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PLUTONIUM RECYCLE CRITICAL FACILITY
FINAL SAFEGUARDS ANALYSIS

By

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ABSTRACT

The Plutonium Recycle Critical Facility (PRCF) is a very low-power experimental reactor and is designed to supplement the Plutonium Recycle Test Reactor (PRTR). The Critical Facility will be used for the determination of basic nuclear constants of heterogeneous reactors recycling plutonium, to assist in planning PRTR fuel loadings, to obtain reactivity measurements of reactor fuels, and for exponential, approach-to-critical, and critical experiments. This report contains a description of the reactor, building, and associated equipment. A description of the anticipated operating program and operating procedures is also included. Reactor safeguards aspects of possible equipment malfunction and procedural errors are analyzed and inherent reactor safety features are described.

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PLUTONIUM RECYCLE CRITICAL FACILITY
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I. INTRODUCTION

The Plutonium Recycle Critical Facility (PRCF) is a very low power experimental reactor designed to supplement the Plutonium Recycle Test Reactor (PRTR) in the Plutonium Recycle Program. The Critical Facility will be used for determinations of basic nuclear constants of heterogeneous reactors recycling plutonium, and for exponential and criticality studies at power levels up to 100 watts and neutron fluxes in the range of 10^8-10^9 nv. Provision is included for changing the reactor lattice and for using irradiated and unirradiated fuel elements.

Initially, the reactor will be operated with a heavy water moderator. At a later time, tests using light water moderation are planned. The physics and loading parameters for the light water moderated reactor have not been determined as yet. Before such operation begins, a supplement to this safeguards report will be issued, presenting the safeguards analysis for the light water moderated reactor.

The safeguards philosophy used in the design of the Critical Facility is outlined as follows:

1. Confinement - Radioactive materials released in a credible incident must be contained within safe limits. A confinement system utilizing activated charcoal filters and an absolute filter will prevent release of unsafe quantities of radioactive materials to the environs from any accident, including the maximum credible accident. *

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* Maximum credible accident is defined as a foreseeable accident based on a logical chain of events, although having a negligibly small probability of occurrence.

2. Control - The nuclear reactions shall be controlled at all times.
   (a) The safety system shall be designed to override any credible nuclear excursion.
   (b) The rate of reactivity increase is to be limited by system design so that reaction time of the safety system will prevent unsafe nuclear excursions.
   (c) Systems shall be designed as fail-safe (malfunction to shut down reactor) to the maximum degree practicable.
3. 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.

Before this facility is operated, the design and plans for operation will also be reviewed by the General Electric Company Technological Hazards Council.

II. SUMMARY

Reactor and Building

The Critical Facility reactor consists of an aluminum tank containing heavy water moderator with fuel elements suspended vertically in the moderator from a top grid plate. The reactor will operate at powers up to 100 watts during critical experiments. Fine reactivity changes will be made with three poison-type control rods. Reactor scram will be accomplished by interrupting the current to electromagnets which hold the control rods and three poison-type safety rods, allowing them to drop into the reactor. Added experimental flexibility is obtained by varying the moderator level with a telescoping weir arrangement. The weir will be adjustable to control the moderator level over the range from five to nine feet.

The reactor tank will be located in a 10 x 13 x 32-foot deep impervious concrete cell located below grade. A crane in the reactor cell will be used for loading and unloading the reactor and for moving equipment within the cell. Personnel access to the cell will be provided by removing cover blocks from openings in the in the four-foot thick, high density concrete cell ceiling.

Irradiated fuel elements will be transferred into the reactor cell from the water-filled Plutonium Recycle Test Reactor (PRTR) loadout canal through an air-to-water fuel transfer lock. Irradiated fuel elements will be cooled during transfer and while positioned in the reactor in aluminum thimble tubes with a recirculating water system supplemented by a single-pass air system. Individual fuel elements will be placed in a thimble in the loadout canal and the canal water blown from the thimble to a separator by the single-pass, air cooling system blower. When dry, the thimble will be refilled with heavy water, the recirculating water system started, and the fuel-containing thimble transferred through the fuel transfer lock to the reactor.

Exhaust air from the Critical Facility will normally be routed directly to the PRTR exhaust ventilation system. A ventilation confinement system, actuated by high exhaust air activity, provides for automatic closure of a valve in the Critical Facility exhaust line. A check valve in a by-pass line around the confinement valve will open if cell pressure should increase. The exhaust air will then pass through two activated charcoal filters (each filter having a 95 percent removal efficiency for halogens) prior to entering the PRTR exhaust system. The PRTR exhaust system consists of a bank of particulate filters capable of removing 99.97 percent of particles 0.3 micron and larger, a bank of activated charcoal filters having a 98 percent removal efficiency for iodine, and a 150-foot high exhaust stack.

Safeguards Analysis

The major safeguards against serious reactor incidents are: fail-safe design of circuitry and components as far as practicable; multiplicity of sensors and circuit components; interlocks to prevent unsafe combinations of functions; use of key locks on controls; and most important, the selection and training of competent operating personnel. Areas of responsibility and decision-making powers will be clearly defined. Procedures for review and approval of process limitations for the PRTR will be extended to the Critical Facility.

Possible nuclear excursions were evaluated, and it was found that core melting would be prevented by the poison-type safety rod system.

Analysis of conceivable failures showed that melting of an irradiated fuel element would require simultaneous failure of the coolant system and the reactor cell crane. Complete coolant system failure would require simultaneous failure of both the normal and emergency cooling systems or simultaneous failure of both the inlet and outlet coolant lines such that it would be impossible to furnish either normal or emergency coolant to the fuel element. Mechanical failure of the cell crane or complete power failure (normal and emergency) when the crane is positioned over the reactor would then prevent movement of the thimble to a position where the fuel element could be removed from the thimble and adequately cooled. If melting of a freshly discharged fuel element should occur, the exhaust air filtering system would limit off-site contamination to insignificant levels. Doses to inhabitants would be minimal and off-site controls would not be required.

III. THE REACTOR AND BUILDING

Description of Reactor

The Critical Facility reactor consists of a cylindrical aluminum tank containing vertically oriented fuel elements suspended in the moderator. Fuel elements, control and safety rods, and flux monitor tubes will be supported by two grid plates at the top of the reactor tank. Initially, the reactor will be operated with a heavy water moderator and an eight-inch equilateral triangular lattice spacing. Tests using light water moderation are planned, and two separate grid plates are provided to allow lattice changes without completely rearranging the core. The reactor is provided with a system for inserting and removing a neutron source. Systems for regulating moderator height and for cooling preirradiated fuel elements are provided, as well as various instrumentation, control, safety, and service systems.

The reactor is located in a concrete cell below grade, adjacent to the PRTR loadout canal. The reactor itself is unshielded, with the cell walls and surrounding earth providing radiation protection for surrounding areas. An air-to-water fuel transfer lock in the wall between the reactor cell and the PRTR loadout canal, and a fuel transfer system, will be used for transferring irradiated PRTR fuel elements to and from the reactor cell. Such operations will be performed remotely and will be observed through a periscope.

Reactor Arrangement

The reactor core and the moderator are contained in a cylindrical aluminum tank, six feet in diameter and nine feet high. Thicknesses of the bottom plate of the tank and the cylindrical wall are 3/4-inch and 1/4-inch, respectively.

Two aluminum grid plates form a leakproof lid for the tank. As shown in Figure 1, unirradiated fuel elements, control and safety rod thimbles, flux monitor tubes, and thimble tubes containing irradiated fuel elements are supported by the top grid plates. Control and safety rod thimbles, and unirradiated fuel elements can be manually placed in any grid position. The outer ring grid plate is supported by a flange rimming the top of the tank. The 1-1/2-inch thick plate is approximately seven feet in outside diameter with a hexagonal hole as shown in Figure 2. The grid pattern consists of six 3-inch holes for flux monitor tubes and forty-two 3-1/2-inch diameter holes arranged on an eight-inch equilateral triangular lattice. Flux chambers, control and safety rod thimbles, and unirradiated fuel elements will be suspended from the outer ring. The hexagonal center edges provide a surface for gasketing and bolting the inner grid plate to the outer plate. The outer grid plate is gasketed and bolted to the tank flange and supports the weight of reactor components suspended from both plates. The grid pattern of the two-inch thick, hexagonal-shaped inner plate consists of 18 four-inch diameter holes and a central six-inch test hole arranged on an eight-inch equilateral triangular lattice. The lattice positions are capable of receiving thimble tubes containing irradiated fuel elements, unirradiated fuel elements, and control and safety rod thimbles. Aluminum inserts to reduce the hole

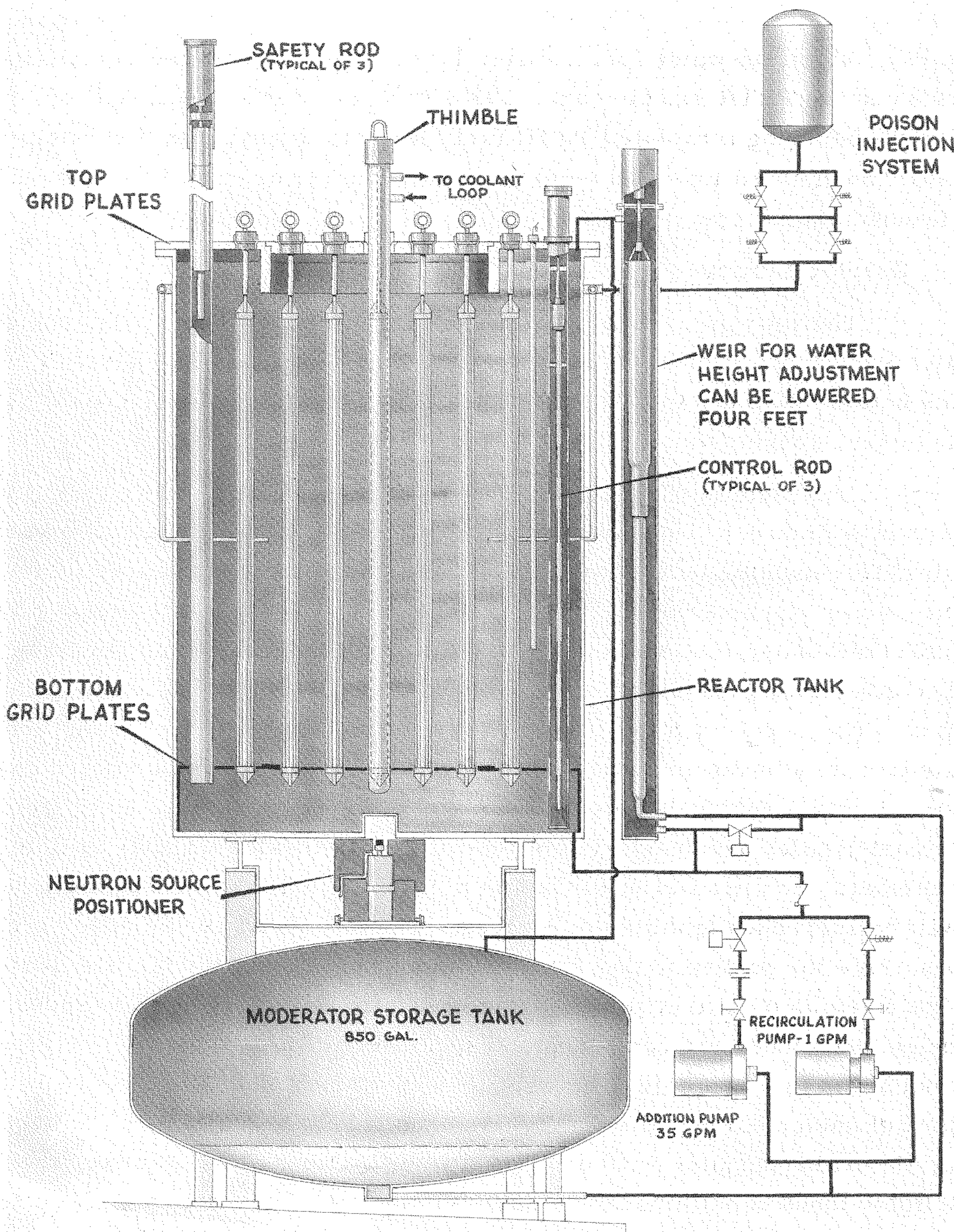

FIGURE 1

Plutonium Recycle Critical Facility, Assembly

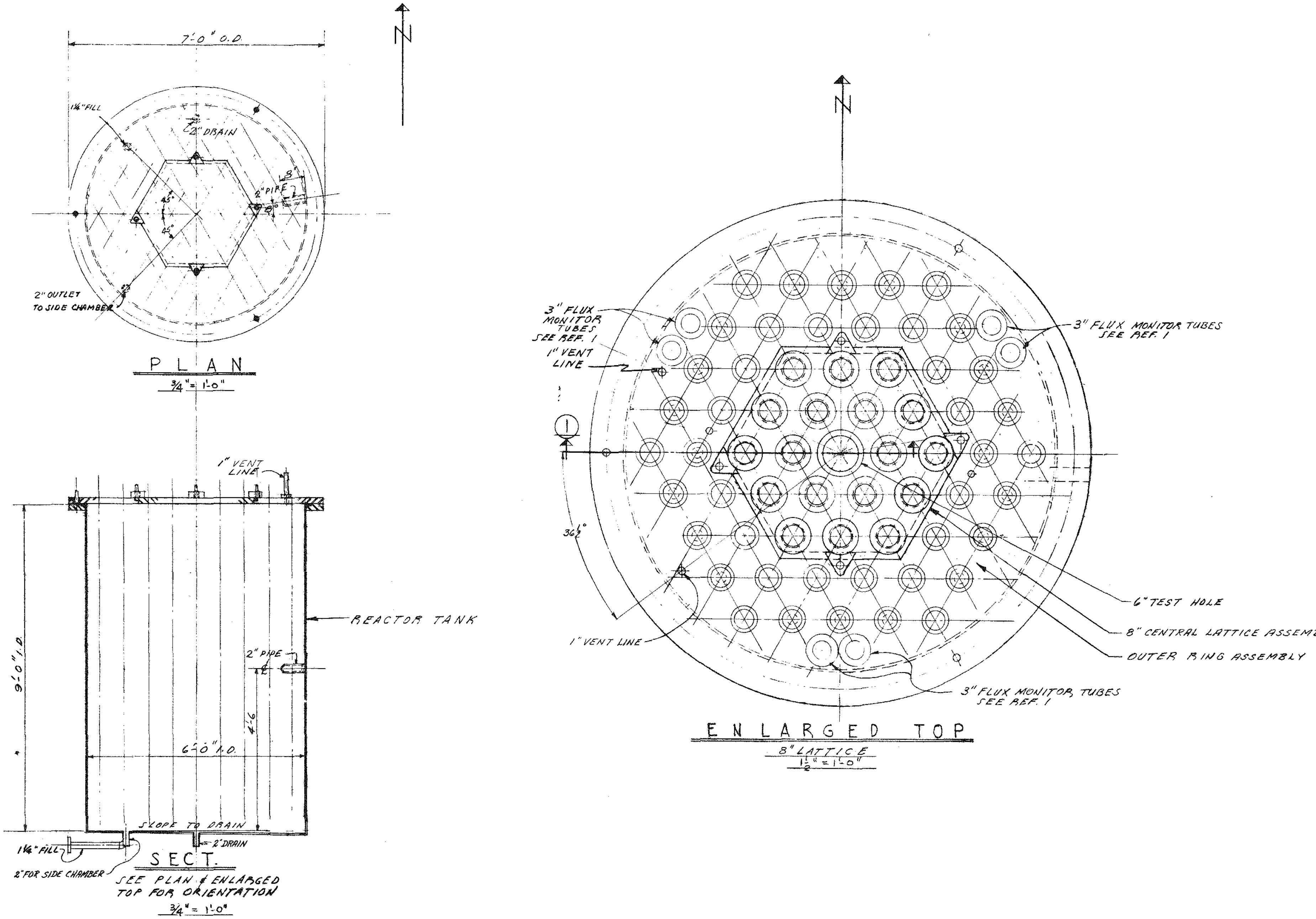

FIGURE 2

Reactor Tank, Section and Plan

size are required to install control and safety rod thimbles and unirradiated fuel elements in the lattice positions of the inner grid plate. Control and safety rod thimbles and flux chamber thimbles are bolted to the top grid plates. Fuel element hangers are gasketed with O-rings to provide a seal but are not mechanically fastened to the top grid plates.

Two 1/2-inch bottom grid plates are installed near the bottom of the reactor tank to prevent the lower ends of the fuel elements from swinging together. The bottom grid plates are similar in appearance to the top plates and have the same lattice spacing as the top grid plates. Hole diameters in the bottom plates are 1/4-inch larger to allow for installation of guide tubes in each fuel element position if this is found desirable as the experimental program progresses. The bottom grid plate is supported by brackets welded to the bottom of the reactor tank.

A poison wrap or a reflecting material will be installed on the outside of the reactor tank. Initially, sheets of cadmium, 0.020-inch thick, will be wrapped around the tank. The cadmium sheets will be held firmly in place by steel banding straps on approximately 6-inch centers. In other tests, the cadmium sheet will be replaced with an appropriate thickness of reflector material, fastened to the tank in a similar manner.

Fuel Elements

Two general types of fuel elements will be used in the reactor. These are enriched, driver fuel elements and natural uranium (metallic or oxide) buffer-zone fuel elements. In cases where driver and buffer-zone fuel elements are similar in appearance, hangers for the two types will be obviously different and will be constructed such that hangers for unenriched elements cannot be attached to enriched elements.

Fuel elements may contain UO_2, Pu-Al, UO_2-PuO_2 mixtures, ThO_2-PuO_2 mixtures, and recycled plutonium fuels. The reactor loadings will consist of zoned (enriched region and buffer region) and quasi-uniform mixtures of the above fuel elements, all enriched fuel element loadings, and all natural uranium fuel element loadings (metallic or oxide) for exponential experiments.

Initially, reactor loadings will consist of fuel elements of the following types:

Driver Fuel Elements

Enriched elements used in the driver zone will be PRTR Mark I, cluster-type assemblies, composed of nineteen 0.504-inch diameter Pu-Al rods clad with Zircaloy-2. This type of element, which will be used with D_2O moderated lattices, is shown in Figure 3.

Buffer Zone Fuel Elements

Buffer zone fuel elements will contain natural uranium in either metallic or oxide form. For the D_2O moderated cores, buffer zone fuel elements initially will be PRTR Mark I type fuel elements with UO_2 cores.

Because of the reactivity of individual fuel elements involved, partial clusters and individual rods from unirradiated PRTR Mark I type fuel elements will also be used in the reactor.

Provisions are also included for testing fuel elements which have been irradiated in the PRTR. The cooling system for these fuel elements is described in the following section.

Moderator and Coolant Systems

The Critical Facility reactor was designed for operation with either light or heavy water moderator and for variation of moderator level to increase the experimental flexibility. Water in the reactor tank will serve not only as the moderator, but also as a coolant for the unirradiated fuel elements. Unirradiated fuel elements will be suspended directly in the moderator and the heat generated during reactor operation will be transferred directly to the moderator. Irradiated fuel elements will be charged in thimbles which are connected to a separate, forced circulation, cooling system. The cooling system consists of a closed, water-cooled loop supplemented by single pass air and emergency water cooling facilities. System piping is arranged to recirculate light or heavy water.

