1959; HW-59431-D, July 6, 1959).

TABLE VII

QUANTITIES OF RADIOISOTOPES IN 70 MW REACTOR

(Curies)

Decay Time	10 min	1 hr	10 hr	1 day	10 days	30 days	100 days
Noble Gases	1.6×10^7	1.3×10^7	8.4×10^6	6.2×10^6	1.6×10^6	1.3×10^5	1.4×10^4
Halogens	1.5×10^7	1.4×10^7	9.1×10^6	6.2×10^6	1.1×10^6	1.5×10^5	3.7×10^2
Vol. Solids	1.8×10^7	1.1×10^7	5.3×10^6	4.7×10^6	2.5×10^6	1.6×10^6	6.7×10^5
Non-vol. Solids	1.2×10^8	8.4×10^7	5.5×10^7	4.8×10^7	2.9×10^7	2.0×10^7	9.8×10^6
Total FP	1.7×10^8	1.2×10^8	7.7×10^7	6.4×10^7	3.4×10^7	2.2×10^7	1.1×10^7
Np^{239}	4.7×10^7	4.6×10^7	4.2×10^7	3.5×10^7	2.3×10^6	5.6×10^3	→
Pu^{239}	8×10^2	→					
Pu^{240}	4×10^3	→					
Pu^{241}	5×10^5	→					
Pu^{242}	20	→					

TABLE VIII

FISSION PRODUCT RELEASE DATA

Atmosphere	Burnup	I (%)	Noble Gases (%)	Cs (%)	Sr (%)
Uranium					
Air	Trace	2 - 90	8 - 80	0.1 - 6	0.001 - 0.18
	High	1 - 80	100	2 - 60	0.05 - 16
Steam	Trace	0.3 - 23	0.2 - 6	0.03 - 2	0.01 - 0.6
	High	---	---	---	---
Stainless					
Air	Trace	0.4 - 34	16 - 48	0.4 - 11	0.006 - 0.03
	High	4 - 18	16 - 50	6 - 72	---
Steam	Trace	11	39	0.3	0.4
	High	---	---	---	---
Zirconium					
Air	Trace	1 - 8	42 - 50	0.41	0.6 - 1.4
	High	14 - 32	100	7 - 13	0.2 - 0.9
Steam	Trace	26	58	0.48	1.1
	High	47	100	19	1.8

This simple table cannot characterize the entire pattern of release but it does summarize the magnitude of the results obtained. In general, the time of heating and the temperature play important roles. For example, a plot of the information available on strontium release indicates a general increase in fraction released from about 0.005% at 500 C to about 1% at 1900 C. Several anomalous points for strontium have been obtained by Parker with uranium at 1200 C and heating times of five to twenty minutes.[49]

(49) Parker, G. W., W. J. Martin and G. E. Creek. Fuel Element Catastrophe Studies: Hazards of Fission Product Release from Irradiated Hanford Uranium, June 19, 1959.

The highest value here (16%) was attributed to the production of oxide particles which carried the strontium by trapping.

In order to permit calculations on the consequences of accidents, a general formula has been devised by the GE Reactor Safeguards Council utilizing these data.(50) This formula is given in Table IX.

TABLE IX

RELEASE FORMULA

Fuel Containing 50% of Fission Products Melts

Release from Fuel	
Noble Gases	100%
Halogens	50%
Volatile Solids	50%
Non-volatile Solids	1%
Release from Reactor Structure	
Noble Gases	100%
Halogen	50%
Volatile Solids	30%
Non-volatile Solids	30%

The factors for release from the reactor structure are included to make allowance for the retention on cold surfaces by condensation or on tortuous passages by impaction, etc.

Only the factors for release from the fuel were applied in the calculation of the quantities escaping from the reactor to the containment vessel.

The estimated quantities of fission products released to the atmosphere of the containment vessel are given in Table X.

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(50) Cohen, K. Private communication.

TABLE X

RADIOISOTOPES RELEASED TO THE CONTAINMENT VESSEL

(Curies)

Decay Time	10 min	1 hr	10 hr	1 day	10 days	30 days	100 days
Noble Gases	7.7×10^6	6.2×10^6	4.1×10^6	3.1×10^6	7.7×10^5	6.7×10^4	7.0×10^3
Halogens	3.9×10^6	3.6×10^6	2.2×10^6	1.5×10^6	2.7×10^5	3.8×10^4	90
Vol. Solids	4.4×10^6	2.7×10^6	1.3×10^6	1.2×10^6	6.0×10^5	4.0×10^5	1.7×10^5
Non-vol. Solids	5.9×10^5	4.3×10^5	2.7×10^5	2.4×10^5	1.5×10^5	1.1×10^5	5.0×10^4
Total FP	1.7×10^7	1.3×10^7	7.7×10^6	6.0×10^6	1.8×10^6	6.1×10^5	2.2×10^5
Np^{239*}	2.4×10^5	2.3×10^5	2.1×10^5	1.8×10^5	1.2×10^4	2.8×10^2	→
Pu^{239*}	4	→	→	→	→	→	→
Pu^{240*}	20	→	→	→	→	→	→
Pu^{241*}	2500	→	→	→	→	→	→
Pu^{242*}	0.1	→	→	→	→	→	→

* The release fractions of neptunium and plutonium were taken to be those of the non-volatile solids.

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C. Gamma Energy

The gamma spectrum for the total fission products was derived from the tabulations used to estimate the quantities of fission products.[51] The spectra were derived in 14 energy groups in units of photons per second per watt of reactor power. The energies of the isotopes were obtained from the tabulations of the National Bureau of Standards[52] supplemented by the current data given in Nuclear Science Abstracts.[53] The total energy release for each of the energy groupings was obtained by multiplying the number of photons per second by the average energy of the group. The 14 groups were then combined into the six groups reported in order to facilitate the calculations.

This procedure was carried out separately for each of the release categories and the appropriate release factors applied. Data for the period up to one day after shutdown were not obtained for the non-volatile solids because of the uncertainty engendered by the large fraction of short-lived isotopes whose half-lives and energies are not well known. Since this fraction accounted for only 2 per cent of the total energy release at one day, 4 per cent at ten days and 9 per cent at 100 days, its omission was judged to be unimportant within the error of the estimates.

The energy data was summed for the halogens, volatile solids and non-volatile solids and retained separately for the noble gases. These data are presented in Table XI.

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(51) Healy, J. W., G. E. Pilcher and C. E. Thompson. Computed Fission Product Decay, HW-33414. December 1. 1954.

(52) United States Department of Commerce, National Bureau of Standards. Nuclear Data, NBS circular 499. September 1, 1950 (Supplement 1, April 25, 1951; Supplement 2, November 26, 1951).

(53) Nuclear Science Abstracts. New Nuclear Data 1952 Cumulation, Vol. 6, 24B. December 31, 1952 (1953 Cumulation, Vol. 7, 24B. December 31, 1953; 1954 Cumulation, Vol. 8, 24B. December 31, 1954; 1955 Cumulation Vol. 9, 24B. December 31, 1955; 1956 Cumulation, Vol. 10, 24B. December 31, 1956).

TABLE XI

GAMMA ENERGY RELEASE OF FISSION PRODUCTS IN CONTAINMENT SHELL

(Mev/second)

Decay Time	10 min	1 hr	10 hr	1 day	10 days	30 days	100 days
Energy - Mev	Halogens plus Solids						
0 - 0.19	3.6×10^{15}	2.1×10^{15}	1.4×10^{15}	1.3×10^{15}	6.1×10^{14}	4.0×10^{14}	1.3×10^{14}
0.2 - 0.59	3.9×10^{16}	3.8×10^{16}	3.3×10^{16}	2.7×10^{16}	1.0×10^{16}	5.8×10^{15}	1.8×10^{15}
0.6 - 0.99	4.0×10^{16}	3.6×10^{16}	3.2×10^{16}	2.9×10^{16}	5.8×10^{15}	1.9×10^{15}	1.5×10^{15}
1.0 - 1.79	3.9×10^{16}	3.6×10^{16}	1.8×10^{16}	8.4×10^{15}	1.1×10^{15}	2.0×10^{14}	6.0×10^{13}
1.8 - 2.49	6.6×10^{15}	6.2×10^{15}	4.0×10^{15}	2.8×10^{15}	4.4×10^{14}	4.4×10^{13}	2.2×10^{13}
2.5 - 3	----	----	----	1.8×10^{12}	1.2×10^{12}	3.9×10^{11}	9.3×10^{10}
	Noble Gases						
0 - 0.19	8.6×10^{15}	8.4×10^{15}	7.1×10^{15}	6.7×10^{15}	2.2×10^{15}	1.6×10^{14}	----
0.2 - 0.59	4.3×10^{16}	4.3×10^{16}	3.6×10^{16}	2.8×10^{16}	7.4×10^{15}	5.5×10^{14}	----
0.6 - 0.99	2.6×10^{15}	2.2×10^{15}	2.7×10^{14}	9.8×10^{12}	----	----	----
1.0 - 1.79	----	----	----	----	----	----	----
1.8 - 2.49	1.4×10^{16}	1.2×10^{16}	1.5×10^{15}	5.6×10^{13}	----	----	----
2.5 - 3	9.9×10^{14}	8.1×10^{14}	1.0×10^{14}	----	----	----	----

D. Dose Rates from Container

From the dimensions of the containment vessel, it was estimated that 38 per cent of the fission products released would be contained in the hemispherical shell; 50 per cent would be contained in the cylindrical portion of the vessel with steel and one foot of concrete shielding; and the remaining 12 per cent would be below grade where the shielding of the earth would eliminate their contribution.

The dose rate from each of the energy bands was calculated assuming an inverse square decrease with distance and allowing for the absorption and buildup in the steel, concrete and air.

$$\text{Dose rate} = \frac{\text{Mev/sec}}{4\pi r^2} e^{-(\mu_A r_A + \mu_s r_s + \mu_c r_c)} \left[\frac{\mu - \sigma_s}{\rho} \quad \frac{B(\mu x)(3600)}{6.24 \times 10^5 \times 87.7} \right] \quad (1)$$

r - distance from the center of the containment vessel to point at which the dose rate was calculated;

μ_A, μ_s, μ_c - absorption coefficients in air, steel and concrete respectively;

r_A, r_s, r_c - thickness of air, steel and concrete effective as absorber;

$\frac{\mu - \sigma_s}{\rho}$ - electronic absorption coefficient in air;

$B(\mu x)$ - buildup factor;

3600 - seconds per hour;

6.24×10^5 - Mev per erg;

87.7 - ergs per gram of air per roentgen.

Calculations were performed separately for the material in the hemispherical portion of the containment vessel and then summed. Buildup coefficients were as given in the Radiological Health Handbook[54], and absorption coefficients were obtained from the tabulations by Snyder.[55]

(54) Federal Security Health Agency, Public Health Service. Radiological Health Handbook. March, 1953.

(55) Snyder, W. S. and J. L. Powell. Absorption of Gamma Rays, ORNL-421. March 14, 1950.

The design of the containment vessel provides for the introduction of 500 gallons per minute of water in the form of 50 micron fog particles immediately following the incident. Although the fog is primarily intended to lower the temperature and pressure in the containment vessel, it will also serve to wash at least part of the fission products from the air. The introduction of 500 gallons of water per minute into a container 80 feet in diameter will provide a total of 9.6 inches of precipitation per hour as it condenses and settles. Chamberlain(56) has estimated that 0.2 inches of natural rainfall per hour will remove 1×10^{-3} per second of a suspended aerosol with particle size of 25 microns and 7×10^{-4} per second of particles five microns in diameter. In view of the incomplete sweeping of the volume of the containment vessel by the condensed fog and the unknown efficiency of sweeping by the fog particles, a value of 2×10^{-4} per second for removal of the suspended aerosol was adopted. The same value was used for removal of the halogens since Chamberlain's work again indicated a removal rate of 2×10^{-4} per second for iodine vapor with a rainfall of 0.2 inches per hour. For conservatism, it was assumed that no washout would occur in the first ten minutes.

Table XII presents the results of the calculations before correction for washout.

The estimated values corrected for washout of the solids and halogens are given in Table XIII.

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(56) United States Department of Commerce. Weather Bureau, Meteorology and Atomic Energy. July, 1955.

TABLE XII

DOSE RATES AROUND CONTAINMENT VESSEL NOT CORRECTED FOR WASHOUT

(Roentgens per hour)

Decay	Noble Gases					Solids plus Halogens				
Time	100'	330'	660'	1600'	2300'	100'	330'	660'	1600'	2300'
10 min	500	40	5	0.1	0.02	800	60	10	0.3	0.05
1 hour	440	30	5	0.1	0.02	750	50	9	0.3	0.05
10 hours	320	20	4	0.06	0.01	560	40	7	0.2	0.03
1 day	250	16	3	0.04	0.01	440	30	5	0.1	0.02
10 days	70	4	0.7	0.01	0.002	120	8	1	0.03	0.006
30 days	5	0.3	0.05	$<10^{-3}$	$<10^{-3}$	60	4	0.6	0.01	0.002
100 days	$<10^{-3}$	$<10^{-3}$	$<10^{-3}$	$<10^{-3}$	$<10^{-3}$	20	2	0.3	0.006	0.001

TABLE XIII

DOSE RATES AROUND CONTAINMENT VESSEL

(Roentgens per hour)

Decay Time	100'	330'	660'	1600'	2300'
10 min	1300	100	15	0.4	0.07
1 hr	800	60	10	0.3	0.05
10 hr	320	20	4	0.06	0.01
1 day	250	16	3	0.04	0.01
10 days	70	4	0.7	0.01	0.002
30 days	5	0.3	0.05	$<10^{-3}$	$<10^{-3}$

E. Escape from Containment Vessel

The specifications for the containment vessel require that the leakage be less than 1000 scf per day. At this maximum leakage, the fraction of the contents lost per day would be 2.7×10^{-3}. From the noble gas inventory in the shell, this would permit the loss of 8000 curies per day after 24 hours, 2000 curies per day after ten days and 20 curies per day after 100 days. Since the shell will be cooled to atmospheric pressure shortly after the accident, releases should be even lower than this.

The evaluation of the dose rates contributed by the noble gases leaking from the shell was made under the most stringent atmospheric conditions: a strong inversion with a wind speed of one meter per second. The concentrations downwind were evaluated from previous computations[57] with the dose rate from a finite cloud calculated by the method presented by KAPL.[58] These values are presented in Table XIV.

(57) Healy, J. W. Calculations on Environmental Consequences of Reactor Accidents, HW-54128. December 11, 1957.

(58) Fitzgerald, J. J., H. Hurwitz and L. Tonks. KAPL-ADM-867, Supplement 1. September 11, 1953.

TABLE XIV

DOSAGE RATES DUE TO ESCAPING NOBLE GASES

(mr/hr)

Time Since Incident (days)	100'	330'	660'	1650'	2200'
1	30	15	13	8	7
10	7	4	3	2	2
30	6	<1	<1	<1	<1
100	1	<1	<1	<1	<1

These values are peak dosage rates which will be reached in the cloud. Wind shifts and changes in stability will reduce the average over a long period to values well below these.

The solid fission products will escape by leakage only as long as they are air-borne. The total quantity escaping was evaluated from equation 2.

$$Q = EA\int_0^{\infty} e^{-Wt}\,dt + EA(10) \tag{2}$$

Q - Quantity escaping

A - Quantity initially present in the shell

E - Rate of escape from the shell

W - Washout rate

10 - Time between incident and start of washout

Table XV indicates the leakage of these materials which will occur before they are washed from the atmosphere of the containment vessel.

TABLE XV

MATERIALS LEAKING FROM THE CONTAINMENT VESSEL

(Curies measured at one day)

Halogens	260
I^{131}	40
Solids	250
Np^{239}	30
Pu^{239}	7×10^{-4}
Pu^{240}	3×10^{-3}
Pu^{241}	0.4
Pu^{242}	2×10^{-5}

The release of this quantity of materials at ground level could result in technical overexposure to people but little real damage. Evaluation assuming release in unfavorable meteorological conditions (strong inversion with wind speeds of one meter per second) and assuming a point source, would indicate external doses on the order of 10 - 20 mr at distances of 100 feet. Thyroid doses would be on the order of 40 rads for individuals in the narrow plume at a distance of 600 feet, 9 rads at a distance of 2500 feet and 2 rads at 6600 feet. Gastrointestinal tract doses resulting from the swallowing of inhaled material would be 25 rads at 660 feet, 6 rads at 2500 feet and 2 rads at 6600 feet. The actual doses should be considerably lower than this, however, due to the turbulence engendered by the containment vessel and the additional mixing caused by this turbulence.

Decontamination or limitation of access to areas in the path of the plume will be required as well as some control of grazing of dairy herds. Decontamination is expected to be required in areas contaminated to levels greater than about 10 $\mu c/m^2$ ($\sim$2000 $d/m/cm^2$) and some limitation on grazing in areas with more than one μc iodine per square meter. Maximum distances to which this will occur assuming a point source emission at ground level are given in Table XVI along with the estimated width of the plume at a distance one-half of the way from the reactor to the maximum distance at which restrictions will be required.

TABLE XVI

MAXIMUM DISTANCES OF RESTRICTIONS AND PLUME WIDTH

Stability	Windspeed m/sec	Decont. or Limitation of Access: Distance feet	Decont. or Limitation of Access: Width feet	Limitation of Grazing: Distance feet	Limitation of Grazing: Width feet
Strong Inversion	1	13,000	550	38,000	1,400
	5	20,000	600	32,000	1,600
	10	22,000	660	31,000	1,500
Moderate Inversion	1	7,200	300	41,000	1,400
	5	13,000	500	39,000	1,600
	10	13,000	500	36,000	1,600
Neutral	1	3,000	600	12,000	1,800
	5	3,000	640	12,000	1,800
	10	3,000	640	12,000	1,800
Unstable	1	1,600	720	6,600	2,400
	5	2,100	900	7,200	3,000
	10	2,200	900	7,900	3,200

Again the turbulence resulting from the finite size of the containment vessel will tend to make the plume wider and shorter. Any wind shifts during the time of emission will tend to minimize the distances involved although contaminating larger areas close to the reactor.