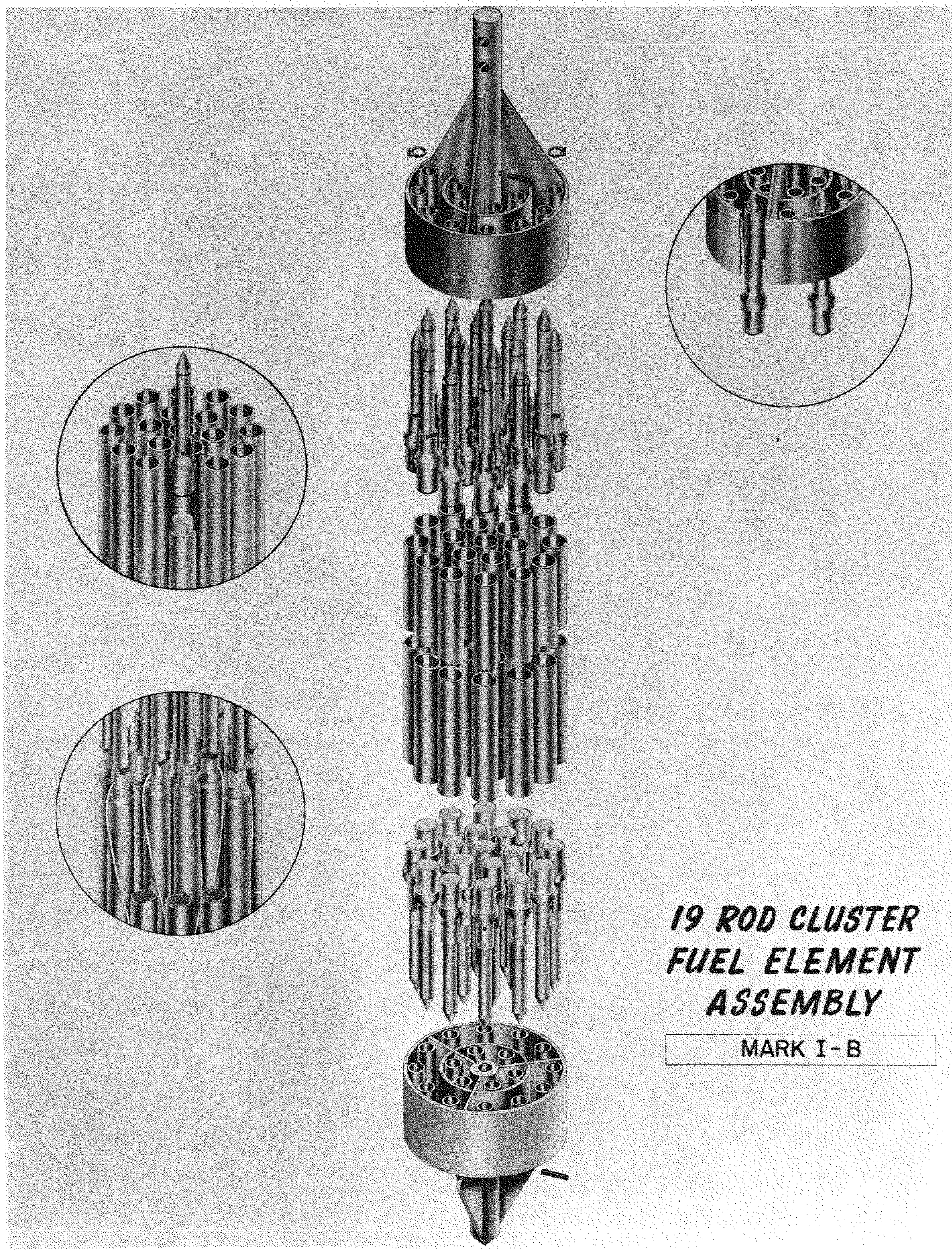

FIGURE 3

Mark I, 19-Rod Cluster Fuel Element Assembly

Moderator System

The moderator system includes:

1. two stainless steel storage tanks, sized to completely drain the reactor tank;
2. pumps and lines for transferring moderator between the storage tanks and the reactor tank, and for adding water to and draining water from the system;
3. an adjustable weir overflow assembly to allow variation of moderator height in the reactor.

A schematic diagram of the moderator system is shown in Figure 4.

During reactor operation the moderator will be maintained at a level between 5 and 9 feet above the bottom of the reactor tank. The moderator will be maintained at the level desired for a particular experiment with an adjustable weir and a one gpm recirculation pump. The weir is installed in a vented side chamber and consists of an outer tube and telescoping inner tube. The outer tube is raised and lowered by a lead screw driven by a reversible motor. Moderator flows in the bottom of the outer tube, up and around the annulus formed by the two tubes and down the inner tube. Instrumentation and controls have been provided to allow adjustment and readout of the weir position from the control area. A mechanical stop, adjustable only from the reactor cell, prevents raising the weir from the control console above a predetermined level. The design of the weir is shown in Figure 5.

Two stainless steel moderator storage tanks are provided. The tanks are sized to completely drain the reactor tank. An 850-gallon storage tank is located beneath the reactor tank. This tank was sized to accommodate the volume of water required to change the reactor moderator level from 5 to 9 feet. An auxiliary storage tank (1060 gallon capacity) will be used to store the remaining moderator volume during fuel element loading and core alterations. The auxiliary storage tank is located outside of the reactor cell in the PRTR service basement.

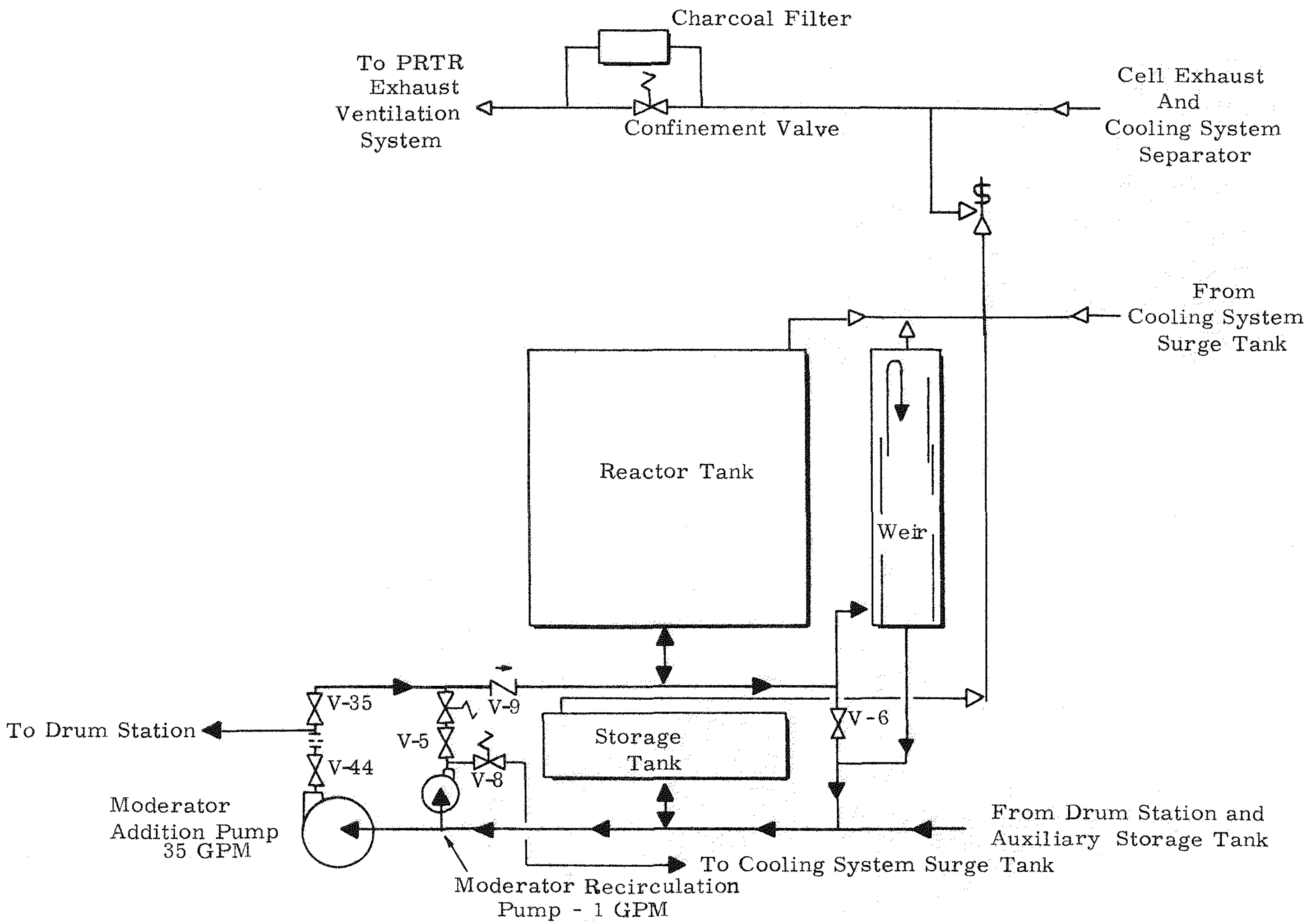

FIGURE 4

Moderator System

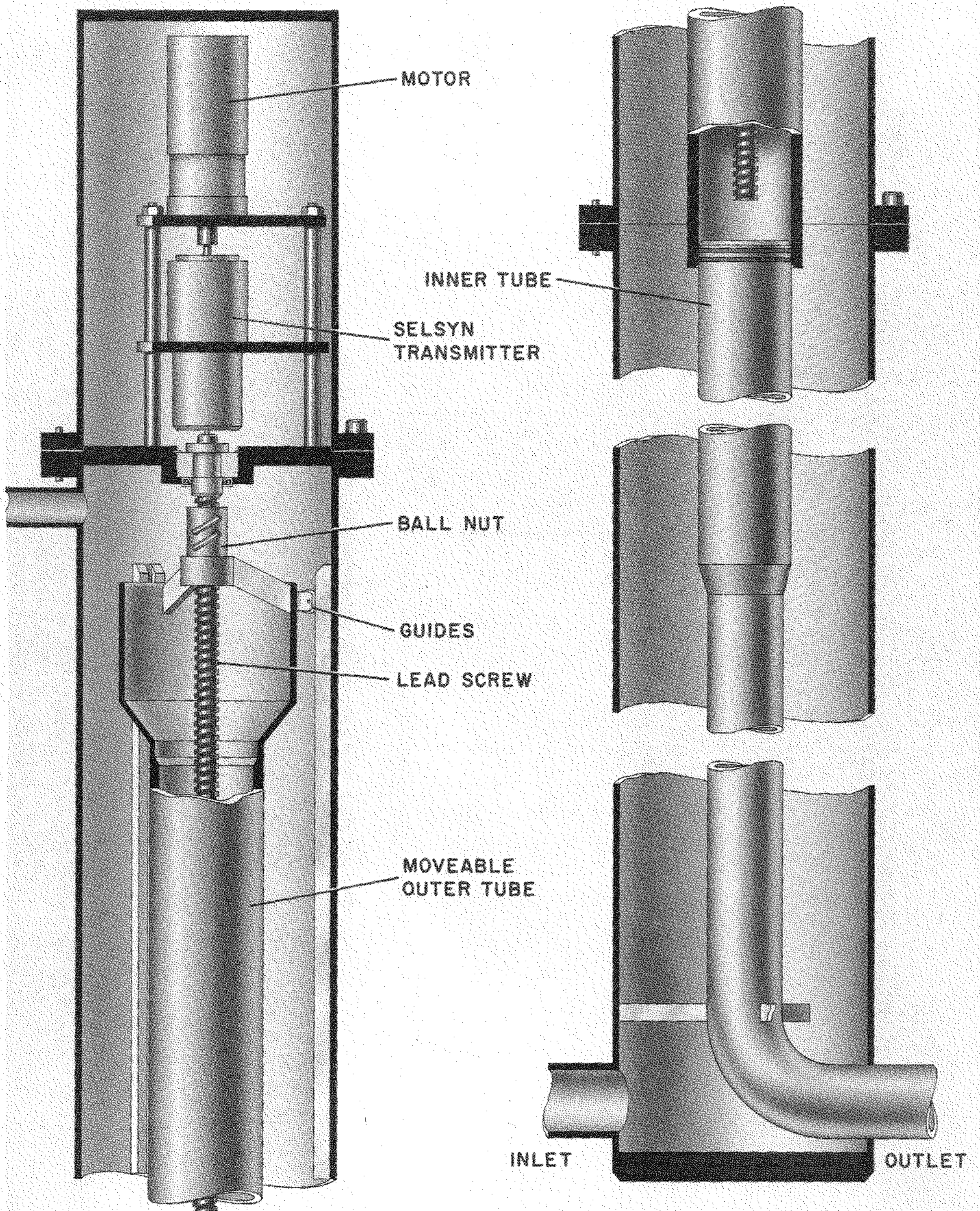

FIGURE 5

Adjustable Weir

Moderator will be pumped from the storage tanks into the reactor tank in two steps. In the first step, the D_2O in the auxiliary moderator storage tank will be pumped into the tank to a predetermined level at a maximum rate of 200 gpm. In the second step, the moderator will be raised to the operating level by pumping moderator from the cell storage tank at a rate of approximately 35 gpm.

Irradiated Fuel Element Cooling System

A recirculating cooling system will remove fission product decay heat from two classes of fuel elements which have been irradiated in the PRTR. These are:

1. one "short-cooled" PRTR element which has a maximum power generation rate of 12 kw (e. g. , a fuel element transferred from the PRTR to the Critical Facility as quickly as possible for short-term reactivity measurements such as xenon transients) and,
2. up to seven "long-cooled" PRTR elements which have been out of the reactor for approximately 30 days. Reactivity measurements for long-cooled elements will be used to determine long term reactivity changes, such as burn-out effects.

The system will be used to cool irradiated fuel elements in transit and while positioned in the reactor and is arranged to provide recirculating water cooling or single pass air cooling. Aluminum thimble tubes (described later in this section) provide the fuel element carrying and cooling functions. System piping and valving is arranged to cool up to seven thimbles containing "long-cooled" fuel elements, or one thimble containing a "short-cooled" fuel element.

The single "short-cooled" fuel element will remain in the same thimble in transit and while positioned in the reactor. The transfer and in-reactor cooling of a "long-cooled" fuel element will require at least two separate thimbles. Up to seven thimbles can be positioned in the reactor. A separate thimble will be used to transfer the fuel element through the fuel transfer lock to the reactor cell. Once inside the cell, the fuel element will be removed from the "transfer" thimble and placed in one of the "in-reactor" thimbles.

The cooling system is designed to maintain the inlet temperature less than 80 F, and allow a temperature rise of less than 2 F for a single 10 kw element, or 5 F for seven elements producing a total power of 25 kw. As shown in Figure 6, the system is arranged to pump water with a 35 gpm, 100-foot head, centrifugal pump through a flow measuring orifice, the heat exchanger, and a supply manifold for in-reactor thimbles. Two diaphragm operated valves are installed in the system piping downstream of the manifold, an isolation valve and a three-way valve (air exhaust). The supply leg of the system ends at a hose connection in the fuel transfer lock. The supply hose to the thimble tube installed in the fuel transfer lock will be attached to this connection. The return hose from the thimble is connected to the effluent portion of the system piping, which includes a diaphragm-operated three-way valve (for the introduction of air to the system), a diaphragm operated isolation valve, surge tank, and the effluent manifold for the in-reactor thimbles.

A portion of the cooling system is arranged to cool the fuel transfer lock thimble with 250 scfm single pass air flow. The single pass portion of the loop is separated from the remainder by the isolation valves. The three-way valves provide an inlet and outlet for the air flow. As shown in Figure 6, air flows from the blower to the thimble through the "effluent portion" of the system piping. Exhaust air from the thimble passes through an entrainment separator to the exhaust ventilation system. The purpose of the air flow is:

1. to remove light water from the thimble after receiving a fuel element,
2. to recover heavy water from the thimble before returning an element to the canal, and
3. to cool and dry the element during (1) and (2).

The 9 foot 7-1/2-inch long thimble tube assembly consists of two concentric aluminum tubes. The bottom of the 4-inch OD, 1/8-inch wall outer tube is closed. The bottom of the 3-1/2-inch OD, 1/8-inch wall inner tube is open. Coolant enters the assembly near the top of the outer

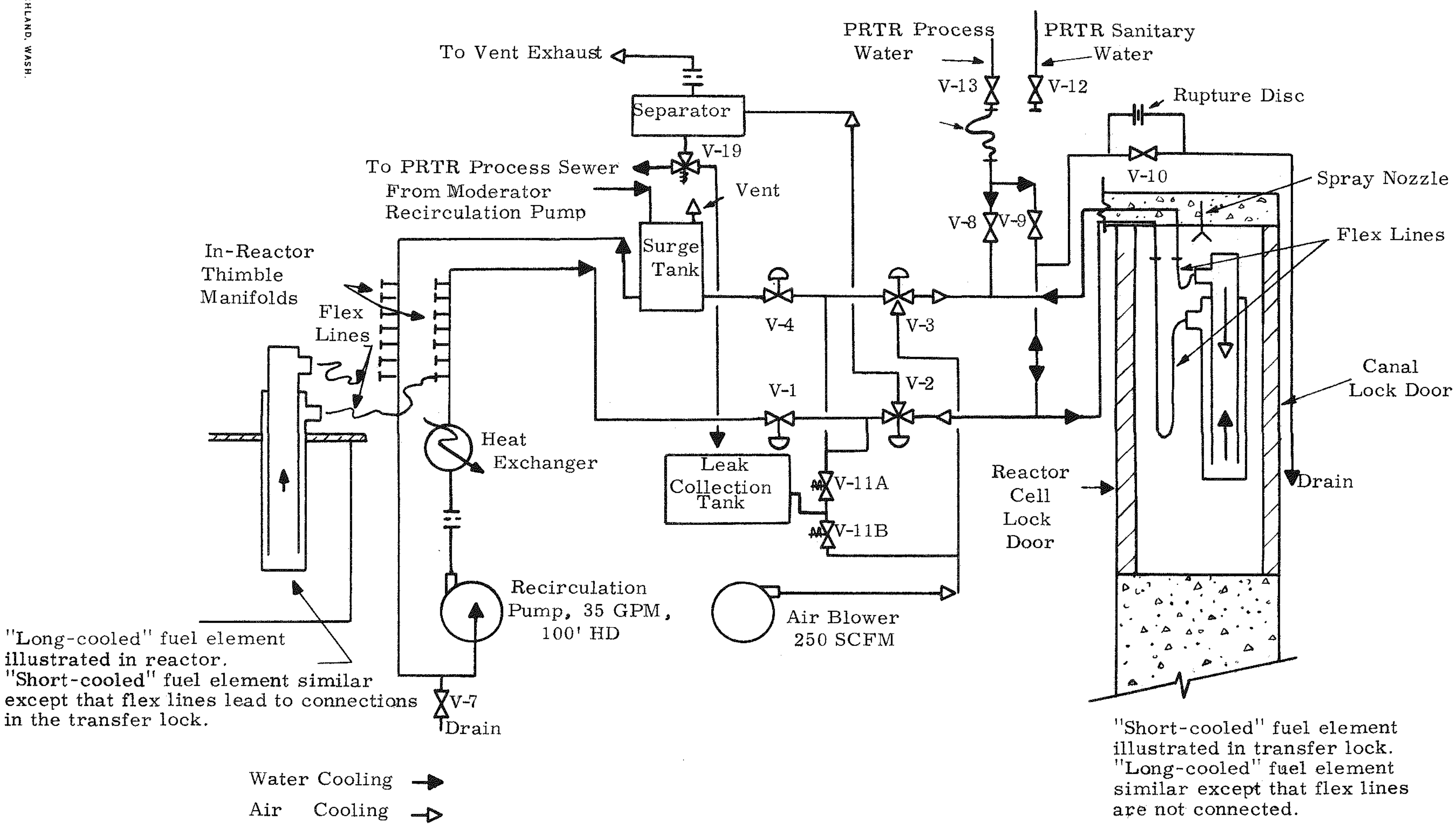

FIGURE 6

Irradiated Fuel Element Cooling System

tube, flows down the annulus formed by the two tubes, up the inner tube which contains the fuel element, and out of the assembly near the top of the inner tube. The inlet and outlet thimble fittings will be connected to the coolant system with flexible lines. Design of the thimbles is shown in Figure 7. Special Critical Facility fuel element hangers will form the cover for the thimble tube and provide a method of sealing against the internal thimble pressure once the irradiated fuel element is placed in the tube. The female portion of a quick-disconnect, water tight latch is welded to the top of the inner tube. The male portion of the latch will be welded to the top of the hanger.