APPENDIXES

APPENDIX A

SUMMARY OF PRTR ENGINEERING DATA

1. GENERAL REACTOR DATA

1.	Power level (nominal)	70 MW
	Heat to primary coolant loop	66.4 MW
	Heat to moderator coolant loop	2.2 MW
	Heat to reflector	0.6 MW
	Heat to shields	0.8 MW
2.	Primary coolant flow	8,400 gpm D_2O at 478 F
3.	Primary coolant pressure	1025 psig @ pressurizer
4.	Reactor inlet temperature	478 F
5.	Reactor outlet temperature (bulk)	530 F
6.	Number of process tubes	85
7.	Lattice	8 in. equil. triangle
8.	Fuel loading - Number of tubes of UO_2	42 - 75
	Number of tubes of Pu	10 - 43
9.	Number effective power tubes	60
10.	Number effective flow tubes	68
11.	Max. tube power	1200 KW
12.	Max. tube flow	123 gpm
13.	Max. tube outlet temperature	542 F
14.	Neutron Flux - thermal, avg.	8.3×10^{13} neut/(cm^2)(sec) radial 7.7×10^{13} neut/(cm^2)(sec) axial
15.	Goal fuel exposure, U	5 - 8000 MWD/T
16.	Xenon override time	Up to 2 hours
17.	Time required for:	
	Shutdown (normal)	10 MW/min
	Cool and depressurize primary coolant system (normal)	2 hours

Pressurize primary coolant system	less than 5 min
Startup (from critical to full power)	
Cold	2 hours (minimum)
Hot	13 MW/min (maximum)

18. Power and Neutron Flux Distribution (Unflattened):

Radial flux dist.	0.72 avg/max (1.39 max/avg)
Vertical flux dist. (7.4 foot Fuel)	0.833 avg/max (1.20 max/avg)

2. REACTOR PIPING

Process Tubes (Figure 5)

1.	Material	Zircaloy-2
2.	Design Stress	14,400 psi
3.	Dimensions I. D. (active zone)	3.250 in. ± 0.010 in.
	Wall thickness (active zone)	0.154 in. ± 0.008 in.
	Length	17 ft 5 in.
	Wall thickness (inlet end)	0.235 in.
	O. D. (inlet end)	2-1/16 in.
4.	Weight	85 lb
5.	D_2O flow area for empty tube (active zone)	8.3 in.^{2} (0.0576 ft^2)
6.	D_2O flow area with cluster element, Mk I, in tube	3.56 in.^{2} (0.0246 ft^2)
7.	D_2O flow area with concentric element, Mk II, in tube	2.27 in.^{2} (0.0158 ft^2)
8.	D_2O flow/tube, maximum	123 gpm
9.	D_2O velocity, empty tube (123 gpm)	4.75 ft/sec
10.	D_2O velocity, 19-rod cluster element (123 gpm)	11.1 ft/sec
11.	D_2O velocity, concentric element (123 gpm)	17.7 ft/sec (avg)
12.	Flow split in concentric element	
	Inner channel	17 per cent
	Middle channel	49.4 per cent
	Outer channel	33.6 per cent

13.	D_2O inventory (85 tubes and nozzles)	450 gal w/19 rod, Mk I
14.	Pressure drop across cluster element	~7 psi
15.	Pressure drop across concentric element	~15 psi

Inlet and Outlet Piping

16.	Jumper material (inlet and outlet)	316 SS
17.	Jumper size (inlet and outlet)	1-3/4 in. O. D. tubing (0.083 in. wall) seamless
18.	Pipe material (outlet)	316 SS
19.	Pipe size (outlet)	1-1/2 in. Sch 10S seamless
20.	Valve type (inlet)	angle
21.	Valve material (inlet)	316 SS
22.	Valve size (inlet)	1-1/2 in.
23.	Ring header material (inlet and outlet)	304 SS
24.	Ring header size (inlet and outlet)	10 in. Sch 100 seamless
25.	Ring header diameter (inlet)	13 ft 0 in.
26.	Ring header diameter (outlet)	13 ft 6 in.
27.	D_2O velocity, jumpers (123 gpm)	20.6 ft/sec
28.	D_2O velocity, ring header, max. (4200 gpm)	19.7 ft/sec
29.	D_2O inventory, inlet jumpers and piping	78 gal
	Outlet jumpers	70 gal
	Inlet ring header	144 gal
	Outlet ring header	144 gal
	Total	436 gal

3. FUEL ELEMENTS

19-Rod Cluster Elements (Figure 6)

1.	Length of active core in element	7 ft 4 in.
2.	Length of element	8 ft 3 in.
3.	Length of hanger	6 ft 2 in.
4.	Length of fuel assembly (hanger and fuel element assembly)	14 ft 5 in.

5.	Bare rod diameter	0.504 in.	
6.	Sheath material	Zircaloy-2	
7.	Sheath thickness	0.030 in.	
8.	Core cross sectional area (19 rods)	3.76 $in.^2$	
9.	Total cross sectional area (19rods) w/o wire wrap	4.76 $in.^2$	
10.	Core weight:		
	UO_2	122 lb	
	Pu-Al	32.8 lb	
11.	Fuel element assembly weight (19 elements):		
	UO_2	160 lb	
	Pu-Al	70 lb	
12.	Hanger weight	40 lb	
13.	Fuel assembly weight:		
	UO_2	200 lb	
	Pu-Al	110 lb	

14. Relative flux* in reactor core:	Core UO_2	Core Pu-Al
Center rod	1.0138	1.0067
Coolant annulus	1.0378	1.0250
6-Rod ring	1.0628	1.0419
Coolant annulus	1.1426	1.1099
12-Rod ring	1.2284	1.1684
Coolant annulus	1.3842	1.2704
Process tube (Gas gap & shroud tube)	1.4669	1.3485
Moderator	1.7344	1.5993
Average flux in fuel	1.1648	1.1199

15. Maximum heat flux 330,000 Btu/(hr)(sq ft)

* Flux normalized by taking value at center line as one.

Concentric Elements (Figure 7)

1.	Length of UO_2 in element	7 ft 4 in.
2.	Length of element	8 ft 7 in.
3.	Length of hanger	5 ft 10 in.
4.	Length of fuel assembly (hanger and fuel element assembly)	14 ft 5 in.
5.	UO_2 diameters:	
	Rod O. D. (Mark II C)	0.548 in.
	Inner tube I. D.	1.082 in.
	Inner tube O. D.	1.782 in.
	Outer tube I. D.	2.328 in.
	Outer tube O. D.	2.948 in.
6.	Sheath material	Zircaloy-2
7.	Sheath thickness	0.060 in.
8.	UO_2 cross sectional area (total)	4.34 in.^{2}
9.	Total cross sectional area	6.03 in.^{2}
10.	UO_2 weight (10.2 g/cc)	141 lb
11.	Fuel element assembly weight	195 lb
12.	Hanger weight	40 lb
13.	Fuel assembly weight	235 lb
14.	Relative flux* in reactor core:	UO_2 Element
	Center fuel rod	1.0221
	Coolant annulus	1.0672
	Small fuel tube	1.1259
	Coolant annulus	1.2863
	Large fuel tube	1.4667
	Coolant annulus	1.7548
	Process tube (Gas gap & shroud tube)	1.9067
	Moderator	2.4345
	Average Flux in Fuel	1.3192

* Flux normalized by taking value at center line as one.

15. Maximum heat flux	400,000 Btu/(hr)(ft^2)

4. PRIMARY COOLANT SYSTEM

1. Heating load	66.4 MW
2. D_2O flow rate	8400 gpm
3. D_2O inventory (exterior of ring headers)	2200 gal

Steam Generator

4. Type and number	1-U tube and shell
5. Total surface area	6658 ft^2
6. Tube material	304 SS
7. Total evaporation rate (full power)	214,000 lb/hr
8. D_2O velocity	12 ft/sec (max)
9. D_2O pressure drop	14 psi

Pumps

10. Type and number	3 - Mech. seal
11. Spares in circuit	1
12. Operating head	110 psig
13. Capacity (each)	4200 gpm
14. Electrical power (each)	350 HP
15. Material	316 SS

Piping

16. Size	14 in. Sch 100
17. Material	304 SS
18. D_2O velocity	24 ft/sec

Pressurizer

19. Pressurizing method	Helium bleed from hp tanks
20. Pressurizer size	109 ft^3 D_2O
21. Control pressure	1025 psig

Steam System

22. H_2O flow	535 gpm
23. Steam generator pressure	425 psia

5. MODERATOR COOLANT SYSTEM

1. Heating load	2.2 MW
2. D_2O flow rate	1086 gpm
3. D_2O inventory	2544 gal
4. Temperature range	137-149 F

Heat Exchangers

5. Type and number	1, Countercurrent shell and tube
6. Total surface area	1000 ft^2

Pumps

7. Type and number	3, Mech. seal
8. Electrical power (each)	60 HP
9. Spares	1
10. Oper. Head	110 psig
11. Capacity	600 gpm

Secondary Cooling Water

12. H_2O flow rate	750 gpm at 68 F inlet

6. REFLECTOR COOLANT SYSTEM

1. Heating load	0.6 MW
2. D_2O flow rate	176 gpm
3. D_2O inventory	2580 gal
4. Temperature range	137 - 160 F

Heat Exchangers

5.	Type and number	1, Countercurrent shell and tube
6.	Total surface area	89 ft^2

Pumps

7.	Type and number	2, Vertical centrifugal
8.	Electrical power (each)	25 HP
9.	Spares	1
10.	Operating head	100 psig
11.	Capacity	200 gpm

Secondary Cooling Water

12.	H_2O flow rate	750 gpm at 73 F inlet

7. HEAVY WATER - GENERAL

1.	D_2O inventory, total	8210 gal
2.	Purity specification	99.8%
3.	D_2O density at 505 F	0.858 g/cc(53.4 lb/ft^3)
4.	D_2O heat capacity at 505 F	1.16 Btu/(lb)(F)
5.	Resin quantity (IX-1)	22 ft^3
6.	Resin operating temperature	130 F
7.	Total capacity - corrosion products	19.5 gram equiv./ft^3
8.	Storage capacity (Moderator storage tank and primary D_2O storage tank)	7000 gal