Because of high radiation levels in the reactor cell when it contains preirradiated fuel elements, transfer and charging of such elements must be done remotely. All reactor coolant system connections must be made before fuel elements are brought into the reactor cell. As indicated earlier in this section, procedures for handling short-cooled and long-cooled fuel elements will be different.

Uncooled short decay time fuel elements would overheat because of fission product decay. Therefore, "short-cooled" fuel elements will be provided with continuous cooling whether the thimble and element are in the PRTR loadout canal, transfer lock, Critical Facility, or enroute. The cell side fuel transfer lock will be opened and the thimble will be connected to the coolant system inlet and outlet connectors in the lock by flexible lines. The cell-side fuel transfer lock door will then be closed, and the fuel transfer lock pumped full of water. The open thimble will also be filled with water. The canal side fuel transfer lock door will then be opened, the thimble transferred from the fuel transfer lock to the loadout canal, where the fuel element will be inserted. Air flow will force the light water from the thimble and flexible lines to the separator and cool the fuel element until evaporation has dried the system. When drying has been completed, the system will be filled with heavy water from the coolant system surge tank, and the recirculation pump started. The fuel containing thimble will

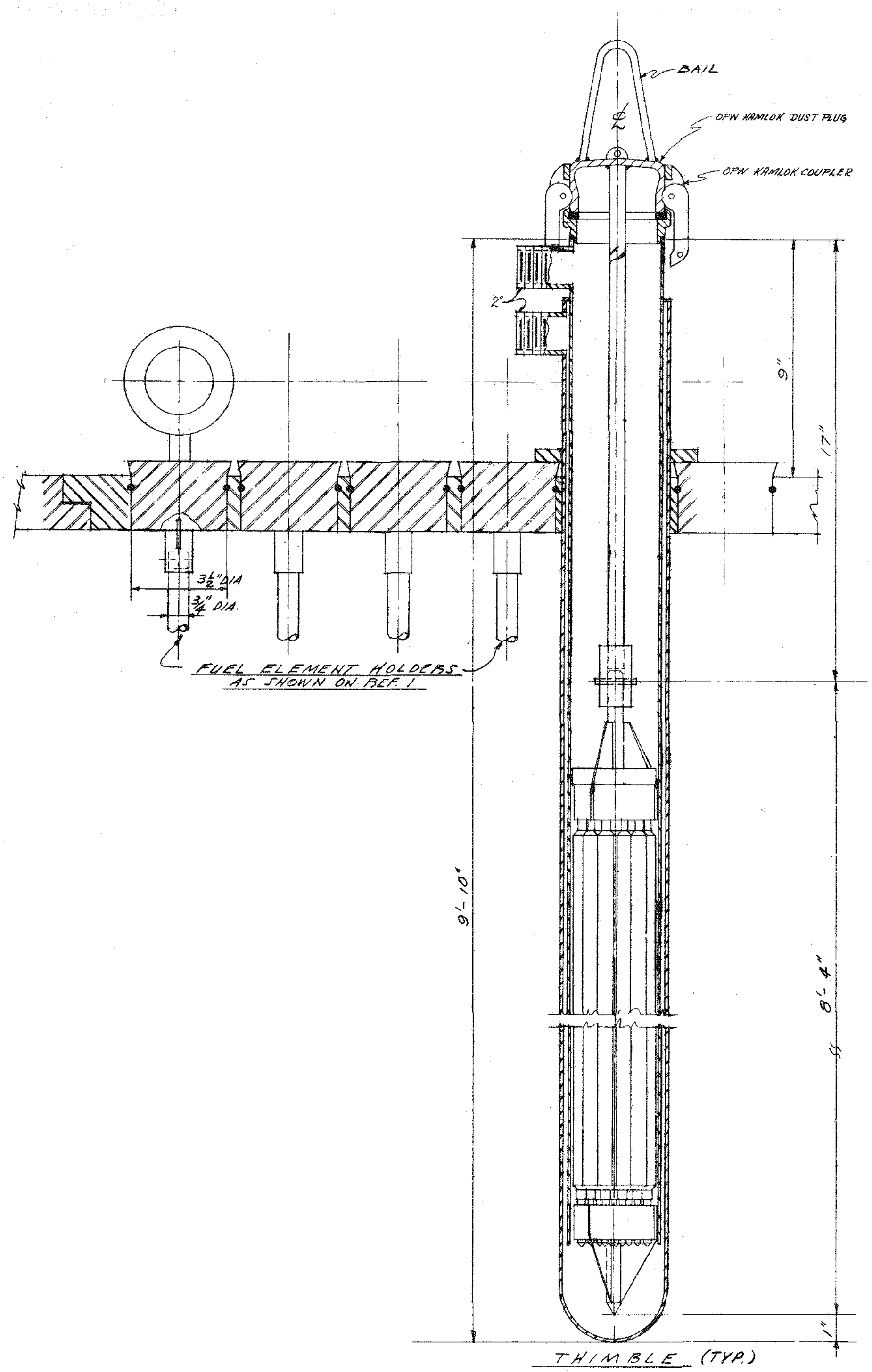

FIGURE 7

Fuel Element Thimble

then be transferred through the fuel transfer lock and placed in the reactor. The flexible line connections are inside the fuel transfer lock and the cell-side door will remain open while the thimble is in the reactor.

The following procedures will be used during the transfer of "long-cooled" fuel elements from the canal to the Critical Facility. Since circulating coolant will not be required during transfer, the isolation valves in the supply and effluent lines of the cooling system will be closed. The required number of thimbles (up to seven) will be placed in-reactor, connected to the cooling system manifold with flexible lines, and filled with heavy water. The transfer thimble will be moved from the cell to the water-filled canal where the fuel element will be inserted. The assembly will then be transferred from the canal back to the cell. Once inside the cell, the fuel element will be removed from the transfer thimble with the overhead crane, placed in an in-reactor thimble, and the assembly sealed with the latch mechanism. The same transfer procedure will be followed until the required number of "long-cooled" fuel elements are placed in the in-reactor thimbles. The recirculation pump is then started to supply forced circulation cooling to the fuel elements during reactor operation. During such operation both fuel transfer lock doors may be closed.

The above procedures for transfer of long-cooled and short-cooled fuel elements are tentative and are included for descriptive purposes.

Emergency Fuel Element Cooling System

An emergency coolant system has been provided to prevent melting the "short-cooled" fuel element in case of system ruptures, hose failures, or loss of flow. Emergency water supply is normally obtained by connecting a flexible line to the PRTR process water system. The flexible line may also be connected to the PRTR sanitary water system.

As shown in Figure 6, the system is arranged to provide flow to both the supply and effluent piping in the event of a system rupture, or in the case of a flow obstruction or pump failure to provide flow through the transfer thimble to a line which drains to the loadout canal. A rupture disc is installed in parallel with a valve in the line which drains to the canal. In the event that emergency water is required because of pump failure and the drain valve is closed, pressure buildup in the thimble will rupture the disc and allow water to flow through the thimble to the canal. The hose couplings and valves are located in the control area above the reactor cell.

Instrumentation and Control

Control Rods

Reactor control will be accomplished by means of three poison-type control rods. Each control rod consists of two hollow and concentric aluminum cylinders. The assembly is designed to allow the smaller, inner cylinder to travel vertically inside the outer cylinder. Total travel of the inner cylinder from the "full-in" position to the "full-out" position is six inches. Both cylinders are divided into 6-inch increments. Starting at the bottom, every other section of the outer surface of the outer cylinder is coated with cadmium (0. 020-inch thick). Essentially all thermal neutrons impinging on the cadmium surface will be absorbed. The remaining sections of the outer cylinder are transparent to neutrons. The inner cylinder is divided in a similar fashion except that starting at the bottom, every other section is transparent to neutrons. The remaining sections are coated with cadmium. When the inner cylinder is in the "dropped" or "full-in" position, the total poison surface is exposed to the reactor. The reactor is controlled by raising the inner cylinder, thus "shading" the poison surfaces of the inner cylinder with the poison surfaces of the outer cylinder.

The 1. 040-inch OD (including 0. 020-inch cadmium coating) inner cylinder is raised and lowered inside the 1. 200-inch ID outer cylinder with a lead screw driven by a reversible motor. The lead-screw ball nut is attached to an electromagnet, which holds the inner cylinder when energized and releases it when de-energized, allowing the cylinder to drop. Minimum diametral clearance between the inner and outer cylinder is 0. 144 inch.

The control rod assembly is enclosed in a 1.660-inch OD, 0.140-inch wall, air filled, aluminum thimble tube, to minimize frictional resistance while falling. The design criteria requires that the control rods shall be capable of inserting their entire negative reactivity contribution into the reactor in one second. The rod motor control switches are designed such that only one of the three control rods can be withdrawn at a time. A spring-loaded toggle switch is provided for each control rod, which requires the reactor operator to have his hand on the switch to move the rod in or out.

Design of the control rods is shown in Figure 8.

Safety Rods

Three vertical poison-type safety rods have been provided. Each rod consists of a 2.705-inch OD aluminum cylinder with the outer surface coated with a 0.020-inch layer of cadmium. The length of the poison surface is approximately 85 inches. The rod will be withdrawn from the reactor core with a lead-screw reversible-motor arrangement similar to that for the control rods. The lead-screw ball nut is attached to an electromagnet which holds the safety rod. The safety rod will be released when the electromagnet is de-energized, allowing the rod to fall freely to the maximum poisoning position. Lead screw rotation drives the ball nut and electromagnet down to magnetically engage the safety rod for withdrawal. The safety rod is surrounded by a 3.500-inch OD, 0.250-inch wall, aluminum-thimble tube which is pressurized to 5 psig with air. Minimum diametral clearance between the safety rod and the thimble tube is 0.250 inch. The portion of the safety rod thimble that extends above the grid plate is surrounded by a 3-1/2-inch Schedule 10 aluminum pipe. The pipe is to prevent damage that could be caused by the cell crane striking the thimble. Design of the safety rods is shown in Figure 9.

As in the case of the control rods, the design criteria requires that the total safety rod worth shall be inserted into the core in one second. Before the reactor is placed in operation, tests will be conducted to ensure

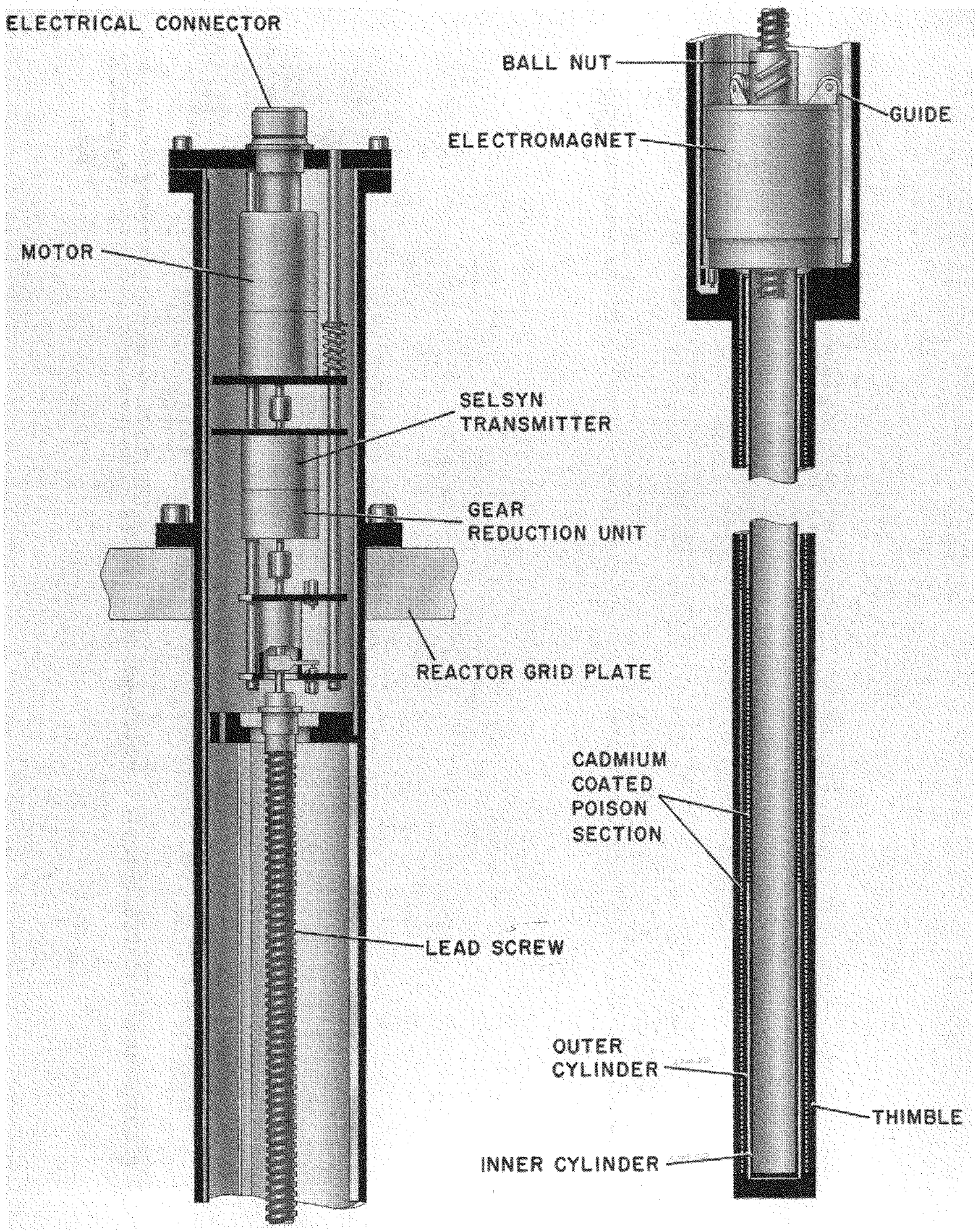

FIGURE 8
Control Rod

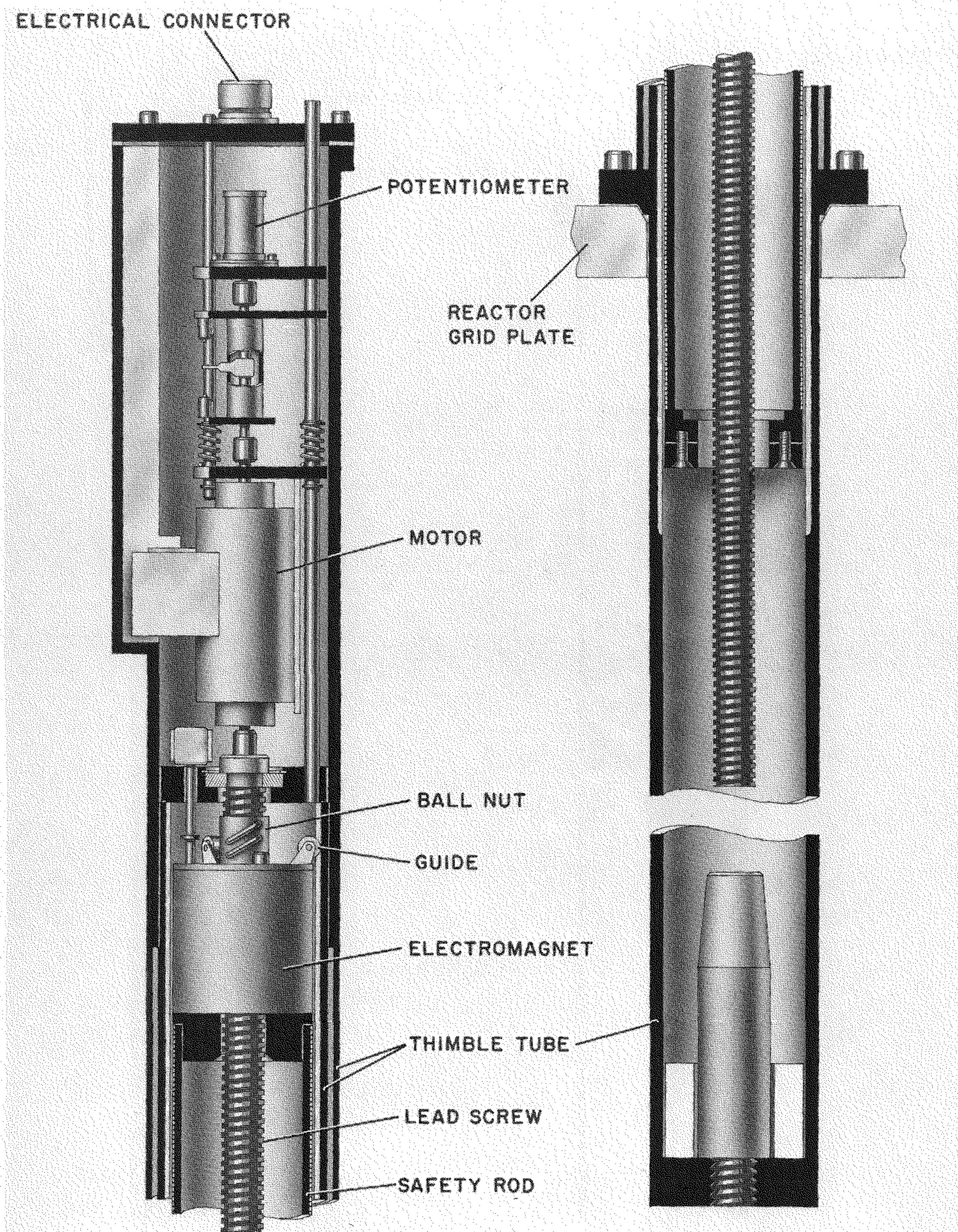

FIGURE 9
Safety Rod

that this criterion has been met for both the safety and control rods. The design is such that only one of the three safety rods can be withdrawn at a time. The "drive" switches for the safety rods are also spring loaded.

Neutron Source

A source capable of producing at least 10^6 neutrons per second is provided. The source will be inserted vertically into an aluminum thimble at the center of the reactor tank bottom plate and withdrawn into a shielded chamber by an air cylinder. The source will be inserted in a cup mounted on the end of the air cylinder piston rod. The piston rod is spring loaded to assure that the source is inserted into the core in case of air failure. The control rod drive motors are interlocked with the source positioner in such a way that the rods cannot be withdrawn unless the neutron flux instrumentation indicates a level equal to at least a 20-fold multiplication of the source. The source is inserted into the core automatically whenever the safety circuit trips.

Instrumentation

Instrumentation is provided to perform the following functions:

1. Provide the necessary information for operating the reactor and its auxiliaries.
2. Provide an automatic safety system which will protect the reactor and its auxiliaries from damage caused by equipment malfunction, operating errors, or other off-standard conditions.
3. Provide monitoring of radiation levels at various locations in the facility for the protection of operating personnel.

As shown in Figure 10, the nuclear instrumentation includes two startup channels; two logarithmic, intermediate-level channels; and two high-level safety channels.

The startup channels consist of a log count rate meter and scaler. Both channels provide startup flux level information. The log count rate channel also provides neutron flux and period protection during reactor start-up. The two intermediate channels overlap the upper part of the range of the startup instrumentation and cover reactor operation up to twice the normal maximum reactor power level. The intermediate channels provide flux level information and protection and reactor period information and protection. The two high level safety channels provide level information at high levels and actuate high flux level safety circuit trips. A level galvanometer and differential galvanometer, furnished in conjunction with one of the high level channels, provide the operator with an accurate indication of the reactor power level in terms of neutron flux. The six channels use either fission chambers or neutron-sensitive ion chambers as detectors.