8. REACTOR CONTROLS

1.	Primary control	Moderator level
2.	Shim control	18 units (54 half-rods)
3.	Scram	Moderator dump

Moderator Level Control

4. Method of level control	Helium balance system
5. Moderator level, full power	9.25 ft maximum 7.25 ft minimum
Controllable moderator height	1.4 ft to 9.25 ft
6. Minimum rate of scram	decrease 2 ft in less than one second

Shim Control

7. Number of shim rods	54
8. Material	Inconel
9. Type construction	18 groups of half rods, 3 rods per group. Geared electric motor drives. One rod of each group moves independently; other two are geared together to give 3:2 speed ratio. Units contained within calandria and top and bottom shields.
10. Poison strength	~115 mk (2.1 mk per half rod)
11. Maximum rate of motion	36 inches/minute

Instrumentation

12. Signals used for primary control	
From 1 startup channel	period
From 2 log channels	period
From control channel	linear flux level
From Btu calculator	thermal power
13. Location of primary chambers	holes through side shield and reflector

14.	Controller action	Automatically adjusts moderator level to maintain desired power level and/or period
15.	Number of in-core flux monitor channels	13*

9. REACTOR CORE COMPONENTS (Figures 2, 3, and 4)

Calandria

1.	Over-all length (top of vessel to bottom of vessel)	outside 9 ft 10-1/2 in.
2.	I. D. of moderator shell	7 ft 0 in.
3.	Wall thickness of shell	1/4 in.
4.	Over-all diameter of calandria including dump chamber	11 ft 0 in.
5.	Material of construction	Aluminum
6.	Design pressure	5 psig minimum
7.	Test pressure	6.25 psig
8.	Number of process channel shroud tubes	85
9.	O. D. of process channel shroud tubes	4.25 in.**
10.	Wall thickness of shroud tubes	0.065 in.**
11.	Gas gap between shroud and process tubes	0.258 in. minimum
12.	Number of flux monitor channels	13
13.	O. D. of monitor channel tubes	1.5 in.
14.	Wall thickness of monitor channel tubes	0.049 in.
15.	Material of monitor channel tubes	Aluminum
16.	Number of shim control channels	18
17.	O. D. of shim control channels	2.5625 in.
18.	Wall thickness of shim control channel tubes	0.065 in.
19.	Volume of heavy water in moderator	284 ft^3

* Calandria tubes provided; monitor chambers not provided for initial design

** Center shroud tube 6 in. O. D., 0.085 in. thick

20. Bottom drain line	8 in.
21. Top overflow line	8 in. to 4 in.

Reflector

22. Type of reflector used	Heavy water tank type
23. Length of radial reflector (top to bottom)	6 ft 7-3/4 in.
24. Thickness of radial reflector	23-3/8 in.
25. Over-all diameter of outer reflector tank	11 ft 0 in.
26. Total metal thickness between inner reflector and moderator	1/4 in. of aluminum

10. SHIELDING

Top and Bottom Shields (Removable)

1. Weight of top shield	About 23 tons	Not including water or iron pellets outside of tube pattern
2. Weight of bottom shield	About 16 tons	
3. Thickness		40 in.
4. Material		Iron pellets and about 37% water
5. Density of mixture		285 lb/cu ft minimum
6. Neutron attenuation		about 10^6
7. Equipment activation tolerance		10^4 neut/(cm^2)(sec) at connectors

Side Thermal Shields (Removable)

8. Thickness	6 in.
9. Material	Iron
10. Neutron attenuation	about 10^1

Side Biological Shield	Operations Side (Cells B and C)	Process Side (Cell A)
11. Thickness	71 in.	71 in.
12. Material	Iron-Limonite concrete	Magnetite-Limonite concrete

13.	Density	265 lb/ft^3 (minimum)	210 lb/ft^3 (minimum)
14.	Neutron attenuation	About 10^{10}	About 10^9
15.	Radiation level at outside of shield	<1 mrem/hr	<10 mrem/hr

Secondary Top Shield

16.	Thickness	27 in.
17.	Material	Iron-loaded concrete
18.	Density	320 lb/ft^3 (minimum)
19.	Weight of discs	52, 14, 6, and 0.7 tons, respectively
20.	Radiation level at main floor	<10 mrem/hr

Coolant		Top and Bottom Shield	Thermal, Side and Secondary Shield
21.	Type	Recirculated water (treated)	Once-through water (sanitary)
22.	Quantity (total)	100 gpm	105 gpm
23.	Inlet temp. (max.)	112 F	70 F
24.	Outlet temp. (max.)	150 F	110 F

11. FUEL ELEMENT HANDLING

1.	Volume, storage basin	88,200 gal
2.	Area, Water, Storage basin	480 ft^2
3.	Volume, reactor hall water pit	13,700 gal
4.	Surface area, reactor hall water pit	29 ft^2
5.	Storage - irradiated, underwater	128 fuel assemblies
6.	Water shielding over element tip	15 ft
7.	Cooling period	120 days
8.	Capacity of storage basin crane	4000 lb
9.	Cold incoming storage, dry	85 fuel assemblies
10.	Charge-discharge location	top of reactor

11.	Charge-discharge mechanism	Self-propelled cask (fueling vehicle)
12.	Fueling vehicle motions	Raises and lowers fuel assembly and process tube w/ or w/o nozzle. Rotates and orients process tube and nozzle, indexes over tube and waterpit.
13.	Cooling	Forced air 250 cfm
14.	Transfer mechanism through containment vessel wall	Sliding carrier in water pit, winch-driven
15.	Useful lift in fueling vehicle	21 ft
16.	Fueling vehicle speed	variable to 40 ft/min
17.	Fueling vehicle transverse carriage speed	variable to 20 ft/min
18.	Discharge hook lift speed, lift	variable to 25 ft/min
19.	Charge cask hook speed, lift	25 ft/min

12. HELIUM SYSTEM

1. Uses of helium
   - a. Primary coolant pressurization
   - b. Moderator level control and gas sweep
   - c. Tube leak detection sweep gas between process tubes and shroud tubes
   - d. Reactor core blanket gas surrounding calandria, reflector and inner surfaces of shielding.

2.	Helium content of system	8000 scf
3.	Helium differential pump-number	1
4.	Helium differential pump-capacity	50 cfm
5.	Helium lines to top of calandria	4 - 8 in. lines
6.	Helium lines to bottom of calandria	4 - 8 in. lines
7.	Number and size of dump valves	4 - 8 in. valves

13. INSTRUMENTATION

Flow Monitor

1.	Channels	85
2.	Differential, max.	25 psi
3.	Unrecoverable pressure drop	Not to exceed 5 psi
4.	Normal operation %	80%
5.	Accuracy	±1%
6.	Venturi location	Inlet jumper
7.	High and low flow scram	Moderator dump
8.	Digital readout	Flow

Temperature Monitor

9.	Channels	85
10.	High temperature alarm	Annunciator
11.	Resistance thermometer location	Outlet jumper
12.	Range	0 to 600 F
13.	Accuracy (thermometer)	±2 F
14.	Response (63% point)(thermometer)	6 sec
15.	Digital readout	Temperature + Δ T

Activity Monitor - Primary Coolant

16.	Channels (process tubes)	85
17.	Type	Scintillation detector gamma spectrometer
18.	Alarm	Annunciator

Activity Monitor - Secondary Coolant

19.	Channels	4
20.	Alarm	Annunciator

Power Calculator

21. Channels	2
22. Total power level recorders	1
23. Deviation recorders	1
24. Range	0 - 100 MW
25. Accuracy	±1 MW

Shroud Tube Moisture Detection

26. Sampler lines	85
27. Channels	12
28. Sample line size	1/4 in.
29. Sample line flow	1/10 cfm
30. Type of moisture detection	Dewcell
31. Dewcell flow	1 ft^3/hr
32. Range	-20 to 30 F
33. Alarm	Annunciator
34. Channel indication switching	Automatic
35. Shroud tube indication switching	Manual

Reactor Radiation

Startup

36. Channels	1
37. Type of chamber	Fission
38. Logarithmic CRM	1
39. Period amplifiers	1
40. Short period trip	Scram (bypassable)

Log Instrumentation

41. Log amplifiers	2
42. Period amplifiers	2
43. Type of chamber	Compensated ion chamber

44. Short period trip	Scram (one bypassable at a time)

High Level

45. Linear amplifiers	3
46. High level trip	Scram
47. Type chamber	Uncompensated ion chamber

Safety Circuit

48. Process tube low flow trip	Scram
49. Process tube high flow trip	Scram
50. Reactor period - short period trip	Scram
51. Safety circuit failure trip	Scram
52. High neutron flux trip (2 of 3 detectors)	Scram (not bypassable)
53. Manual trip	Scram
54. Instrument air low pressure trip (2 of 3 detectors)	Scram (not bypassable)
55. Seismoscope trip (2 of 3 detectors)	Scram (not bypassable)
56. High pressure in pressurizer trip (2 of 3 detectors)	Scram (not bypassable)
57. Low pressure in pressurizer trip (2 of 3 detectors)	Scram (not bypassable)
58. High level in pressurizer trip (2 of 3 detectors)	Scram (not bypassable)
59. Low level in pressurizer trip (2 of 3 detectors)	Scram (not bypassable)
60. High steam generator pressure trip (2 of 3 detectors)	Scram (not bypassable)
61. Low steam generator liquid level trip (2 of 3 detectors)	Scram (not bypassable)
62. High building air activity	Scram
63. High secondary coolant activity	Scram
64. Electrical Power Failure trip	Scram (not bypassable)