The following process variables will be indicated or recorded at the Critical Facility control console:

1. liquid level indication - reactor tank,
2. liquid level indication - cell storage tank,
3. position indication - weir,
4. temperature indication - inlet or outlet coolant temperature, coolant system thimble,
5. temperature record - relative reactor moderator temperature, nine locations,
6. flow indication - coolant system (water),
7. radiation level indication - reactor cell and operating area (also recorded in PRTR control room) and
8. position indication - control and safety rods.

Exhaust air activity will be indicated and recorded in the PRTR control room. An inclined manometer, installed at the Critical Facility control valve panel, will be used to monitor coolant system exhaust air flow.

Safety Circuit

A relay-type safety circuit will automatically shut the reactor down whenever dangerous or off-standard conditions exist. Reactor shutdown will be accomplished by interrupting the current to electromagnets which hold the safety rods out of the reactor. Control rod insertion will also occur when the safety circuit is tripped. Fail-safe design and/or duplication

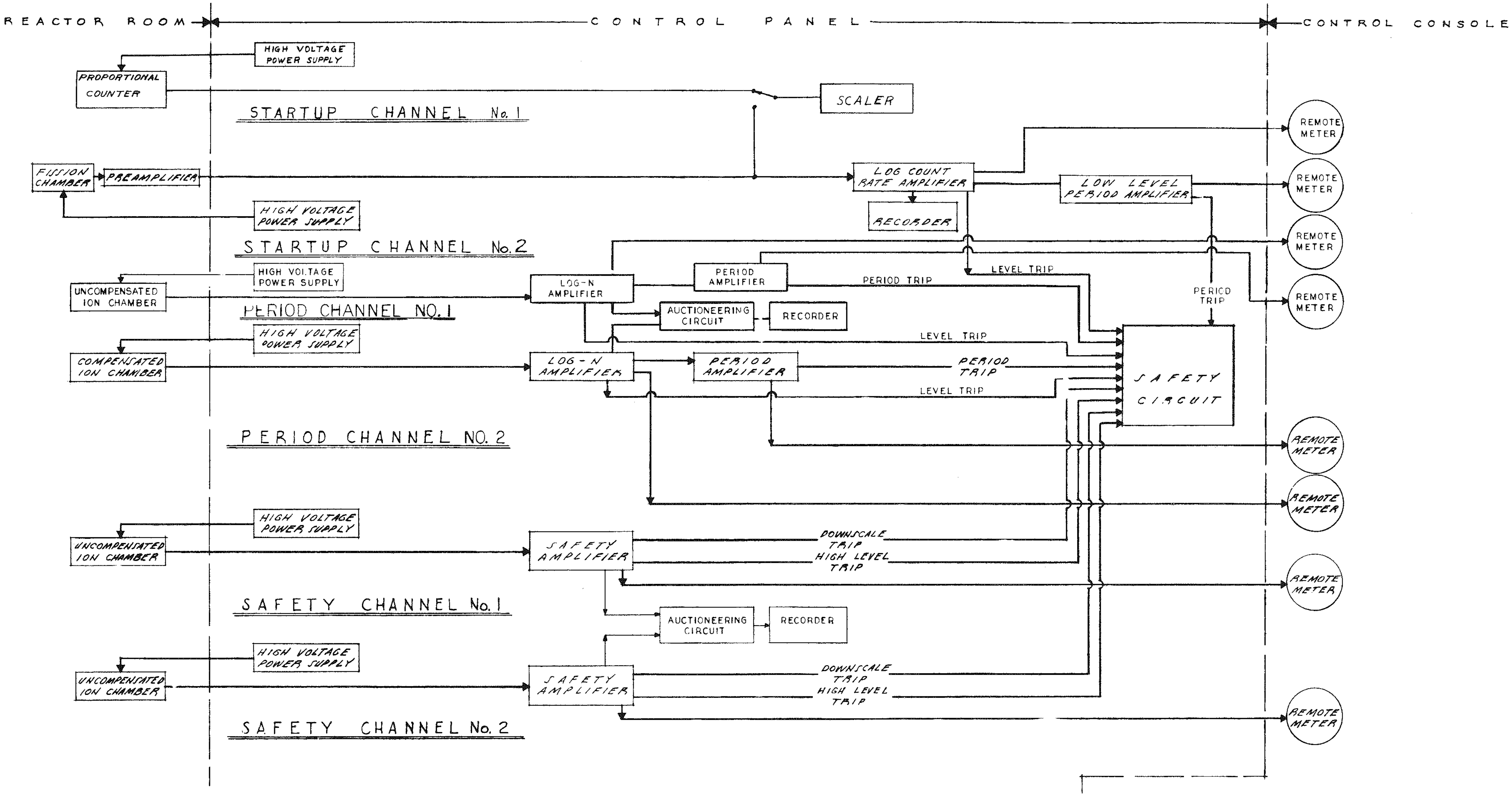

FIGURE 10
Nuclear Instrumentation

protects against failure of instrumentation and circuitry in the safety circuit. Conditions which will cause reactor scram are summarized in Table I.

TABLE I

SAFETY CIRCUIT TRIPS

	Trip Function	Number Channels	Trips For Scram	Bypass Switch
1.	Startup Channel High Flux	1	1	Yes[a]
2.	Startup Channel Short Period	1	1	Yes[a]
3.	Intermediate Channel High Flux	2	1	Yes[b]
4.	Intermediate Channel Short Period	2	1	Yes[b]
5.	High Level Channel, Low Level	2	1	Yes[b] [c]
6.	High Level Channel, High Flux	2	1	Yes[b]
7.	Earthquake Signal (PRTR Seismoscopes)[d]	3	2	- -
8.	Exhaust Air Activity (PRTR)[d]	3	2	- -
9.	Aqueous Effluent Activity (PRTR)[d]	3	2	- -
10.	Removal of Reactor Cell Access Plugs	1	1	No

(a) Bypassed by operator after intermediate channels are on scale.

(b) One of four combinations may be bypassed at the discretion of the operator. A bypass switch with the following five positions is used:

1. Bypass High Level Channel Number 1 Low Level and High Flux
2. Bypass High Level Channel Number 2 Low Level and High Flux
3. Unbypassed
4. Bypass Intermediate Channel Number 1 Short Period and High Flux
5. Bypass Intermediate Channel Number 2 Short Period and High Flux

(c) Bypassed during startup until indication is above High Level Channel Low Level Trip

(d) PRCF safety circuit actuated by PRTR safety circuit relays. PRTR trips are triplicated and coincident circuitry is used; trip of two of three sensing elements is required to initiate scram.

Safety circuit relay trip contacts, except the reactor cell access plug contact, are in series with four parallel rod-circuit-actuating relays. Two of these relays have contacts in series with the electromagnets of one safety rod and two control rods ("A" circuit). The contacts of the remaining two relays are in series with the electromagnets of two safety rods and one control rod ("B" circuit). Opening of safety circuit relay contacts (except the cell access plug contact) will interrupt current to all safety and control rod electromagnets. The cell access plug contact will trip the "A" circuit only. The safety circuit may also be actuated manually by push buttons installed at the control console and in the reactor cell. In addition, the safety circuit will automatically trip if the flux indication is above the high level channel low trip point and at least two of the safety rods are not out of the reactor core. The safety circuit is shown in Figure 11.

The startup channel high flux and short period trip functions may be bypassed by the operator after the intermediate channels are on scale. The startup channel requires high sensitivity to reliably monitor low neutron fluxes. The high sensitivity needed for startup is a disadvantage at higher flux levels and results in saturation of the startup channels at relatively low neutron flux levels.

Of the four high level trips in the two intermediate channels and the two high level channels, one of the four may be bypassed at any time at the discretion of the operator. When the high level trip in either intermediate channel is bypassed, the period trip for that channel will also be bypassed. When the high level trip in either high level channel is bypassed, the low level trip for that channel will also be bypassed.

The low level trip feature of the high level channels will be bypassed on startup until the channels indicate above the trip point. When both channels are above the trip point, an annunciator will automatically sound to warn the operator to unbypass these trips.

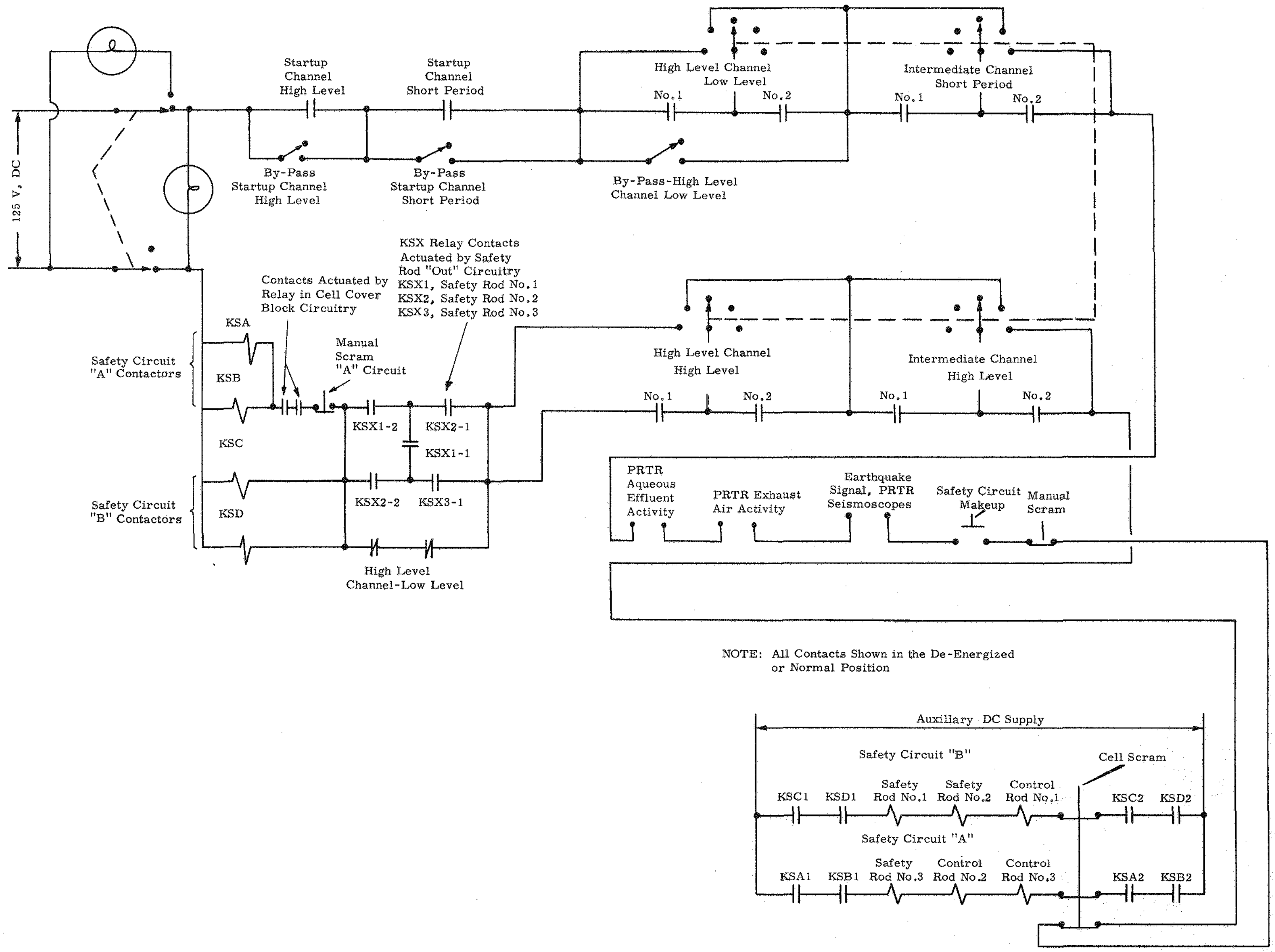

FIGURE 11

Safety Circuit

Annunciators and Alarms

The following signals will actuate audible and visual alarms at the control console:

1. high and low liquid level - cooling system surge tank,
2. high liquid level - fuel transfer lock sump,
3. coolant high temperature - coolant system thimble inlet or outlet,
4. high temperature - reactor moderator,
5. abnormal coolant flow - coolant system,
6. moisture on the reactor cell floor,
7. high radiation level in the reactor cell or the operating area,
8. safety circuit A or B tripped,
9. high flux level channels above low level trip and bypassed,
10. startup or intermediate channels down scale,
11. ion chamber power failure, and
12. PRTR seismoscope tripped.

A monitor, furnished in conjunction with the scaler, will provide an audible indication of increasing subcritical flux levels. The monitor, which provides a signal of increasing intensity upon increasing flux levels, has speakers in the operating area and the reactor cell. A criticality alarm is also installed in the reactor cell.

Description of Building

The Critical Facility building consists of a reactor cell and an operating area. It adjoins the PRTR service building and is adjacent to the PRTR loadout canal. Figures 12 and 13 show the building layout and its position in relation to existing PRTR facilities.

The operating area, located above grade in a 37-foot by 14-foot building, houses the reactor control console. The space not required as a reactor control area serves as a work area. Removable sections in the light weight concrete roof deck provide access for installation and removal

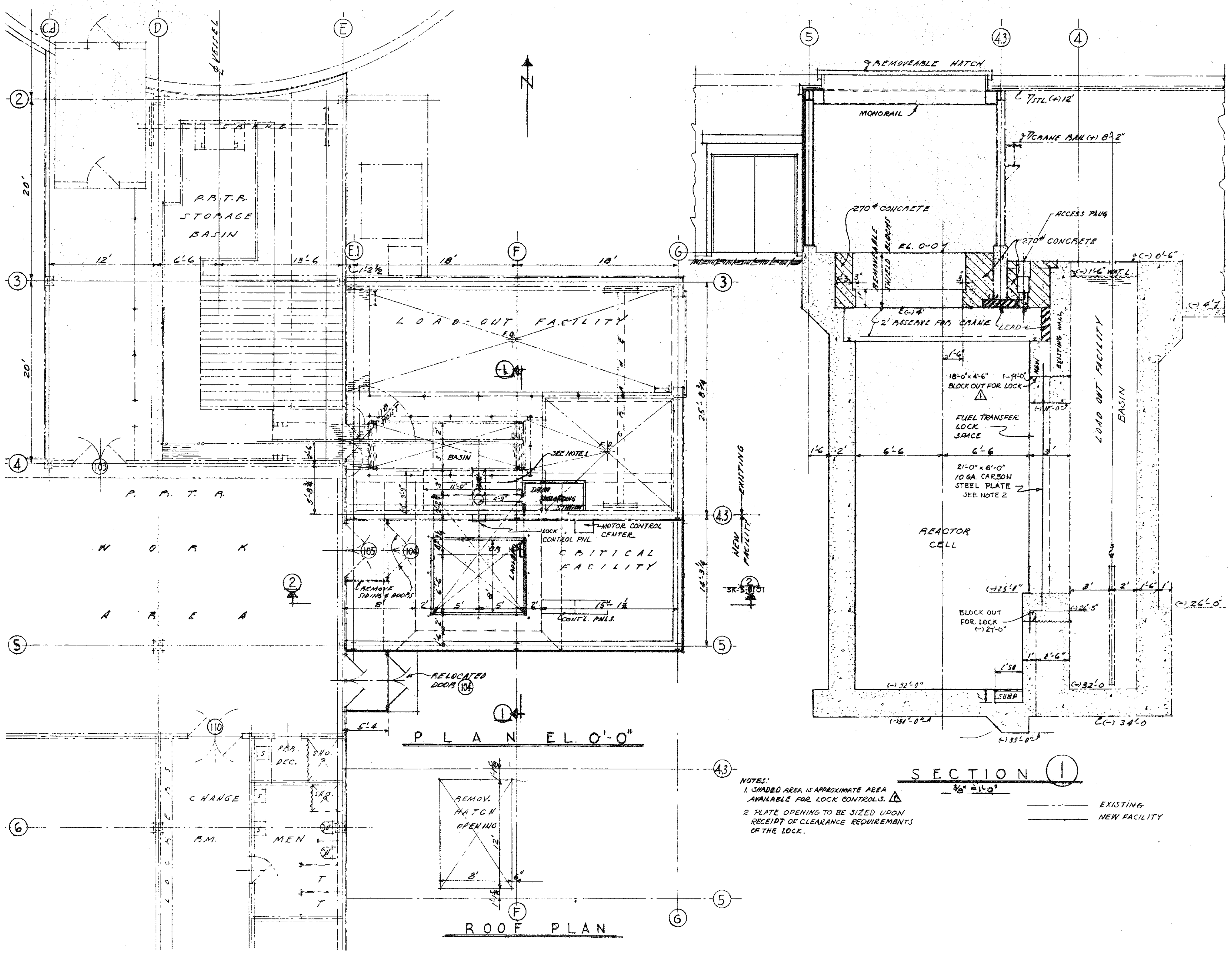

FIGURE 12

Building Plans at Elevation 0' - 0" and West Elevation

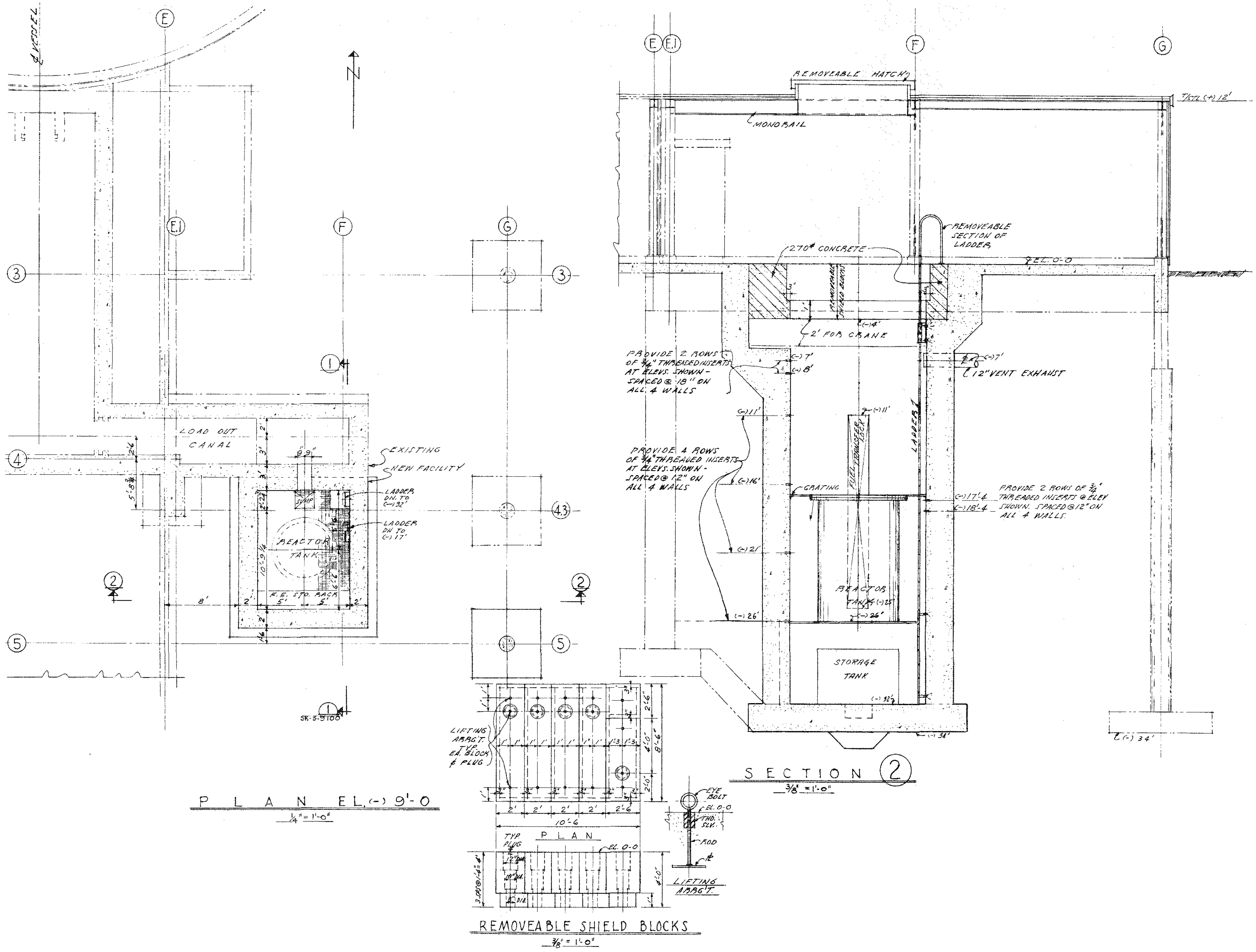

FIGURE 13

Building Plans at Elevation (-) 9' - 0" and North Elevation of heavy equipment using a mobile crane. Approximately 500 square feet of floor space are provided in the operating area.

The concrete reactor cell is located below grade. The 10-foot wide by 13-foot long by 32-foot deep cell houses the reactor tank, fuel element transfer facilities, moderator and cooling systems, and unirradiated fuel element storage racks. A grating at the minus 17-foot level provides access to the top of the reactor tank.

The reactor cell walls are designed to shield the surrounding facilities during irradiated fuel element transfer and reactor operation. The shielding is also designed to give a maximum radiation level of 10 mr/hr in the operating area. The wall between the reactor cell and the PRTR loadout canal is three-foot thick ordinary reinforced concrete, except for the 1-1/2 x 14-foot irradiated fuel transfer lock. The other three walls and the bottom floor of the cell are two-foot thick ordinary reinforced concrete, surrounded by earth. The reactor cell cover is four-foot thick, high density (~270 lb per cu ft) concrete. An eight-by ten-foot access hole through this cover will be closed with stepped, high-density concrete blocks.

The reactor cell is designed to provide confinement in the event that radioactive materials are released. The access opening cover blocks are equipped with replaceable neoprene gaskets. The inlet and exhaust ventilation lines are equipped with confinement valves which automatically close upon receipt of a PRTR exhaust air activity trip signal.

A periscope allows viewing of operations in the cell. Viewing is possible from the south, west, and east walls of the cell using a single, movable, monocular periscope. Periscope sleeves with viewing outlets at -15, -21, and -30 feet are installed in the three walls.

Auxiliary Services

Auxiliary services provided in the Critical Facility include the following:

1. Untreated process water from the PRTR process water system. This water will be used for cooling in the heat exchanger, as emergency backup cooling for the fuel element thimbles and for filling the fuel transfer lock. This system will provide flows up to 100 gpm.
2. Control and instrument air obtained from the PRTR compressed air system will be used to inflate the fuel transfer lock door seals and supply pneumatic instruments and controls.
3. A breathing air manifold, supplied by the PRTR system, is located in the reactor cell.
4. Electrical services will be supplied by the Bonneville Power Administration (BPA) system. Critical components such as the coolant system recirculation pump and air blower, the reactor cell crane and sump pump, and instrumentation systems will be supplied with power from the PRTR emergency diesel generator in the event of BPA power failure.
5. Liquid wastes will be piped to the PRTR sewer, from which they can be diverted to either uncontaminated or contaminated sewers. Manually operated valves permit diverting the Critical Facility wastes to the PRTR underground waste holdup tanks. A valve, which automatically closes upon receipt of a PRTR aqueous effluent trip signal, is installed in the Critical Facility waste line.
6. A heating and ventilating system consisting of a heat pump supplemented with steam heating coils will maintain the operating area temperature at 75 F. The reactor process cell is equipped with a supply and exhaust system with a 500 scfm capacity. The cell supply system consists of a blower, confinement valve, damper, and six-inch supply line installed in the west wall of the cell and extending into the cell at the minus 30-foot level. The exhaust system consists of a 12-inch line connecting the cell with the PRTR ventilation exhaust fan.

The line extends into the cell at the minus seven-foot level, then underground to the PRTR exhaust fan pit. Inside the pit the line is reduced to 6 inches, equipped with a confinement valve and damper, and tied into the PRTR exhaust system between the PRTR containment valve and the PRTR exhaust fan. A bypass line is installed in the exhaust side, around the Critical Facility confinement valve. A check valve (fully open - 2 psig) and two activated charcoal filters (each filter having a 95 percent removal efficiency for halogens) are installed in the bypass line.

If the air leaving the Critical Facility becomes contaminated, signals from the activity monitors in the PRTR exhaust line will close the PRTR ventilation containment valves, stop the PRTR exhaust fan and the Critical Facility supply blower, and close the Critical Facility ventilation confinement valves, shutting off the normal exhaust path for the reactor cell air. If the pressure in the cell reaches 2 psig, the exhaust air will be routed through the check valve and the charcoal filter in the bypass line, through the PRTR absolute filter (99.97 percent removal efficiency for particulates 0.3 micron and larger), the PRTR charcoal filter (98 percent removal efficiency for iodine), to the 150-foot high PRTR exhaust stack.

Fuel Handling System

Unirradiated fuel elements will be lowered into the reactor cell through access openings in the operating area floor, and will be placed in storage racks along the reactor cell wall. These fuel elements will be loaded into, and unloaded from, the reactor with the overhead crane in the cell. The moderator will be drained from the reactor tank whenever fuel element assemblies are loaded into the Critical Facility. Electrical and mechanical interlocks are provided to prevent positioning the cell crane over the reactor tank whenever there is moderator in the core. A limit switch actuated by the crane will de-energize the moderator pump motors when the crane is moved from the parked position, which is clear of the reactor core. A float-switch-actuated interlock will de-energize the crane drive motor whenever there is any moderator in the tank. A drain valve, opening to the cell and large enough to lower the moderator level 3 feet in the time required to move an element over the core, is installed on the bottom of the reactor tank. This drain valve is equipped with a rising stem which actuates a mechanical stop to prevent moving the crane from the parked position unless this valve is open. The mechanical stop is also designed to prevent closing the drain valve when the crane is not in the parked position. The valve-crane mechanical interlock is shown in Figure 14.

The system for transfer of fuel elements which have been irradiated in the PRTR includes the following:

1. fuel element thimbles, designed so that they may be attached by flexible lines to the irradiated fuel element coolant system,
2. a 1-1/2 x 14-foot fuel transfer lock in the wall between the PRTR loadout canal and the reactor cell,
3. a swinging bracket attached to the side of the fuel transfer lock to transfer fuel element thimbles through the lock, and
4. the overhead crane in the reactor cell.

The fuel element thimbles and coolant system have been described in Section III (Description of Reactor: Moderator and Coolant System; Irradiated Fuel Element Cooling System).* The fuel transfer lock is designed as shown in Figure 15. Deflation of the inflatable seals on the fuel transfer lock doors would result in a maximum leak rate of 50 gpm, the capacity of the sump pump. Piping systems for filling and draining the lock are provided.

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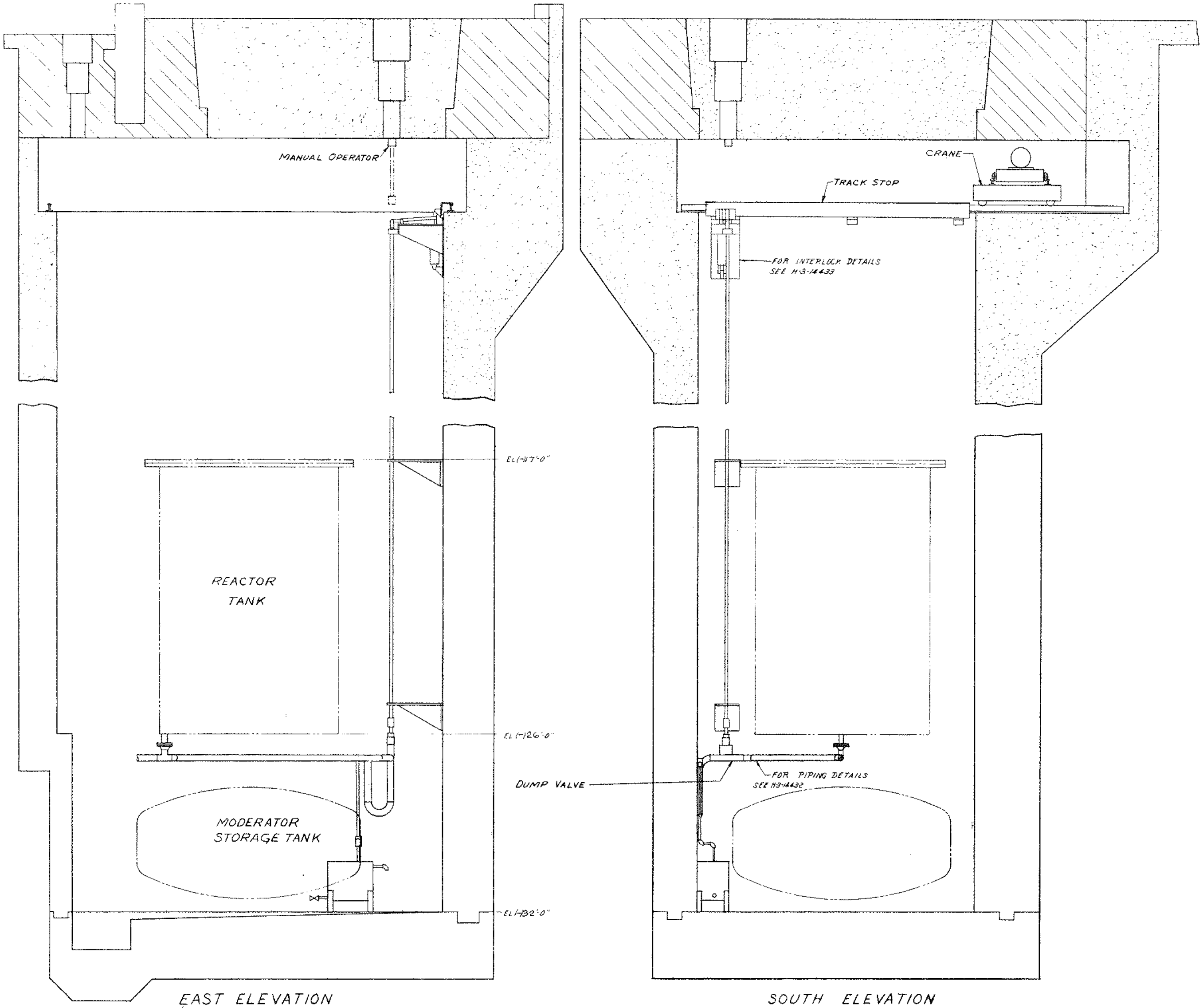

FIGURE 14
Cell Crane-Reactor Tank Dump Valve Interlock

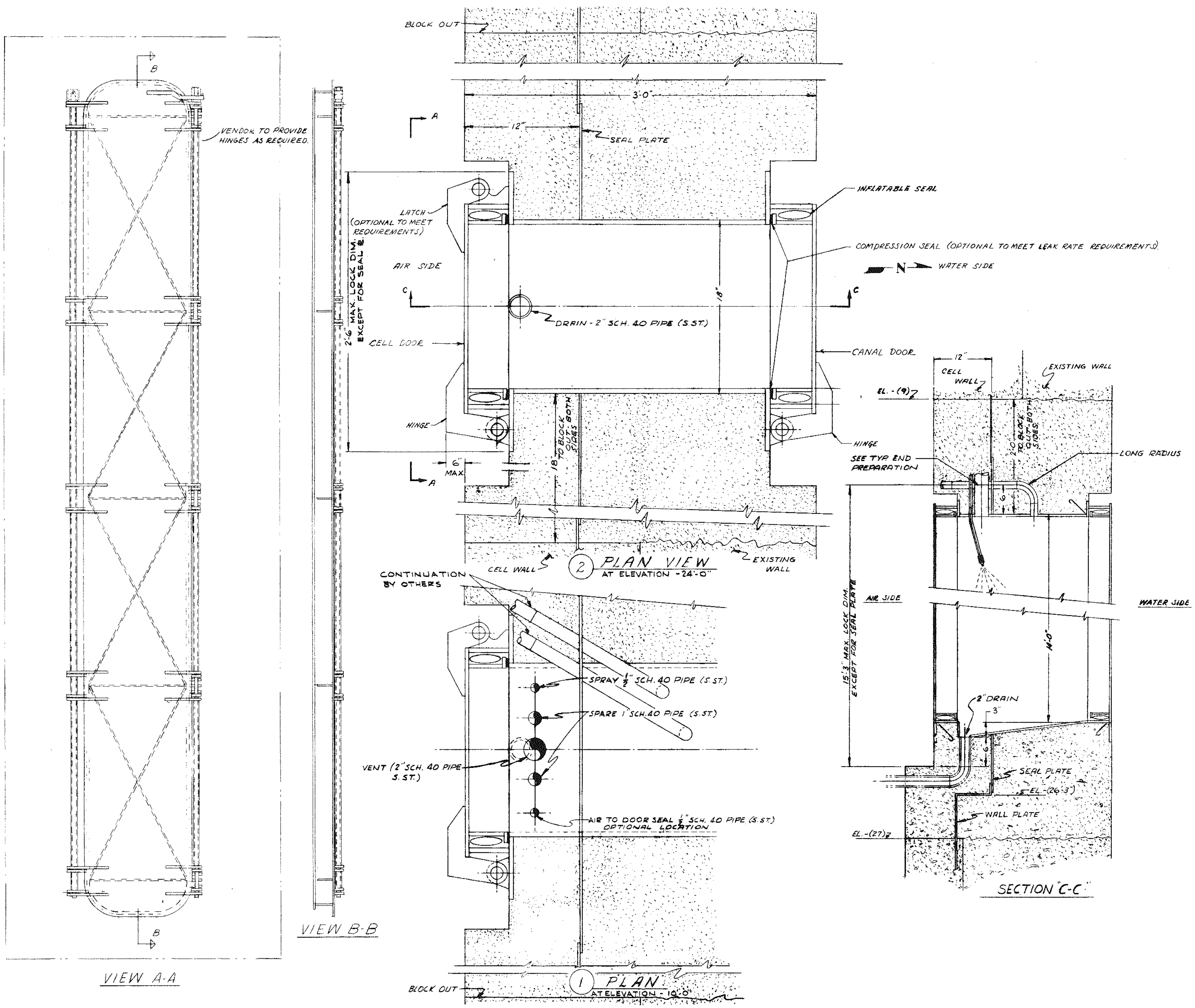

FIGURE 15

Air-To-Water Fuel Transfer Lock

IV. SITE

The Critical Facility is located adjacent to the loadout and work areas of the Plutonium Recycle Test Reactor. This site is at the southeast corner of the Hanford Works restricted area in Southeastern Washington. The location of the PRTR facility in relation to other Hanford Works 300 Area facilities, and the distribution of communities around the restricted area are shown in Figures 16 and 17.

A detailed discussion of the site was presented in the PRTR Safeguards Analysis(1) and will not be repeated here. Included in this discussion were the following:

1. location,
2. geology,
3. hydrology,
4. meteorology,
5. seismology,
6. 300 Area facilities, and
7. makeup of surrounding area.

FIGURE 16

Plot Plan

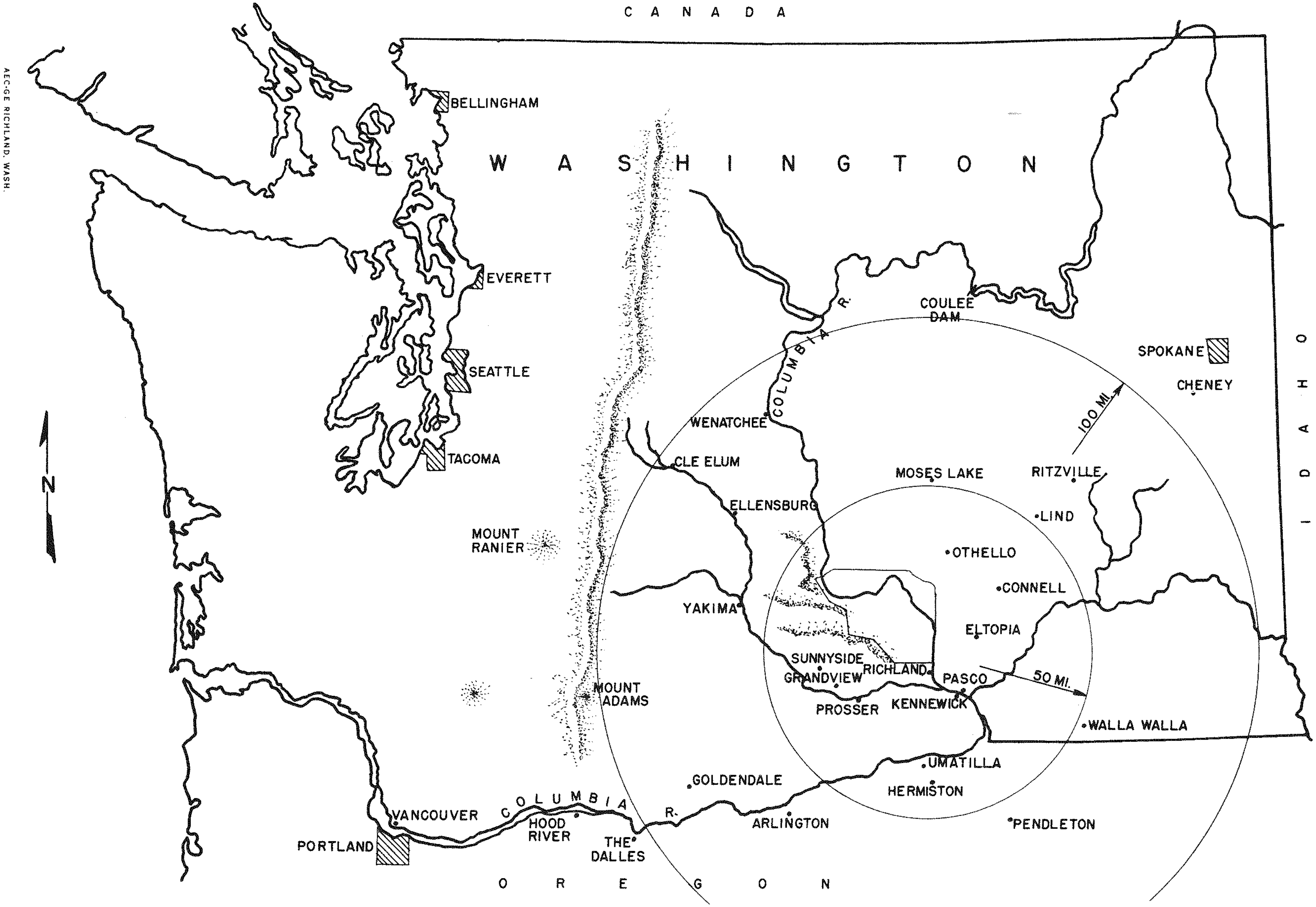

FIGURE 17

Hanford Site Location

V. OPERATION OF FACILITY

Operating Program

The Critical Facility will be used for the determination of basic nuclear constants of heterogeneous reactors recycling plutonium, for reactivity measurements, and for reactor physics experiments (exponential and criticality studies) which provide data needed for the design, operation, and optimization of reactors which use plutonium as enrichment. The proposed program for the facility includes the following types of experiments:

1. reactivity measurements of irradiated and unirradiated PRTR fuel elements in a D_2O moderated lattice,
2. measurements which will assist in planning new PRTR fuel loadings and in interpreting data from the PRTR,
3. measurements which will assist in planning fuel loadings for reactors recycling plutonium,
4. reactivity measurements of reactor fuels and materials,
5. exponential, approach-to-critical, and critical experiments with reprocessed PRTR fuels, and
6. measurements of the buckling of lattices fueled with plutonium-enriched fuel elements.