Reactor Thermocouples

65.	Top shield	1
66.	Top shield coolant outlet	1
67.	Bottom shield	1
68.	Bottom shield coolant outlet	1
69.	Reflector	2 (inlet-outlet)
70.	Thermal shield	3
71.	Thermal shield coolant outlet	1
72.	Reactor moderator	18
73.	Moderator inlet-outlet	2
74.	Biological shield (side)	5
75.	Biological shield coolant outlet	1
76.	Lower access room wall	1
77.	Biological shield (top) coolant outlet	1
78.	Top tube sheet	1
79.	Top tube sheet coolant return	1
80.	Top sheet shroud tube bellows	2
81.	Display	Recorder and indicator
82.	Type	Iron-constantan, mineral insulated, stainless steel sheath

Building Radiation Detection (Beta-Gamma Detection)

83.	Number of Chambers	20
84.	Location and number	
	Experimental cell	1
	Storage basin work area	1
	B Cell loops	6
	Spare	4
	Instrument cell	2
	Process cell	2

Reactor hall	2
Hot shop	1
Tunnel	1
85. Indication	Recorder
86. Alarm	Annunciator

Tritium Detection

87. Number of chambers	1
88. Type	Kanne
89. Sample lines	6
90. Location and number	
Experimental cell	1
Instrument cell	2
Lower reactor face access room	1
Upper reactor face access room	1
Process cell	1
91. Indication	Recorder
92. Alarm	Annunciator

Instrument Panel

93. AC power required	
Type	Isolated (MG set)
Voltage, AC	120 volts
Regulation	Local (Sola type)
94. Galvanometer light	6 V battery
95. Galvanometer chamber	200 V battery

Instrument Air

96. Rate	50 cfm
97. Supply pressure	90 psig
98. Dew point	-20 F
99. Quality	Filtered and oil-free

14. ELECTRICAL SERVICES

1. AC Power supply	
Voltage	2400, 480, and 120 V at 60 cycles
Source of normal power	BPA
Reliability, BPA	Approx. 5 failures in 5 years; duration range from momentary to 86 minutes.
2. DC Battery	
Voltage	125 V. (Used only for 2400 V switchgear operation and emergency instrument power)
3. Power Backup	
Voltage	2400 V/480 V/120 V
Total emergency power available 375 KVA from emergency diesel generator installation	supplied @ 480 V

APPENDIX B

SUMMARY OF METEOROLOGICAL DATA[59]

The data reported here were obtained from remote-recording wind stations near the site of interest and from the Meteorology Tower records. The Meteorology Tower is located about 22 miles northwest of the PRTR site. Although the climatic regimes of the two locations differ in some respects, the Meteorology Tower records are included to provide an indication of the precipitation and atmospheric stability conditions. Precipitation and atmospheric stability are not currently measured at the remote-recording wind station near the PRTR site.

PRTR Site

The PRTR site lies along the Columbia River on fairly level terrain. The nearest significant topographic features are the bluffs across the river to the northeast, which rise 400 to 600 feet above the surface. The nearest other obstructions to air flow are the Horse Heaven Hills and Red Mountains at a distance of about 10 miles to the south and southwest. The terrain rises gently to the north and northwest up the Cold Creek and Columbia River Valleys. It slopes downward toward the south and southwest.

Table XVII shows wind speed and direction frequencies for the station located approximately three-quarter miles north of the PRTR site. This station has a mean wind speed of 10.3 mph; the highest average wind speed of any station in the network except one located on the north slope of Rattlesnake Mountain, about 500 feet above the valley floor. The southwest wind has the highest mean speed of 14 mph and also the highest frequency of occurrence. The direction associated with the lowest speed is east.

(59) Fuquay, J. J. and J. F. Scoggins, Personal Communication, November 5, 1955.

TABLE XVII

WIND SPEED AND DIRECTION FREQUENCY - 300 AREA

Wind Speed in Units of mph. Wind Direction Frequency in Per Cent of Time. Day: 0700-1900 PST. Night: 1900-0700 PST.

	Season														
	Winter			Spring			Summer			Fall			Annual		
	Day	Night	Total	Day	Night	Total	Day	Night	Total	Day	Night	Total	Day	Night	Total
NE	3.5	1.1	2.3	6.1	3.8	4.9	9.8	4.3	7.0	5.2	1.3	3.2	6.1	2.6	4.4
Speed	6.2	3.9	5.0	8.1	8.3	8.2	10.3	8.2	9.2	6.9	7.9	7.4	7.9	7.1	7.5
E	3.0	2.2	2.6	7.7	3.3	5.5	9.5	2.7	6.1	2.0	0.8	1.4	5.6	2.2	3.9
Speed	7.2	8.3	7.8	9.0	10.5	9.8	9.5	7.4	8.4	6.0	6.7	6.3	7.9	8.2	8.0
SE	15.5	17.2	16.4	16.9	16.6	16.8	19.8	16.6	18.2	11.8	17.7	14.8	16.0	17.0	16.5
Speed	8.4	8.8	8.6	9.2	8.0	8.6	10.6	7.4	9.0	7.9	7.8	7.8	9.0	8.0	8.5
S	12.5	13.9	13.2	10.1	9.5	9.8	10.4	6.2	8.3	11.5	9.4	10.4	11.1	9.8	10.4
Speed	14.5	14.0	14.2	17.5	13.4	15.4	14.9	12.9	13.9	12.2	10.2	11.2	14.8	12.6	13.7
SW	22.3	21.0	21.6	25.2	17.1	21.2	30.0	17.5	23.8	15.9	12.1	13.6	23.1	16.9	20.0
Speed	16.5	14.1	15.3	13.6	12.2	12.9	17.5	11.5	14.5	15.9	11.3	13.6	15.9	12.3	14.1
W	5.6	6.7	6.2	8.1	7.1	7.6	5.2	9.4	7.3	5.2	5.4	5.3	6.0	7.2	6.6
Speed	9.6	6.1	7.8	19.4	10.4	14.9	15.5	9.3	12.4	13.7	6.4	10.0	14.6	8.0	13.3
NW	15.0	19.3	17.2	9.7	22.7	16.2	6.8	21.2	14.0	15.2	20.0	17.6	11.7	20.8	16.2
Speed	10.1	8.3	9.2	16.1	12.2	14.1	14.8	11.5	13.2	11.7	8.7	10.2	13.2	10.2	11.7
N	15.5	11.8	13.6	12.6	12.4	12.5	7.7	16.9	12.3	20.8	13.6	17.2	14.1	13.7	13.9
Speed	10.2	9.5	9.8	14.1	11.1	12.6	14.2	10.7	12.4	10.8	9.7	10.2	12.3	10.3	11.3
Variable	6.4	6.5	6.4	2.8	6.0	4.4	0.5	2.0	1.2	9.7	13.1	11.4	4.9	6.9	5.9
Speed	2.2	2.2	2.2	1.9	2.0	2.0	2.0	2.0	2.0	2.2	2.5	2.4	2.1	2.2	2.2
Calm	2.7	0.3	1.5	0.8	1.3	1.0	0	0	0	0.4	0.5	0.4	1.0	0.5	0.8

UNCLASSIFIED

Another wind station with a slightly longer period of record is located in Richland about 6 miles south of the PRTR site. The winds observed at this station are greatly influenced by the mountains to the south, southwest, and west. Table XVIII shows the wind speed and direction frequencies for the Richland station for the period, September, 1952, to June, 1955. The mean wind speed is 8 mph. The strongest winds are from the southwest at 13 mph and the direction of highest frequency is west.

TABLE XVIII

WIND SPEED AND DIRECTION FREQUENCY - RICHLAND

Wind Speed in Units of mph. Wind Direction Frequencies in Per Cent of Time.

	Season				
	Winter	Spring	Summer	Fall	Annual
NE	3.5	5.1	7.0	5.6	5.3
Speed	4.4	7.3	4.8	3.8	5.1
E	7.8	12.4	12.1	10.4	10.7
Speed	4.4	5.0	4.5	3.6	4.4
SE	6.2	2.8	3.6	11.2	6.0
Speed	6.3	5.0	5.2	4.6	5.3
S	11.8	7.0	6.0	5.3	7.5
Speed	9.8	8.0	8.0	5.4	7.8
SW	22.0	14.2	18.1	14.0	17.1
Speed	15.9	14.2	11.1	11.0	13.1
W	20.3	28.4	25.8	19.9	23.6
Speed	7.2	11.5	9.0	7.6	8.8
NW	11.9	9.5	10.0	9.8	10.3
Speed	6.7	9.2	8.2	6.4	7.6
N	8.8	13.2	10.9	12.1	11.2
Speed	5.6	7.8	7.3	5.7	6.6
Variable	1.9	3.2	3.2	1.1	2.4
Speed	3.3	3.5	3.3	2.7	3.2
Calm	4.8	4.0	3.6	9.5	5.5

Meteorology Tower Records

1. Temperature Stratification

The differences in temperature between the 3-foot and 200-foot levels at the meteorology tower are used as a measure of the stability of the atmosphere. If the difference, $T_{200} - T_3$, is greater than -0.5 F a stable condition exists while if the difference is equal to or less than -1.5 F an unstable condition exists. The intermediate zone is considered as neutral. Table XIX presents the frequency of occurrence for each season and for the year for the period 1951 through 1953.

TABLE XIX

FREQUENCY OF OCCURRENCE OF STABLE, NEUTRAL, AND UNSTABLE LAPSE RATES, 3 FOOT - 200 FOOT, AT HANFORD METEOROLOGY TOWER

(Based on hourly observations for the years 1951 - 1953)

$$\Delta T = (T_{200} - T_3)^\circ F$$

	Season				
Item	Spring	Summer	Fall	Winter	Year
% of time $\Delta T > -0.5$ (stable)	45	38	57	57	49
% of time $-1.5 < \Delta T < -0.5$ (neutral)	8	8	10	15	10
% of time $\Delta T < -1.5$ (unstable)	47	54	33	28	41

Only a very limited amount of information about temperature stratification above 400 feet is available at Hanford. Observations which are available indicate that temperature inversions up to 10,000 feet above Mean Sea Level (MSL) are very rare during late spring, summer, and early fall months but are quite frequent during the colder seasons.