One of the primary aims of the Critical Facility will be to obtain measurements of changes in reactivity associated with changes in exposure of PRTR fuel elements. A typical reactor loading for a test of this type is included for descriptive purposes. The reactor loading will consist of a test region, a buffer region to produce a predetermined flux spectrum in the test region, and a driver region containing enriched fuel elements. The reactivity change of a fuel element during an irradiation period in the PRTR can be estimated well enough in advance so that by using reactivity standards, as a stand-in for the irradiated fuel element and substitution techniques, the measurement can be made with minimum transfers of the irradiated fuel element in and out of the reactor cell.

Experiments will be conducted with the reactor subcritical or with the reactor operating at power levels up to 100 watts. Initial experiments will be conducted in a D_2O moderated lattice. At a later time, experiments in light water-moderated lattices are planned. Additional safeguards analyses for such operation will be made before light water experiments are conducted. For at least the first six months of facility operation, the reactor will be moderated with heavy water.

Organization for Operation

Test Reactor and Auxiliaries Operation (TRAO)* will be responsible for the safety, operation, and maintenance of the Critical Facility. Responsibility will be delegated as follows.

Critical Facility Operation

The Critical Facility will be operated by technicians and technologists under the direction of the Supervisor, PRCF operation. The supervisor will plan and direct the operation of the PRCF and assure operational safety of the facility. The supervisor will report directly to the Manager, TRAO. Before operating the Critical Facility, all operations' personnel will be required to pass written and oral examinations analogous to licensing examinations given by the AEC to individual operators of privately owned facilities.

Critical Facility Maintenance

Two components of TRAO will plan and perform maintenance work. Maintenance and Equipment Engineering Operation (MEEO) will prepare maintenance procedures and standards, engineer improvements and modifications and issue design changes, and provide engineering assistance to PRTR maintenance. The PRTR Maintenance Organization will schedule and perform preventive and repair maintenance, and improvement work.

Planning

The Technical Planning Operation of TRAO will schedule experiments and provide detailed plans for maximization of the utility of the facility, issue Process Standards, and provide technical assistance to operating personnel.

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* Test Reactor and Auxiliaries Operation is a Hanford Laboratories section, presently responsible for operation and maintenance of PRTR, as well as the conduct of technical planning for operating the PRTR.

Start-Up Program

Purpose

Various design tests and physics tests of the reactor are planned for the start-up phase. The tests have the following objectives:

1. demonstrate that the facility meets the functional requirements set forth in the Design Criteria,[2]
2. determine the physics, control, and operating characteristics of the facility, and
3. demonstrate that the facility can be operated safely and efficiently.

Design Tests

Safety System

Safety circuit components will be checked by simulating off-standard conditions. Test data will include rate of insertion, time delay for relay actuation, and rate of withdrawal for safety and control rods.

Coolant System

Tests will be conducted to verify the adequacy of the normal cooling system (air and water) and the emergency cooling system. Tests will be conducted to determine flow transients during coolant change over from water to air.

Moderator System

The rate of increase and decrease of the moderator level will be checked. Tests will be performed to determine the rate of rise of the moderator both below the lower limit of weir travel and within the limits of weir travel.

Fuel Element Loading and Transfer

Tests will include functional checks of the cell crane and the fuel transfer lock. The leak rate of the transfer lock will be determined.

General

Design tests will also include items such as electrical system tests (including verification of interlocks), instrumentation calibration, leak rate checks of systems, volumetric calibration of systems, functional tests, and shielding adequacy tests.

Physics Tests

As contemplated, physics tests will be divided into three categories: base-loading tests, safeguards evaluation tests, and sensitivity tests. A two-zone loading to critical (central UO_2 region surrounded by Pu-Al driver fuel elements) is planned for the base loading. Test data will be taken to determine safety rod and control rod strength. The loading will then be changed to accommodate tests of the following types:

1. void coefficient as a function of position in the reactor core,
2. light water substitution in thimbles in the test region of the core,
3. determination of maximum worth of driver fuel elements,
4. kinetics experiments,
5. light water reflector worth experiments,
6. control rod reproducibility,
7. moderator level sensitivity,
8. temperature coefficient (by heating the moderator to approximately 105 F),
9. cell pressure reactivity coefficient,
10. power calibration, and
11. continued verification of safety and control rod strength.

Operating Procedures

There are three basic operations which will be conducted in the Critical Facility. These are:

1. loading and unloading of fuel,
2. handling of irradiated fuel and samples, and
3. operation of the reactor.

Anticipated procedures are described in this section. Detailed written operating procedures for all standard operations (including check lists and operating limits) will be prepared, tested, and approved prior to reactor operation. The procedures will cover items such as loading and unloading fuel, fuel transfer, procedures for approach-to-critical, emergency procedures, etc.

Unirradiated Fuel Loading

The term "unirradiated" as defined here includes elements which have been irradiated only in the Critical Facility reactor. Such fuel elements will be subjected to maximum exposures of 5 kilowatt-days per ton of uranium per year of operation. Unirradiated elements will be brought into the work area and fitted with special hangers. Hangers for enriched and unenriched elements will be obviously different in appearance, and will be designed in such a manner that hangers for unenriched elements cannot be attached to enriched elements. After the hangers have been attached, elements will be lowered into the reactor cell through access holes and placed in storage racks. The operating area crane will be used in this transfer. The fuel elements will be loaded into, and unloaded from, the reactor with the overhead reactor cell crane. Unirradiated fuel elements will be guided by hand. The one safety rod and two control rods controlled by safety circuit A will automatically be inserted when personnel remove a cover block to enter the cell. Incremental loading procedures will be used.

Irradiated Fuel Loading

The handling of irradiated PRTR fuel elements and radioactive samples must be carried out remotely, with no personnel in the reactor cell, and with the access holes sealed. Irradiated fuel elements will be transferred from the PRTR loadout canal to the reactor cell through the fuel transfer lock. The periscope will be used to observe the transfer and loading of the fuel element. Irradiated fuel elements will be cooled during transfer and while positioned in the reactor. The cooling system is described in Section III (Description of Reactor: Moderator and Coolant

Systems; Irradiated Fuel Element Cooling System)* and Section III (Description of Reactor: Moderator and Coolant Systems; Emergency Fuel Element Cooling System).** At least two qualified persons will be present in the facility to perform and observe the transfer and take any necessary emergency actions. Procedures for handling irradiated fuel elements are described in Section III (Description of Reactor: Moderator and Coolant Systems; Irradiated Fuel Element Cooling System).*

Loading Procedures

Fuel Element Charging

Fuel element assemblies (e.g., cluster-type assemblies composed of 19 rods) will not be loaded or unloaded unless the moderator is drained from the reactor tank, as assured by two electrical interlocks and one mechanical interlock. The two safety rods controlled by safety circuit B will be cocked so that a safety circuit trip will cause their insertion. The one safety rod and two control rods controlled by circuit A will be fully inserted in the reactor.

The first fuel elements loaded will be placed in the most reactive positions. Incremental loading procedures will be used. After an increment of fuel is loaded, personnel will leave the reactor cell and replace the access opening shield plug. The one safety rod controlled by safety circuit A will be withdrawn and the moderator will be raised to the operating level.

The final addition of reactivity will be made with the control rods. Before loading the next fuel increment, the one safety rod and two control rods will be reinserted and the moderator drained from the core.

Fuel element loading increments will be chosen conservatively to prevent unexpected achievement of criticality.

The reactivity change of a loading increment will not exceed one half of the calculated amount to achieve criticality (with all safety and control rods withdrawn and the moderator at operating level) for cases where the inverse multiplication curves show a consistent, logical, negative slope for

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** Page 19

a minimum of the three previous consecutive additions. This procedure applies until the loading has progressed to within an estimated 1 percent $\Delta K/K$ of critical.

The reactivity change of initial loading increments will not exceed one quarter of the calculated amount to achieve criticality. After the initial loading increments, if the inverse multiplication curve has a positive slope or a slope which is changing to positive the reactivity change of a loading increment will not exceed one quarter of the calculated amount to achieve criticality (with all safety and control rods withdrawn and the moderator at operating level), and in no case will exceed a calculated 1 percent $\Delta K/K$ of critical.

When the critical loading is calculated to be 1 percent $\Delta K/K$ or less away (with all safety and control rods withdrawn and the moderator at operating level) and the inverse multiplication curves show a consistent negative slope pattern for at least the three previous consecutive steps, the final single step addition may be as large as 1 percent $\Delta K/K$. Insofar as possible, the excess reactivity after the final fuel addition will be limited to one dollar with the control rods withdrawn and the moderator at the maximum attainable position for the experiment.

Other Core Changes

During start-up operation, changes of core configuration other than charging fuel element assemblies will also be accomplished with the moderator drained from the reactor tank. Core changes of this type will be controlled to also limit the available excess reactivity to one dollar.

These core changes may be made by hand. After the items are added or removed, the reactor will be taken critical with inverse multiplication measurements to determine the reactivity status.

Later, the specifications may be relaxed to allow this type of core change to be made when the reactor is subcritical by at least 7.5 mk or twice the expected reactivity change, whichever is larger, provided that start-up experience indicates that adequate control of nuclear safety will be preserved. The reactivity change resulting from any such core alteration would be less than 7.5 mk.

Experiments of a Repeating Nature

Many experiments performed in the Critical Facility will be of a repeating nature. An example of this will be the basic reactor loading used to determine the reactivity worth of fuel elements irradiated in the PRTR. The reactor may be fueled to a previously determined reactivity status based on test experience. Although the reactivity status of the core will be determined at various stages, loading operations of a previously investigated core configuration may be performed without following the detailed incremental loading procedure described in the previous section. After the last fuel element has been charged, multiplication data will be taken to confirm the expected reactivity status.

Reactor Operation

In general, reactor operation will consist of taking period measurements to establish the reactivity worth of various materials, fuel elements, and lattices, and of operating at low power levels to obtain flux distribution measurements.

Criticality will be achieved by withdrawing control rods. Comparisons of reactivity worths of various samples and loadings will be made by comparing the control rod positions at criticality.

In the operation of the Critical Facility, the available excess reactivity will be limited to one dollar, insofar as possible. Most of the presently planned experiments can be performed with the excess reactivity limited to one dollar. For performance of all of the planned experiments, the maximum amount of excess reactivity required is two dollars. Experiments requiring greater than one dollar excess reactivity will be performed only after a thorough review to assure that no reasonable alternate method exists for performing the experiment within the one dollar criterion and that procedures to be followed will assure preservation of reactor safety.

During startup, reactor instrumentation will be observed continuously. This instrumentation includes period and flux level meters. In addition, an audible monitor will indicate increasing flux levels. During approach-to-critical experiments, neutron flux multiplication measurements will be made for prediction of the critical point.

The reactivity worth of the control system will be used to allow reactivity comparisons of a fairly wide range of samples and lattices. The operating conditions will produce no significant temperature changes, fuel burnout, or poison buildup. Thus, the control system will not need to compensate for such effects.

The design of the control rods is such that there will always be poison in the reactor core when the control rods are positioned in the reactor. Special procedures will be used to physically remove the control rods from the reactor core when experiments are performed to obtain the poison free characteristics of various reactor loadings. The strength of the control rods while positioned in the loading will be determined prior to performing the experiment. The reactor will be taken only to 99 percent $\Delta K/K$ of critical during control rod removal. The source will remain inserted in the reactor throughout the experiment.

Initially, the reactor will be loaded to be subcritical by 1 percent $\Delta K/K$. One control rod will then be moved to a position of less worth. The estimated worth of the rod in the new position will not be less than one half of its worth in the previous position, and the reactivity addition from moving the rod will not exceed one dollar (It will be necessary to factor the placement of a fuel element in the position that formerly contained a control rod when the control rod is moved). After the control rod is moved, multiplication data will be taken. The above steps will be repeated until available excess reactivity upon removal of all or any of the three rods will be equal to or less than one dollar. The moderator will be drained from the tank each time a control rod is moved to a position of less worth and when control rods are physically removed. Multiplication data will be taken after each step of control rod movement or control rod withdrawal.

Control of Experiments

Responsibilities for the control of tests in the Critical Facility will be assigned (as described below) to the sponsor of the test, TRAO, and Reactor and Fuels Research and Development Operation (RAFRAD).* The following steps will be used in planning and approving tests:

1. design of test and request by sponsoring component,
2. review by TRAO of request for feasibility and conformance with existing Process Specifications, and
3. preparation, if necessary, of new or revised Process Specifications by Reactor Engineering Development Operation.

Process Specifications will be prepared and recommended by Reactor Engineering Development Operation (a subsection of RAFRAD), accepted by the TRAO Manager, and approved by the RAFRAD Manager. An independent safeguards analysis will be performed for tests that cannot be conducted within approved Process Specifications. If the results of the analysis show that the test can be conducted without compromising reactor safety, a supplement to the Final Safeguards Analysis will be issued (if necessary) and new or revised Process Specifications will then be issued and approved. In every case Process Specifications will comply with the provisions of the Final Safeguards Analysis and supplements thereto. All supplements to the Final Safeguards Analysis will be transmitted to review bodies for approval.

Responsibilities of Sponsor of Tests

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

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* Reactor and Fuels Research and Development Operation is a section of Hanford Laboratories. Reactor safeguards studies are conducted within the Reactor Engineering Development (REDO) subsection, which includes specialists in the functions of heat transfer, fluid flow, design development, reactor process engineering, reactor physics, reactor instrumentation, shielding, and mechanical equipment development. Specialists in fuels, materials, water quality, and corrosion technology are included in other subsections of Reactor and Fuels Research and Development Operation.

produced as well as the technical feasibility of conducting the test. In addition to the foregoing, the sponsor will be responsible for:

1. furnishing information on the effects of the test on reactor physics, safety, and operation,
2. preparing a specific test procedure, and
3. specifying the form and frequency of experimental data to be obtained in conjunction with the test.

Responsibilities of Test Reactor and Auxiliaries Operation (TRAO)

Test Reactor and Auxiliaries Operation will be responsible for the safe conduct of the test in such a manner that desired data are obtained. TRAO 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, TRAO will:

1. conduct analyses of the proposed test to determine its feasibility,
2. determine whether the test can be conducted within existing, approved Process Specifications,
3. review the effect of the test on reactor safety,
4. prepare the operating procedures and train personnel, as necessary, for the test, and
5. provide consulting assistance to the sponsor.

Responsibilities of Reactor Engineering Development Operation

Reactor Engineering Development Operation is responsible for conducting reactor safeguards analyses, publishing safeguards reports, writing the Process Specifications for the Critical Facility, and auditing reactor operation for compliance with Process Specifications. REDO responsibilities in connection with the performance of tests will be:

1. continuing, independent review and audit of performance of tests for compliance with existing approved Process Specifications and
2. initiation of action for tests which cannot be conducted within approved Process Specifications. If a proposed test cannot be conducted within approved Process Specifications or involves

conditions not covered in the Final Safeguards Analysis, REDO will:

a. conduct an independent safeguards analysis of the test and recommend changes to permit performance of the test without compromising reactor safety,

b. publish, if necessary, a supplement to the Final Safeguards Analysis presenting the results of the foregoing analysis, and

c. revise existing, or prepare new Process Specifications for the proposed test provided the specifications do not violate the provisions of the Final Safeguards Analysis and supplements thereto.

Performance of Tests

Responsibility for Operation of Critical Facility

The manager of TRAO is responsible for and held accountable for the safety and operation of the Critical Facility and its associated experimental tests.

No one outside this organization is permitted to perform any manipulations in or around the reactor without direct authorization from the operating supervisor on duty at the time. Authorization will be given only to formally qualified individuals.

Sponsors' Representatives

A reasonable number of technical representatives of sponsoring components will be granted written approval by Manager, TRAO, for the purpose of observing and facilitating the conduct of their experiments. Certain test sponsors will be designated as qualified experimenters. Qualification will allow these individuals to observe the performance of their tests and to provide advice and instruction during their tests.

Process Specifications - Operating Limits

Process Specifications will be provided for the Critical Facility where safety of the facility is concerned. The specifications will consist of limits to critical process conditions, safety circuit trip settings, annunciator alarm settings, procedural restrictions on critical operations such as fuel element transfer and loading, and emergency procedures affecting reactor safety. Safety circuit trip settings and major operating limits are summarized in Tables II and III.

TABLE II

SAFETY CIRCUIT TRIP POINTS

Trip Function	Setting
1. High Flux Level	Settings equivalent to 150 watt level, maximum
(a) Start-up Channel	
(b) Intermediate Channel	
(c) High Level Channel	
2. Reactor Period	10 seconds, minimum
(a) Start-up Channel	
(b) Intermediate Channel	
3. Seismoscope (PRTR)	
(a) High Sensitivity	II (Intensity of Modified Mercalli Scale of 1931)
(b) Low Sensitivity	V (Intensity of Modified Mercalli Scale of 1931)
4. Exhaust Air Activity (Automatic Confinement Trip)	5×10^{-2} $\mu c/cc$, maximum
5. Aqueous Effluent (Automatic Confinement Trip)	5 $\mu c/cc$, maximum

TABLE III

OPERATING LIMITS

1. Safety System	
(a) Insertion Time (from signal initiation to time full reactivity worth is inserted)	1 second, maximum
(b) At-The-Ready Worth	
During Operation	70 mk, minimum
During Core Alterations	50 mk, minimum
2. Reactivity Addition Rate	
(a) Control Rods	10 cents/second, maximum
(b) Safety Rods	10 cents/second, maximum
(c) Moderator Addition	0.05 mk/sec/mk subcritical, maximum, when reactor is more than 10 mk subcritical. 10 cents/second, maximum when reactor is less than 10 mk subcritical.
3. Nuclear Instrumentation	
(a) Neutron flux level channels in service with safety circuit high flux trips	3, minimum
(b) Reactor period channels in service with safety circuit trips	1, minimum
4. Critical Moderator Level	5 feet, minimum

VI. SAFETY ANALYSIS

General Safety Features

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.

Doppler Coefficient

The Doppler coefficient is primarily a result of changing the U^{238} resonance capture contribution, and consequently, enriched fuel elements (Pu-Al) are assumed to exhibit a negligible temperature effect compared with that of the UO_2 fuel elements. The Doppler coefficient for the reactor as a whole will, therefore, depend on the particular reactor loading. Because of the relatively low thermal conductivity of the UO_2 fuel elements, almost all of the energy released in this type of fuel element in a nuclear excursion would be initially available for increasing the fuel temperature.

Recent experiments indicate a temperature dependence for the U^{238} resonance integral such that:

$$\Sigma_r (T) = \Sigma_{r_o} \left[1 + \alpha \left(\sqrt{T} - \sqrt{T_o}\right)\right]$$

This expression yields:

$$\frac{\Delta k}{k} = - \alpha \left(\sqrt{T} - \sqrt{T_o}\right) \ln p_o$$

where p_o = initial resonance escape probability
T = average UO_2 fuel temperature, °K
T_o = initial temperature, °K
α = constant.

For PRTR Mark I UO_2 fuel, 19-rod cluster elements, $\alpha = 0.74 \times 10^{-2}$ and $p_o = 0.868$. An estimate of the effect in the reactor when both UO_2 and Pu-Al elements are present can be obtained by adjusting p_o for the fractions of the loading consisting of UO_2 and enriched elements.

For a two-zone, unreflected reactor loading (Pu-Al and UO_2 fuel elements) designed to be critical at full moderator level, the negative reactivity introduced by the Doppler coefficient is estimated to be -8.5 mk when the temperature of the maximum power fuel element reaches the melting point of the jackets (3314 F).