The frequency of occurrence of higher level inversions in various intervals of height up to 10,000 feet above Mean Sea Level (MSL) for the period November 1955 through March 1956 are shown in Table XX.

TABLE XX

PER CENT OF UPPER AIR OBSERVATIONS WHICH SHOWED A TEMPERATURE INVERSION WITHIN THE INDICATED HEIGHT INTERVAL

Height Interval (Ft. above MSL)	Frequency of Temperature Inversion (% of Observations)
1000 - 3000	23
3000 - 5000	10
5000 - 10,000	12

2. Wind Speed Vs. Lapse Rate

The joint frequency distribution, in per cent of total time, of various class intervals of wind speed at 200 feet, and the temperature lapse rate from 3 feet to 200 feet are shown in Table XXI.

3. Wind Roses at Two Hundred Feet Above Ground

The wind roses for the 200-foot level on the Hanford Meteorology Tower are summarized in tabular form in Table XXII to XXVI. These tables show the per cent of time the 200-foot wind blew from each of 16 directions and with an hourly mean speed in each of five class intervals of speed during each season and during the entire year.

4. Wind Direction During Periods of Precipitation

The frequency of wind direction occurrence during periods of precipitation is shown in Table XXVII.

TABLE XXI

FREQUENCY DISTRIBUTION OF 200-FOOT WIND SPEED AND TEMPERATURE LAPSE RATE

2-Foot to 200-Foot (Based on years 1951 - 1953)

Per Cent of Time

200-Foot Level

Wind Speed (mph)	0-4			5-9			10-14			15-19			≥20		
Stability*	S	N	U	S	N	U	S	N	U	S	N	U	S	N	U
Season															
Winter	16.5	6.0	12.0	15.7	2.6	7.1	12.5	2.6	5.4	6.2	1.7	1.7	5.7	2.1	2.2
Spring	7.5	1.4	9.4	11.6	1.2	14.5	13.6	1.5	9.1	8.0	1.8	5.6	4.6	2.4	7.8
Summer	3.7	0.9	11.6	8.8	1.0	21.2	12.7	1.2	10.6	8.5	1.8	4.7	4.4	3.0	5.9
Fall	16.2	4.0	14.7	16.8	1.5	9.2	14.1	1.5	4.4	6.9	1.2	2.5	2.7	1.4	2.9
Year	10.9	3.1	11.9	13.2	1.6	13.0	13.2	1.7	7.5	7.4	1.6	3.7	4.3	2.2	4.7

* S = Stable

N = Neutral

U = Unstable

TABLE XXII

FREQUENCY DISTRIBUTION OF WIND SPEED AND WIND DIRECTION AT 200-FOOT LEVEL

(Based on data for period 1951-1953)

PER CENT OF TIME

SPRING

Wind From:	Hourly Average Wind Speed (MPH) 0-4	5-9	10-14	15-19	≥20	Total %
NNE	1.14	1.14	0.95	0.31	0.19	3.73
NE	1.19	1.22	0.76	0.24	0.04	3.45
ENE	1.03	0.95	0.36	0.06	0.00	2.39
E	1.20	0.89	0.33	0.09	0.00	2.51
ESE	1.29	0.91	0.18	0.06	0.00	2.44
SE	1.01	0.88	0.22	0.04	0.03	2.19
SSE	0.68	0.94	0.46	0.33	0.12	2.53
S	0.91	0.91	0.37	0.25	0.21	2.65
SSW	0.65	1.09	0.76	0.58	0.45	3.52
SW	0.65	1.41	1.32	1.01	2.54	6.94
WSW	0.64	1.61	2.50	2.22	2.02	8.98
W	0.80	2.60	3.18	1.32	0.89	8.80
WNW	0.86	3.23	5.20	4.31	3.60	17.21
NW	1.49	4.28	4.89	3.72	4.51	18.89
NNW	1.61	2.30	1.64	0.86	0.73	7.14
N	1.49	2.05	1.12	0.30	0.02	4.97
Variable	1.04	0.28	0.00	0.00	0.00	1.32
Calm	----	----	----	----	----	0.31
Total	17.99	26.69	24.24	15.70	15.35	99.97

TABLE XXIII

FREQUENCY DISTRIBUTION OF WIND SPEED AND WIND DIRECTION AT 200-FOOT LEVEL

(Based on data for period 1951-1953)

PER CENT OF TIME

SUMMER

Wind From:	Hourly Average Wind Speed (MPH) 0-4	5-9	10-14	15-19	≥20	Total %
NNE	0.59	1.13	0.88	0.14	0.11	2.84
NE	1.12	1.41	0.65	0.32	0.18	3.68
ENE	0.62	0.92	0.35	0.20	0.06	2.15
E	0.97	0.95	0.17	0.03	0.00	2.12
ESE	1.26	1.04	0.09	0.00	0.00	2.39
SE	0.67	1.12	0.18	0.00	0.00	1.97
SSE	0.59	0.82	0.56	0.18	0.02	2.16
S	0.83	1.98	0.36	0.04	0.00	3.22
SSW	0.60	1.41	0.64	0.11	0.08	2.83
SW	0.85	2.24	2.01	0.91	0.60	6.61
WSW	0.71	2.01	2.06	1.45	0.59	6.82
W	0.80	2.47	2.95	0.94	0.39	7.55
WNW	0.76	3.31	6.78	6.35	3.72	20.92
NW	1.03	3.87	4.90	3.92	7.46	21.18
NNW	0.86	2.83	1.13	0.35	0.08	5.25
N	1.48	2.36	0.83	0.14	0.02	4.82
Variable	1.98	1.04	0.02	0.00	0.00	3.04
Calm	----	----	----	----	----	0.45
Total	16.17	30.91	24.56	15.08	13.31	

TABLE XXIV

FREQUENCY DISTRIBUTION OF WIND SPEED AND WIND DIRECTION AT 200-FOOT LEVEL

(Based on data for period 1951-1953)

PER CENT OF TIME

FALL

Wind From:	Hourly Average Wind Speed (MPH) 0-4	5-9	10-14	15-19	$\geq$20	Total %
NNE	1.89	1.17	0.76	0.14	0.03	3.99
NE	2.45	0.64	0.43	0.03	0.00	3.55
ENE	1.64	0.23	0.06	0.03	0.00	1.96
E	2.39	0.38	0.02	0.00	0.00	2.79
ESE	2.74	0.53	0.03	0.03	0.00	3.34
SE	2.62	1.14	0.47	0.14	0.06	4.43
SSE	1.14	1.23	0.56	0.24	0.08	3.26
S	1.36	0.87	0.37	0.08	0.17	2.83
SSW	1.05	0.79	0.56	0.55	0.73	3.69
SW	0.88	1.05	0.98	0.91	1.36	5.18
WSW	1.23	1.14	1.51	1.60	0.90	6.38
W	1.60	2.36	1.72	0.61	0.29	6.58
WNW	1.81	4.52	5.07	2.85	1.49	15.75
NW	2.45	5.80	6.18	3.18	1.68	19.30
NNW	3.15	3.06	0.90	0.09	0.02	7.22
N	3.00	2.56	0.44	0.11	0.09	6.20
Variable	1.08	0.12	0.00	0.00	0.00	1.20
Calm	----	----	----	----	----	2.35
Total	34.83	27.59	20.06	10.59	6.90	

TABLE XXV

FREQUENCY DISTRIBUTION OF WIND SPEED AND WIND DIRECTION AT 200-FOOT LEVEL

(Based on data for period 1951-1953)

PER CENT OF TIME

WINTER

Wind From:	Hourly Average Wind Speed (MPH) 0-4	5-9	10-14	15-19	≥ 20	Total %
NNE	1.46	0.60	0.24	0.18	0.03	2.51
NE	1.58	0.41	0.00	0.03	0.17	2.19
ENE	1.41	0.34	0.06	0.00	0.00	1.81
E	2.05	0.43	0.11	0.00	0.00	2.59
ESE	2.83	0.43	0.21	0.02	0.00	3.49
SE	2.11	0.67	0.18	0.06	0.03	3.06
SSE	1.59	0.66	0.28	0.23	0.28	3.22
S	1.32	0.72	0.55	0.41	0.48	3.48
SSW	0.98	0.98	0.80	1.20	2.34	6.29
SW	1.00	1.29	1.75	1.68	3.81	9.52
WSW	0.90	1.81	1.39	1.24	1.46	6.80
W	1.67	2.01	1.65	0.66	0.23	6.22
WNW	1.91	4.27	4.75	1.75	0.64	13.32
NW	3.31	6.77	7.20	1.65	0.52	19.45
NNW	3.72	2.70	0.90	0.23	0.00	7.55
N	3.02	1.07	0.29	0.21	0.00	4.59
Variable	1.09	0.17	0.00	0.00	0.00	1.26
Calm	----	----	----	----	----	2.79
Total	34.74	25.33	20.36	9.55	9.99	

TABLE XXVI

FREQUENCY DISTRIBUTION OF WIND SPEED AND WIND DIRECTION AT 200-FOOT LEVEL

(Based on data for period 1951-1953)