Moderator Void Coefficient

Another inherent shut-down mechanism is the void generated in the liquid moderator due to vaporization and radiolytic gas formation. The heavy-water-moderated Critical Facility has a lower moderator void coefficient than the void coefficient of a light-water-moderated reactor. The negative reactivity introduced by vaporization is estimated to be -4.5 mk at fuel element temperatures of 3314 F using methods similar to those developed in analyzing Borax I and Spert I experiments.[3] The above estimate is for a large, two-zone reactor loading designed to be critical with D_2O at full moderator level.

Moderator Temperature Coefficient

The bulk of the moderator temperature coefficient ($\sim 1.7 \times 10^{-4}$/F) is the result of thermal expansion of the moderator. The reactor will be operated with the moderator level regulated by the weir position. The increase in moderator volume from thermal expansion would not result in a moderator level increase, as the excess would flow over the weir to the storage tank. If the reactor were erroneously made critical with the moderator level below the top of the weir, thermal expansion would increase the thickness of the top reflector. However, this means only that the vertical leakage would be more or less unaffected, since the migration area is inversely proportional to the square of the moderator density, and the vertical geometric buckling is essentially proportional to the square of the density. The multiplication constant and the 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.

About 5 percent of the heat produced in the reactor will be generated directly in the moderator through neutron slowing down, scattering, gamma absorption, etc. This heat would appear promptly in the moderator during an excursion. However, the total heat capacity of the moderator is such that the moderator temperature increase would only be about 4.5 F by the time fuel jackets reached the melting point in a fast excursion which allowed little or no transfer of heat from fuel elements to moderator. The corresponding reactivity loss would amount to about 0.75 mk.

In a more gradual "excursion" which would allow transfer of more substantial quantities of heat from fuel to moderator before fuel temperatures became excessive, the negative reactivity introduced by the moderator coefficient would be considerable, on the order of 20 to 25 mk by the time that the average moderator temperature reached the boiling point. Bulk boiling in the moderator would decrease the density further, producing an additional reactivity decrease.

It is concluded that for reactor loadings containing UO_2 fuel elements, the Doppler negative temperature coefficient of the fuel is the chief inherent safety mechanism for fast excursions. For a gradual "excursion", the moderator temperature coefficient could also have a significant effect.

Control Components

Reactivity can be added to the Critical Facility by three methods; fuel addition, moderator addition, and control and safety rod withdrawal. The following section is concerned with operating console control mechanisms. Fuel element loading procedures were covered in Section V (Operating Procedures: Loading Procedures).* Insofar as possible, the capability of reactivity addition from the operating console will be limited to one dollar and operating procedures will require that the final increment of fuel added will result in available excess reactivity no greater than one dollar with the control rods fully withdrawn and the moderator at the maximum attainable position for a particular experiment.

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Control Rods

The primary control system consists of three poison type control rods having an estimated maximum worth of 10 mk each. Because of the low operating power of the reactor, control allowance for temperature coefficients, fuel burnout, and poison buildup will not be required. The control rods will be used to compensate for the reactivity effects of various samples being tested in the reactor and to adjust the reactor to critical at specific moderator levels. The calculated reactivity addition for withdrawal of a 10 mk control rod is 0.15 mk per second. The maximum worth and the travel speed will be limited to prevent reactivity addition rates greater than ten cents per second. The rod control system is designed such that only one control or safety rod can be moved at a time.

The inner cylinder will be held to the drive by an electromagnet. Tripping of the safety circuit will interrupt the current to the magnet allowing the rod to drop to its maximum poisoning position. Operation of the rods will be manual through the use of controls at the control console. The rate of reactivity input for withdrawing a control rod should be nearly linear; in effect short sections of rod are pulled throughout the vertical height of the reactor. The calculated reactivity introduction curve for complete withdrawal of one control rod over the 6-inch travel distance is shown in Figure 18. Analysis of the nuclear excursion resulting from adding reactivity at a rate of 0.15 mk per second is presented in Section VI (Nuclear Excursions: Control System Malfunctions - Operating Errors).*

Moderator Level Addition

The moderator will be maintained at a level between 5 and 9 feet above the bottom of the reactor tank during operation. Core configurations which could result in achieving criticality at moderator levels less than 5 feet will not be permitted in the Critical Facility. Moderator level is controlled by a weir adjustable over the range from 5 to 9 feet and is maintained at the desired level with a 1 gpm recirculation pump. Moderator level adjustment will not be used as a primary control method, but will be provided mainly to allow experiments with various effective heights.

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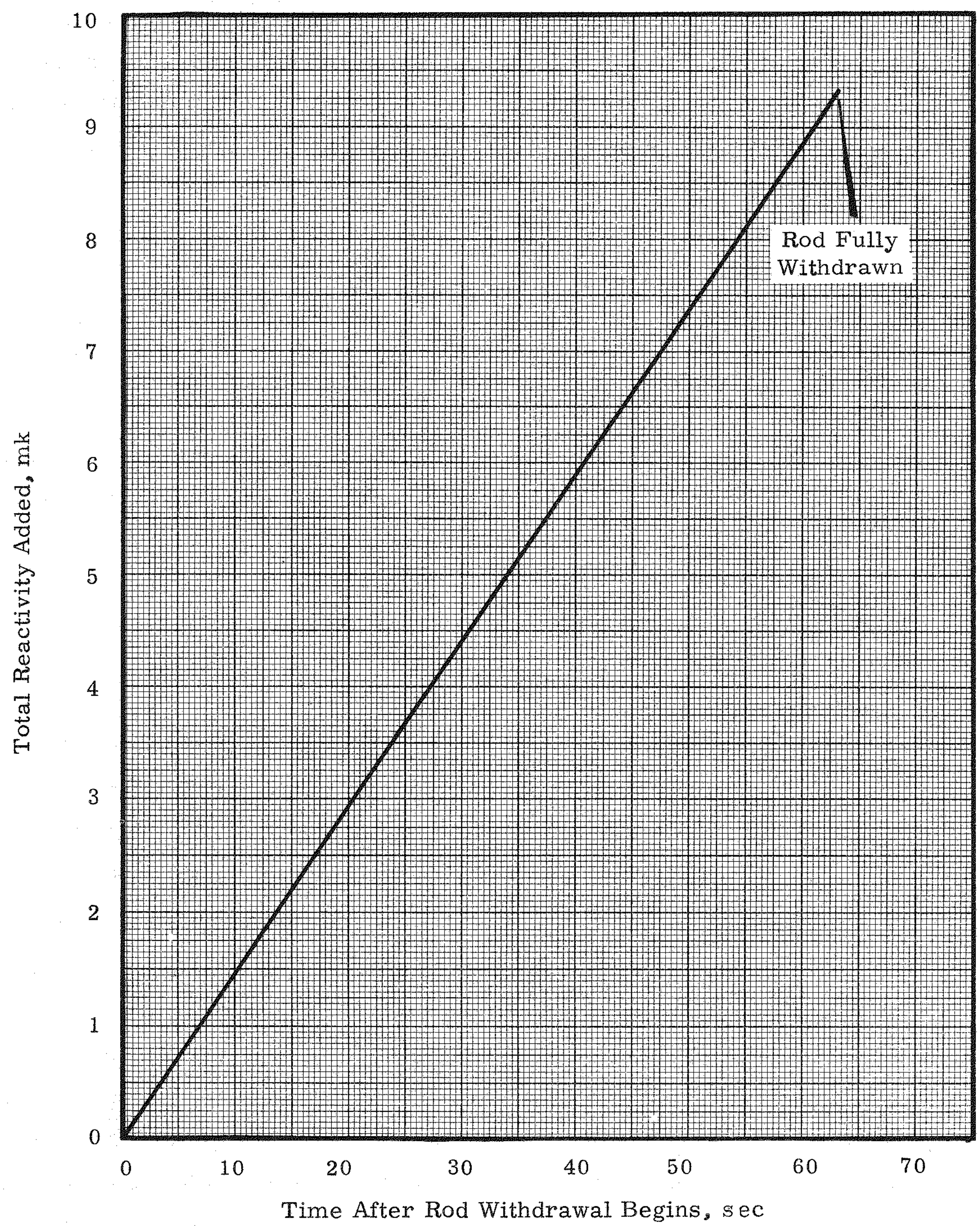

FIGURE 18

Reactivity Addition for Withdrawal of One Control Rod

The moderator will be completely drained from the reactor tank during the loading of fuel element assemblies. Electrical interlocks are provided to prevent positioning the cell crane over the reactor whenever there is any moderator in the reactor tank. In addition, the cell crane is mechanically interlocked with a drain valve on the bottom of the reactor tank. Interlock action is such that the valve will dump the moderator to the cell if the crane is erroneously moved over the reactor when there is moderator in the tank.

When the reactor core is drained, the moderator is stored in two tanks, the cell storage tank beneath the reactor, containing 850 gal., and the auxiliary storage tank, located outside of the cell, with a capacity of 1060 gal. After loading an increment of fuel, the moderator will be raised in two steps. First, the moderator in the auxiliary storage tank will be pumped into the core, stopping the addition of moderator at a level where the rate of reactivity addition is 0.05 mk/sec/mk subcritical. (The cell piping is arranged so that the cell storage tank must be filled before D_2O pumped from the auxiliary storage tank can enter the core tank.) During this first step moderator will be added at a maximum rate of 200 gpm. The second step will be to raise the moderator level to the desired operating level by pumping moderator from the cell storage tank at a rate of approximately 35 gpm, resulting in a calculated maximum subcritical reactivity addition rate of 0.25 mk/sec. The maximum reactivity addition at critical (5-ft critical moderator level) is calculated to be 0.18 mk/sec.

Only one of the moderator addition pumps can be used at a time to add moderator to the reactor tank. Interlocks are provided to prevent simultaneous reactivity addition by control rod withdrawal and raising moderator with the moderator addition pumps. Interlocks are provided to prevent moderator addition with the moderator addition pumps after control rod withdrawal has begun. This will prevent reactivity addition by raising moderator with these pumps while the reactor is operating, since the pump motors are de-energized after control rod withdrawal begins. In addition, interlocks will be provided to prevent moderator addition unless the safety circuit is made up and the safety rods are completely withdrawn from the reactor core.

A mechanical stop, adjustable only from the reactor cell, is provided to prevent raising the weir (by manipulations at the control console) above the desired position for a particular experiment. Insofar as possible, specified procedures will require that the stop be positioned such that the calculated excess reactivity addition from raising the moderator level to the position of the stop and then withdrawing control rods to the full-out position will be no greater than one dollar.

Rod Safety System

Three safety rods have been provided. Each rod is a hollow aluminum cylinder with a 20-mil cadmium coating. These rods are calculated to be worth approximately 35 mk each when located at the inner boundary of the driver ring, with the moderator at full height. The drive for these rods is similar to that for the control rods. The calculated reactivity addition upon withdrawal of a 35 mk safety rod is 0.15 mk per second. For a reactor loading which would allow operation with a moderator level of 5 feet, the minimum planned operating level, the rods are calculated to be worth about 40 mk each. This fact, combined with the fact that a rod movement of 5 feet would result in the addition of the full worth of the rod, could produce reactivity addition rates up to 0.275 mk per second. The rod worth and travel speed will be limited to prevent reactivity addition rates greater than ten cents per second.

Analyses of the nuclear excursions resulting from withdrawal of safety rods at both full and 5-foot moderator levels are presented in Section VI (Nuclear Excursions: Control System Malfunctions - Operating Errors).*

The safety rods will be held to their drives by low retentivity electromagnets. Tripping of the safety circuit will interrupt the current to all three magnets allowing the rods to drop into their maximum poisoning positions.

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The rods will be enclosed in air-filled thimbles to minimize frictional resistance while falling. The thimbles will be pressurized to 5 psig to prevent the air space from filling with moderator in the event the thimble should leak.

The rods are designed to allow the full reactivity worth to be inserted in the reactor within one second after a safety circuit trip occurs. Prior to installation, all of these rods will be tested to insure that the reactivity insertion time criterion has been met. Periodic release and drop time tests will also be conducted after installation. Calculated scram curves for full and 5-foot moderator level are shown in Figure 19.

The control and safety rod thimbles will be supported by, and bolted to, the top reactor grid plate. The thimbles are designed to fit into any opening in the grid plate to allow the rods to be used in optimum positions for various loadings. Process specifications will limit the movement and location of the rods to insure that they remain in effective positions.

With the exception of period trips, all trip circuits are of fail-safe design. All safety circuit trips except removal of a cell access plug will interrupt the current to all three safety rod solenoids, as well as the three control rod solenoids. The multiplicity of detectors and trip circuits, combined with the fact that any one of the three safety rods will have sufficient strength to override any credible reactivity input, provides a highly reliable safety system.

Nuclear Excursions

Procedural Errors

The design and operating characteristics of critical facilities are such that procedural errors present possibilities for nuclear excursions. A number of features were incorporated to minimize the hazards of human error. These are outlined:

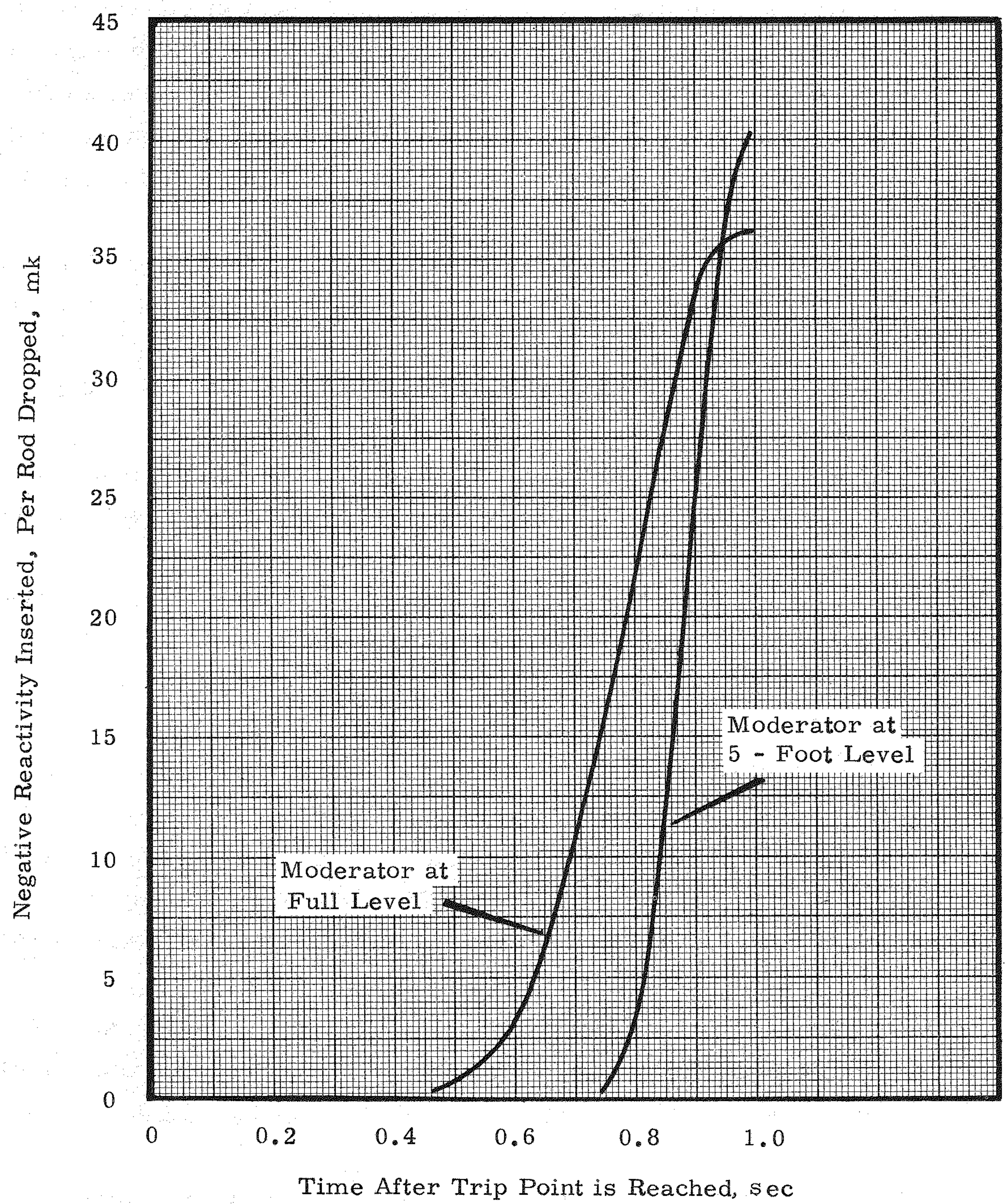

FIGURE 19

Scram Curves for Reactor with Moderator at Full and 5-Foot Levels
(Curves are Values Per Safety Rod Dropped)
(A 0. 25 Second Delay for Circuit Actuation and Rod Release is Included)

1. Interlocks will be provided to prevent movement of control rods in the direction of increasing reactivity unless at least one start-up or intermediate flux monitoring channel is reading at a level sufficient to insure that the channel is operable. One start-up channel and the two intermediate (log) channels will actuate both period and flux level trips.
2. A rod control selector switch will make it impossible to apply power to more than one rod positioner at a time. A spring loaded toggle switch will be provided for each of the control and safety rods, requiring the operator to have his hand on the switch in order to move a rod in the direction of increasing reactivity.
3. All of the moderator addition pumps are interlocked to prevent their operation unless the safety circuit is made up and all safety rods are cocked.
4. An audible monitor has been provided in conjunction with one of the start-up flux channels. This monitor will have speakers in both the operating area and the reactor cell to warn of increasing sub-critical flux levels.
5. Interlocks with a part of the safety circuit will cause two control rods and one safety rod to drop into their maximum poisoning positions when the reactor cell cover block for any access opening is removed. This will prevent entrance into the reactor cell when the reactor is operating and will act as a safeguard against unexpected achievement of criticality while loading fuel elements.
6. Circuitry for the two log-N (intermediate) channels and the two linear high level channels is such that a high flux level trip on any of the four will trip the rod safety circuit. Only one of the four channels can be bypassed at any one time.
7. Interlocks are provided to insure that fuel element assemblies are loaded and unloaded with the moderator drained from the reactor tank.

Fuel Loading Accident

The loading and unloading of fuel element assemblies (e.g., cluster type assemblies composed of 19 rods) in the Critical Facility will be permitted only under the following conditions:

1. The moderator is completely drained from the reactor tank.
2. Two safety rods are cocked.
3. One safety rod and two control rods are inserted in the reactor.
4. Incremental loading procedures described in Section V (Operating Procedures: Loading Procedures) are followed.*

Several design features insure that fuel loading will take place with no moderator in the reactor; these are, first, electrical interlocks on the crane position and the moderator level and, second, a mechanical interlock on the reactor tank drain valve and the crane.

Since there is no moderator in the reactor tank when fuel is being loaded, a fuel element dropped into the core could not add sufficient reactivity to make the reactor critical. Therefore, a fuel element drop-in accident would not lead to a reactor excursion.

Control System Malfunctions - Operating Errors

Control errors, either continued withdrawal of control or safety rods or continued addition of moderator with the reactor supercritical, could result in a nuclear excursion. Consequences of various control errors were analyzed using an analog computer. Kinetic equations and parameters used in the analysis are presented in the appendix. The excursions were terminated by a safety circuit scram with one safety rod at a power level of 150 watts. A delay time of 0.25 seconds for relay actuation and rod release was used.

Cases were analyzed for withdrawal of a control or safety rod with the reactor critical at full moderator level (calculated reactivity addition rate, 0.15 mk per second) and for withdrawal of a safety rod with the reactor critical at a 5-foot moderator level (calculated reactivity addition rate, 0.275 mk per second). Calculated powers, energy releases, and maximum

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fuel temperature for these cases terminated by a single safety rod scram are shown in Figure 20. The analog studies showed no perceptible temperature increase.