PER CENT OF TIME

ANNUAL

Wind From:	Hourly Average Wind Speed (MPH) 0-4	5-9	10-14	15-19	$\geq$20	Total %
NNE	1.27	1.01	0.71	0.19	0.09	3.28
NE	1.58	0.92	0.46	0.16	0.10	3.22
ENE	1.17	0.61	0.21	0.07	0.02	2.08
E	1.65	0.67	0.16	0.03	0.00	2.50
ESE	2.02	0.73	0.13	0.03	0.00	2.91
SE	1.60	0.95	0.26	0.06	0.03	2.91
SSE	1.00	0.92	0.46	0.25	0.12	2.75
S	1.10	1.12	0.41	0.20	0.21	3.04
SSW	0.82	0.07	0.69	0.60	0.89	4.08
SW	0.84	1.50	1.51	1.13	2.08	7.06
WSW	0.87	1.74	1.87	1.63	1.24	7.26
W	1.21	2.36	2.38	0.88	0.45	7.30
WNW	1.33	3.83	5.45	3.82	2.38	16.81
NW	2.06	5.17	5.78	3.12	3.56	19.70
NNW	2.33	2.72	1.15	0.39	0.21	6.79
N	2.24	2.01	0.67	0.19	0.03	5.15
Variable	1.30	0.40	0.00	0.00	0.00	1.71
Calm	----	----	----	----	----	1.46
Total	25.85	26.63	22.30	12.75	11.41	

TABLE XXVII

FREQUENCY OF WIND DIRECTION (50-FOOT LEVEL) DURING PERIODS OF PRECIPITATION

(Based on six years of record, 1950 through 1955)

SEASON

Wind From:	Spring		Summer		Fall		Winter		Year	
	No. Hr.	% Time	No. Hr.	% Time	No. Hr.	% Time	No. Hr.	% Time	No. Hr.	% Time
N	18	2.9	21	4.4	37	4.1	53	3.2	129	3.5
NNE	15	2.4	11	2.3	17	1.9	25	1.5	68	1.9
NE	11	1.8	16	3.4	25	2.8	36	2.2	88	2.4
ENE	10	1.6	8	1.7	15	1.7	23	1.4	56	1.5
E	15	2.4	12	2.5	28	3.1	36	2.2	91	2.5
ESE	20	3.2	9	1.9	36	4.0	40	2.4	105	2.9
SE	35	5.6	19	4.0	49	5.5	55	3.3	158	4.3
SSE	26	4.2	16	3.4	25	2.8	26	1.6	93	2.6
S	29	4.6	16	3.4	58	6.5	53	3.2	156	4.3
SSW	32	5.1	19	4.0	47	5.3	57	3.4	155	4.3
SW	48	7.7	46	9.7	51	5.7	65	3.9	210	5.8
WSW	39	6.2	46	9.7	49	5.5	29	1.8	165	4.5
W	46	7.3	60	12.7	58	6.5	65	3.9	229	6.3
WNW	99	15.8	65	13.7	115	12.9	329	19.9	608	16[illegible]
NW	117	18.7	76	16.1	167	18.7	401	24.2	761	20.9
NNW	41	6.5	18	3.8	50	5.6	139	8.4	248	6.8
Var.	0	0	1	0.2	0	0	0	0	1	0
Calm	25	4.0	14	3.0	65	7.3	221	13.4	325	8.9
Total	626		473		892		1655		3646	
% Time Precip. Occ.	4.7		3.6		6.8		12.7		6.9	

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APPENDIX C

ANALYTICAL FORMULATION OF NEUTRON KINETICS

The time dependent neutron density is given by:

$$\frac{dn}{dt} = \frac{\Delta k - \beta}{l^*} n + \Sigma_i \lambda_i C_i + S$$

and

$$\frac{dC_i}{dt} = \frac{\beta_i}{l^*} n - \lambda_i C_i$$

$$\Delta k = k_{eff} - \Delta k_f - 1$$

where

C_i = concentration of ith group of delayed neutron emitters.

λ_i = decay constant of the ith group

n = thermal neutron density

β_i = fractional yield of the ith group

β = $\Sigma \beta_i$ = total delayed fraction

l^* = effective lifetime of neutrons in reactor, sec

Δk = reactivity disturbance

S = source density, 14 neutron/(cm^3)(sec)

k_{eff} = effective neutron multiplication constant

The delayed neutron data used was that computed by D. Houser and M. V. Davis in HW-48907. Analogue computations were conducted for two different reactor loadings:

λ_i	80% U^{235} $\beta_i \times 10^5$	65% U^{235} $\beta_i \times 10^5$
7.7	17.5	14.5
2.2	102	87.2
0.465	282	245
0.400	17.8	15.7
0.170	130	114
0.0458	139.6	125
β_i (10^5)	688.9	601.4

For PRTR

$$l^* = \frac{8.82 \times 10^{-4}}{1.059 + \frac{1.218}{H^2}}$$

$$k_e = \frac{k_\infty e^{-\left(\frac{1.043}{H^2} + 0.0298\right)}}{1.0367 - \frac{1.35}{H^2}}$$

$$\frac{dH}{dt} = \frac{0.0766 - (0.010183)\sqrt{H^2 + 28.79\,H - 41.62}}{2.611 + 0.0872\,H}$$

where

H = moderator level in feet

k_∞ = effective neutron multiplication constant for infinite reactor

The fuel temperature coefficient is given by

$$\Delta k_f = \ln p_o \left(\frac{1}{\Sigma_r} \frac{d\,\Sigma_r}{dT} \right) \Delta \overline{T}$$

where

p_o = cold resonance escape probability, 0.865 for Mk I UO_2

T = volume average fuel temperature, F

$$\frac{1}{\Sigma_r}\frac{d\Sigma_r}{dT} = \text{resonance integral coefficient, } 8.33 \times 10^{-5}/\text{F}$$

The fuel temperature is given by

$$Cp\frac{dT_f}{dt} = 2.201\,(10)^{-7}\,n - U(T_f - T_c)$$

where

T_f = volume average fuel temperature, F

Cp = heat capacity for Mk I UO_2 fuel elements

$= 1.205 - 6.75\,T_f\,(10)^{-5}$ Mw-sec/F

T_c = coolant temperature, F

U = heat transfer coefficient

$$= \frac{Cp}{\tau} = \frac{Cp}{1.315 \times 10^2\,T_F - 1.5}$$

τ = time constant, seconds for a 63% change in heat transfer after a step nuclear power change.

The analogue circuits used for solving these equations were developed by W. D. Cameron.[60]

---

(60) Cameron, W. D. Safeguards Analysis of PRTR on an Analogue Computer, HW-61657. October 1, 1959.

APPENDIX D

CALCULATION OF THE CONTAINMENT PRESSURE

The method developed by Bailey(61) was used in estimating the equilibrium pressure-excess energy-temperature relationships for the PRTR containment building. The situation analyzed is the case of a reactor system rupture accompanied by a nuclear excursion and/or chemical reaction. All penetrations of the containment vessel are closed by a scram signal and the heavy water coolant is released from the pressurized system to the containment vessel. Vaporization of a fraction of the water and heating of the air results in a pressure increase within the building.

Several assumptions made in developing this calculation method are:

1. Instantaneous release of nuclear and chemical energy.
2. Thermal equilibrium of air and water vapor.
3. No heat transfer to or from structural masses inside the vessel.
4. No dynamic effects analyzed.
5. Air follows the perfect gas relation and has a constant specific heat.

In addition to these basic assumptions, calculations for the PRTR necessitated that it further be assumed that helium follows the perfect gas relation and has a constant specific heat.

Calculation of the equilibrium pressure, temperature, and excess energy for an assumed partial pressure of water vapor, p_m, involves five steps. First, the fraction of water vaporized, x, is determined. Second, the partial pressure of the air, p_a, is calculated. Third, the partial pressure of helium, p_{He}, is calculated. Fourth, the equilibrium total pressure, p_t, is determined. Fifth, the excess energy, Q, is calculated.

(61) Bailey, J. A. Enclosure Pressure Calculation Method, GEAP-0909. October 12, 1956.

Equations used in calculating these values are modifications of those developed by Bailey[62] and are as follows:

$$x = \frac{\left(\frac{V_s + V_{w_1}}{W_w}\right) - \left(\frac{W_h \; v_{g_2}}{W_w}\right) - v_{f_2}}{v_{fg_2}} \tag{1}$$

where:

V_s = free volume of containment vessel, cu ft.

V_{w_1} = volume occupied by water, cu ft.

W_w = weight of water, lb.

W_h = weight of moisture in air (humidity), lb.

v_{g_2} = specific volume of vapor at terminal conditions, cu ft/lb.

v_{f_2} = specific volume of saturated liquid at terminal conditions, cu ft/lb.

v_{fg_2} = specific volume of evaporation at terminal conditions, cu ft/lb.

The term V_{w_1}, the volume occupied by water before an incident, is so small in comparison to the free volume, V_s, that it can be neglected in the calculations. Similarly, v_{f_2} is negligible compared to the other terms in the numerator. Equation (1) can be simplified to

$$x = \frac{\left(\frac{V_s}{W_w}\right) - \left(\frac{W_h \; v_{g_2}}{W_w}\right)}{v_{fg_2}} \tag{2}$$

$$p_{a_2} = p_{a_1} \frac{T_2}{T_{a_1}} \tag{3}$$

---

(62) Bailey, J. A. Ibid. GEAP-0909, page 281.

where:

p_{a_2} = partial pressure of air at terminal conditions, psia.

p_{a_1} = partial pressure of air at initial conditions, psia.

T_2 = terminal equilibrium temperature, R.

T_{a_1} = initial air temperature, R.

$$p_{He_2} = p_{He_1} \frac{T_2}{T_{He_1}} \times \frac{V_{He_1}}{V_s} \tag{4}$$

where:

p_{He_2} = partial pressure of helium at terminal conditions, psia.

p_{He_1} = helium pressure before incident, psia.

T_{He_1} = initial helium temperature, R.

V_{He_1} = initial volume of helium, cu ft.

$$p_t = p_{a_2} + p_{He_2} + p_m \tag{5}$$

where:

p_t = containment vessel pressure at terminal conditions, psia.

p_m = partial pressure of water vapor, psia, (this value is initially assumed in starting the calculation).

$$Q = W_w(xu_{fg_2} + u_{f_2} - u_{f_1}) + W_h(u_{g_2} - u_{h_1}) + W_a c_{v_a} (T_2 - T_{a_1}) + W_{He} c_{v_{He}} (T_2 - T_{He_1}) \tag{6}$$

where:

Q = excess energy contribution, Btu.

u_{fg_2} = heat of evaporation at terminal conditions, Btu/lb.

u_{f_2} = enthalpy of liquid at terminal conditions, Btu/lb.

u_{f_1} = enthalpy of liquid at initial conditions, Btu/lb.

u_{g_2} = enthalpy of vapor at terminal conditions, Btu/lb.

u_{h_1} = enthalpy of water vapor in air at initial conditions, Btu/lb.

W_a = weight of air in containment vessel, lb.

c_{v_a} = specific heat of air at constant volume, Btu/(lb)(F).

W_{He} = weight of helium, lb.

$c_{v_{He}}$ = specific heat of helium, Btu/(lb)(F).

In the above equations there are seven unknowns: the terminal temperature ($T_2 = T_m$); the terminal vapor pressure (p_m); the terminal air partial pressure (p_{a_2}); the terminal helium partial pressure (p_{He_2}); the fraction of water vaporized (x); the excess energy contribution (Q); and, the final containment pressure (p_t). Assuming a value for the terminal partial vapor pressure (p_m) and determining the terminal temperature (T_m) from steam table relationships, the values of each of the remaining unknowns can be calculated using equations (2) through (6). Interrelationships of any two of these variables can be shown by plotting the values determined by repeating the calculations for several values of p_m. The most useful relationship is the one between Q, the excess energy contribution, and p_t, the final containment vessel pressure. When the course of an accident has been analyzed, the heat generation (from nuclear and chemical sources) can be calculated, and assuming no heat transfer from the containment vessel the heat generation becomes Q. From the plot of Q versus p_t the final containment pressure can be determined.

Sample Calculation

Containment vessel free volume	400, 000 cu ft
Weight of water* in primary coolant system	19, 950 lb
Initial water temperature	505 F
Initial air conditions	
Temperature	100 F
Pressure	14. 7 psia
Relative humidity	50 per cent
Partial pressure of dry air	14. 225 psia
Weight of helium	540 lb
Temperature of helium	160 F
Pressure of helium	14. 7 psia
Volume of helium	27, 800 cu ft

The weight of air in the containment vessel is

$$W_a = \frac{M\, p_a\, V_s}{R\, T} = \frac{28.83 \times 14.225 \times 400,000}{10.73 \times 560} = 27,300 \text{ lb.}$$

and the weight of water vapor in the air for 50 per cent relative humidity is

$$W_H = V_s \frac{r\,h}{v_g} = 400,000 \times \frac{0.5}{350.4} = 571 \text{ lb.}$$

For example let:

$p_m = 10$ psia

$T_m = 193$ F $= 653$ R

$u_{f_2} = 161.17$ Btu/lb

$u_{fg_2} = 982.1$ Btu/lb

$v_{f_2} = 0.01659$ cu ft/ft.

$v_{fg_2} = 38.403$ cu ft/lb.

$v_{g_2} = 38.420$ cu ft/lb.

$c_{v_a} = 0.172$ Btu/(lb)(F)

* For these calculations the thermodynamic properties of light water as given in steam tables were used.

$u_{g_2} = 1143.3$ Btu/lb. $\qquad c_{v_{He}} = 0.75$ Btu/(lb)(F)

$$x = \frac{\left(\frac{V_s}{W_w}\right) - \left(\frac{W_h v_{g_2}}{W_w}\right)}{v_{fg_2}} = \frac{\frac{400,000}{19,950} - \frac{571 \times 38.420}{19,950}}{38.403} = 0.494$$

$$Q = W_w(xu_{fg_2} + u_{f_2} - u_{f_1}) + W_h(u_{g_2} - u_{h_1}) + W_a c_{v_a}(T_2 - T_{a_1}) + W_{He} c_{v_{He}}(T_2 - T_{He_1})$$

$$= 19,950\,(0.494 \times 982.1 + 161.17 - 494) + 571\,(1143.3 - 1105)$$
$$+ 27,300 \times 0.172\,(653 - 560) + 540 \times 0.75\,(653 - 620)$$

$$= 3,507,000 \text{ Btu}$$

$$p_{a_2} = p_{a_1}\frac{T_2}{T_{a_1}} = 14.225\,\frac{653}{560} = 16.59 \text{ psia}$$

$$p_{He_2} = p_{He_1} \times \frac{T_2}{T_{He_1}} \times \frac{V_{He}}{V_s} = 14.7 \times \frac{653}{620} \times \frac{27,800}{400,000} = 1.08 \text{ psia}$$

$$p_t = p_m + p_{a_2} + p_{He_2} = 10 + 16.59 + 1.08 = 27.67 \text{ psia or } 12.97 \text{ psig}$$

Figure 64 is a plot of the containment vessel pressure, p_t, versus the excess energy contribution, Q, for the PRTR.

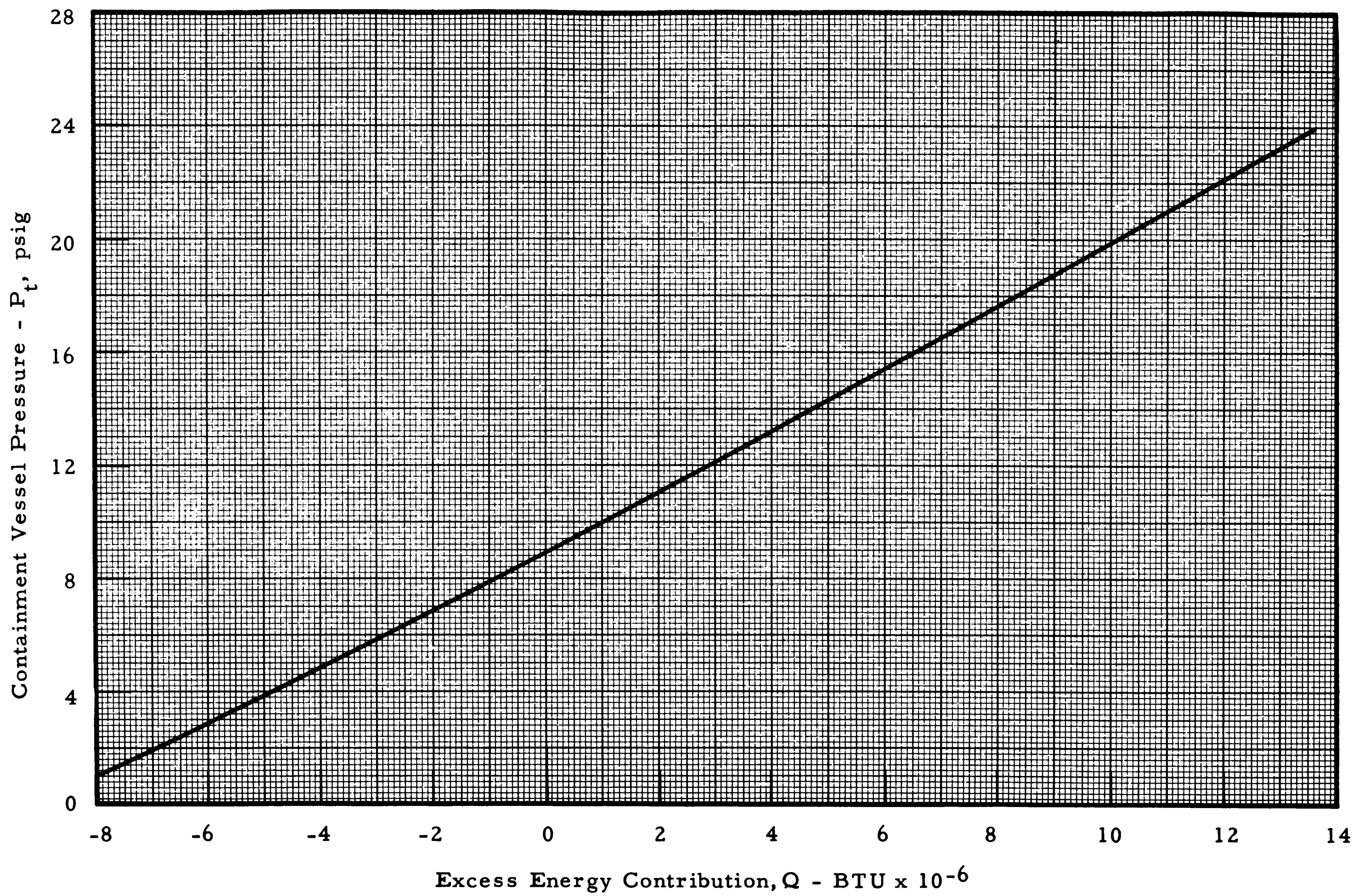

FIGURE 64

Containment Vessel Pressure vs. Excess Energy Contribution

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