Studies of the excursion from continued addition of moderator were also conducted (calculated initial reactivity addition rate, 0.135 mk/sec). For this case it was assumed that the reactor was critical at a moderator level of five feet, giving the maximum planned incremental reactivity worth for a typical two-zone loading. Results of this study, presented in Figure 21, show no perceptible fuel temperature increase. Tripping of the safety circuit would stop operation of the moderator addition pumps, and thus halt the addition of reactivity.

It is concluded that reactor control errors present no hazard of fuel melting, the analog calculations showing no perceptible temperature increase.

Control Rod Blowout

The control and safety rods will be inserted from the top of the reactor and when fully inserted will be in their maximum poisoning positions. Accidential dropping of a rod would shut down the reactor rather than cause a nuclear excursion and, thus, presents no hazard.

The only other conceivable mechanism for rapid removal of rods from the reactor is the application of high pressures in the tank and blowing the rods and thimbles out the top. In the Critical Facility the rods will be enclosed in thimbles, each of which will be bolted to the grid plate. The pressures required to break these bolts would be sufficient to rupture the reactor tank, thus draining the moderator from the tank and making the reactor subcritical by several hundred mk. Two vent paths from the reactor tank will be provided to prevent pressures which would damage the tank. These are a 1-inch line from the tank to the vent header and a 1-inch line which will connect the tank with the reactor cell when a relief valve opens.

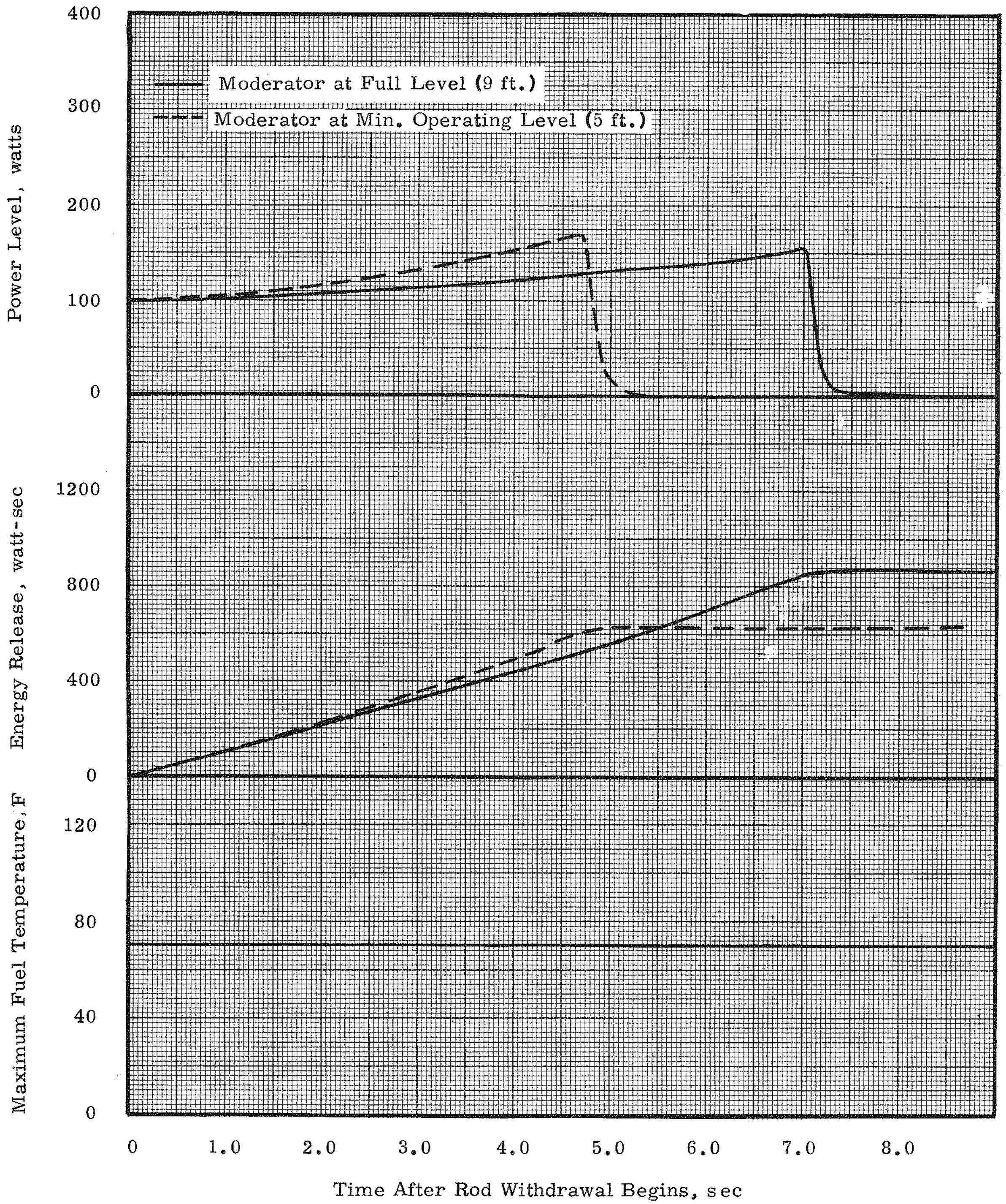

FIGURE 20

Nuclear Excursion Following Withdrawal of a Safety or Control Rod with Moderator at Full Level and Withdrawal of a Safety Rod with Moderator at the Five-Foot Level; Safety Circuit Scram with One Safety Rod

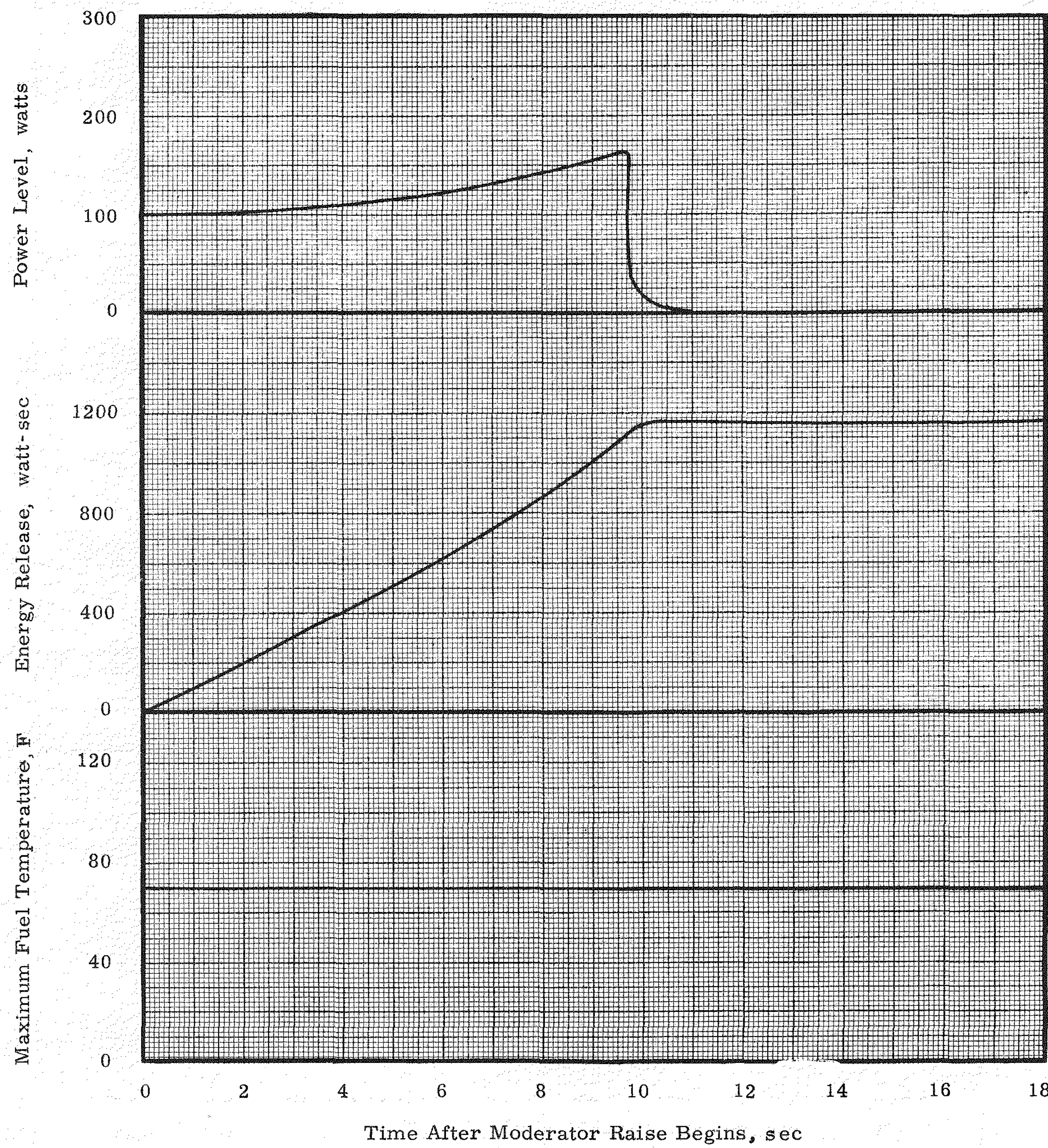

FIGURE 21

Nuclear Excursion from Continued Addition of Moderator; Safety Circuit Scram with One Safety Rod

Further, as the fuel element hangers are not fastened to the grid plates, an internal pressure in excess of 8 psi would raise the fuel elements. This would unseal the holes in the grid plates, providing further relief. In a violent pressure surge, fuel elements would be blown from the reactor more easily than would the control and safety rods. Such fuel blowout would decrease reactivity and shut down the reactor.

The reactor will operate at atmospheric pressure. Moderator and coolant temperatures will normally be 70 to 90 F. The only possible mechanism for sudden applications of large pressures to the reactor tank would be a violent nuclear excursion. The excursion studies discussed previously show that the safety system would terminate any credible nuclear excursion before the energy release was sufficient to cause any pressure increase from heating or boiling of the moderator.

Coolant Thimble Rupture

Rupturing of a fuel element coolant thimble while in the reactor should not result in any reactivity increase. Water entering the reactor tank from the coolant system would drain over the weir and the moderator level would remain constant. However, if the reactor were being operated erroneously with the moderator below weir level, such an accident could result in a reactivity increase from adding the coolant to the moderator. The 35 gpm cooling system pump will raise the moderator level at a rate comparable to that obtained with the 35 gpm moderator storage tank addition pump. The cooling system will contain only 20 to 30 gallons of heavy water, which would increase reactivity approximately 8 mk (at a moderator level of 5 feet). After this supply has been exhausted, light water from the emergency cooling system would be added. The effect of the light water addition would be a decrease in reactivity. It is calculated that a 1 percent degradation of the moderator would result in a reactivity loss of about 12 mk, assuming uniform mixing. With the light water concentrated in a localized region, as would be the case with the light water entering the reactor from a ruptured thimble, the negative effect would be decreased, but would still be sufficient to compensate for the positive contribution from the moderator level increase (maximum: ~2.5 mk for a 1 percent level increase).

The excursion resulting from an accident of this type would be no more severe than that resulting from raising moderator at the maximum possible rate, shown in Figure 21.

Fuel Element Failure

Violent failure of enriched fuel elements, resulting in a rapid distribution of plutonium throughout the moderator, and occurring in several elements simultaneously, could initiate a serious power transient. However, under the operating conditions planned for the Critical Facility, any fuel element failure would proceed very slowly. With the fuel element temperatures being maintained at 70 to 90 F, neither the zirconium jackets nor the aluminum cores will corrode rapidly.

The Pu-Al elements which will be used initially as driver elements are made up of 19-rod clusters. Simultaneous failure of many feet of fuel element, involving a number of fuel rods, and rapid dispersion of the enrichment throughout the moderator would be required to cause an excursion. Consequently, it is considered incredible that failure of enriched elements in the reactor would cause an excursion.

Flooding of Reactor Cell

Initially, sheets of cadmium will be wrapped around the tank. For other tests, the cadmium wrap will be replaced by an appropriate thickness of reflector material. In either case, flooding of the reactor cell will not have any net positive effect on reactivity.

Loss of Coolant

The principal radiological hazard in the facility would be the possibility of melting an irradiated fuel element following loss of coolant.

Loss of the moderator, which serves as the coolant for all fuel elements except those which have been irradiated in the PRTR, would present no risk of fuel melting. Such a loss would shut down the reactor. The fission product decay heat generation rate following a shutdown from the design power level of 100 watts would be less than one watt per fuel element. Air in the reactor tank would be a more than adequate heat transfer medium.

Loss of coolant during transfer or testing of "long-cooled" irradiated elements presents no threat of fuel melting. "Long-cooled" elements will not be tested in the Critical Facility until their heat generation rate from fission product decay has declined such that the maximum temperature would not exceed 572 F with the element suspended in stagnant air for an infinite time. Thus, following loss of coolant, temperatures would not approach melting for either the UO_2 or Pu-Al elements.

In order to follow certain short-term reactivity transients (e.g., xenon transient in a fuel element following reactor shutdown) "short cooled" fuel elements will be transferred in thimbles from the Plutonium Recycle Test Reactor to the Critical Facility as quickly as possible after a PRTR shutdown. The maximum power generation rate of a "short cooled" fuel element will be 12 kw. Loss of coolant to one of these fuel elements for an extended time could result in melting of the jacket of the fuel element if the fuel element remained in the thimble. The geometry of the fuel elements and the thimble is such that heating of the inner fuel element rods would be essentially adiabatic, at least until very high temperatures were reached. Assuming no heat transfer from these rods, with the element generating heat at a rate of 12 kw, melting of Pu-Al cores would begin about 14 minutes after loss of coolant. Core melting would occur over a period of about 7 minutes at the maximum heat generation point in the fuel rod. Thus, at the maximum heat generation point, core melting out to the zirconium jacket would not be complete until about 21 minutes after coolant loss, even with adiabatic heating. The zirconium jackets would reach their melting point in about 48 minutes. For a UO_2 fuel element, jacket melting temperatures would be reached after 44 minutes. The heating of the fuel element for this case is shown in Figure 22.

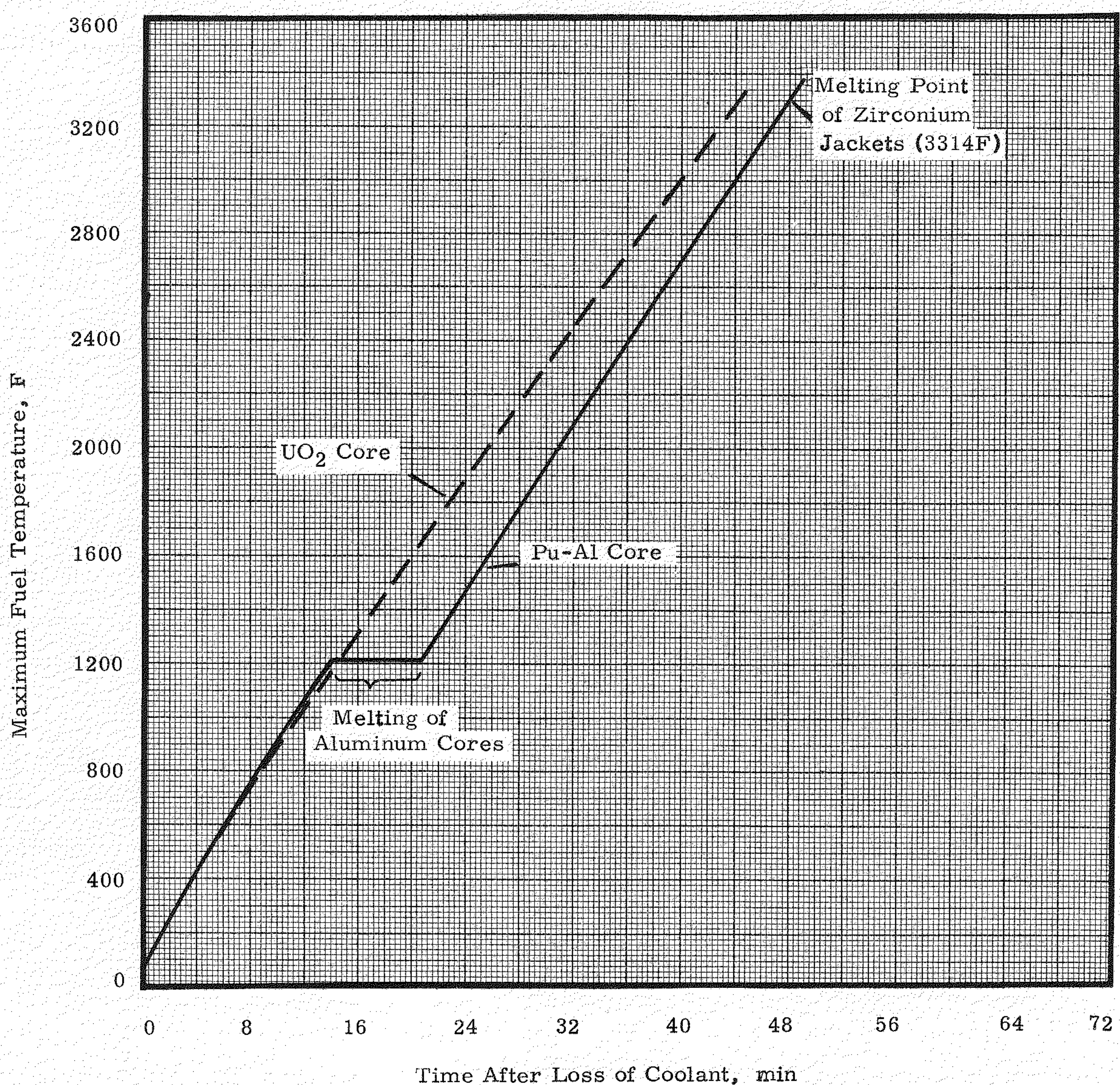

FIGURE 22

Temperature of 19-Rod Fuel Element After Coolant Loss
(Assuming 12 kw Fuel Element; Adiabatic Heating of Inner Rods)

Calculations indicate that no fuel element jacket melting will occur if the "short cooled" fuel element is removed from the thimble after a loss of coolant accident and left hanging in air. Temperature calculations for a cluster type fuel element were based on free air convection only, using the following equation developed by Jakob and Linke:[4]

$$hD/_k = 0.13 \, (N_{Gr} \cdot N_{Pr})^{1/3}$$

It was assumed that the cell air temperature remains constant at 75 F and that air exchange in and around the cluster type fuel elements is good. Conduction across air gaps and radiation were ignored. For this case the effect of these methods of heat transfer would be small as compared with that of free convection for the central rod and, to a lesser extent, for the 6-rod ring. For the 12-rod ring, radiation and free convection combined would probably limit the temperatures well below the melting point of aluminum. It was also assumed that the power generation rate remained constant rather than decreasing during the time of heating and that the fuel element had operated at a tube power of 1200 kw in the PRTR.

As shown on Figure 23, the maximum Pu-Al core temperature attained for a "short cooled" fuel element hanging in air is 1655 F. No fuel element jacket melting would occur (jacket melting point 3315 F). Some melting of the central rod core of a Pu-Al fuel element might occur, depending on the tube power at which it had been irradiated in PRTR. Melting of cores in the 6-rod ring is improbable. No core melting in the outer, 12-rod ring would occur.

The fuel element could not be reused since jacket temperatures above 572 F will not be tolerated for fuel elements to be recharged in the PRTR.

An emergency coolant system is provided to cool the fuel element if the normal cooling system should fail. This system would provide cooling in the event of a system rupture or obstruction, or equipment failure. Emergency coolant lines are connected to both the inlet and outlet thimble piping. These lines extend through the reactor cell cover to the control area

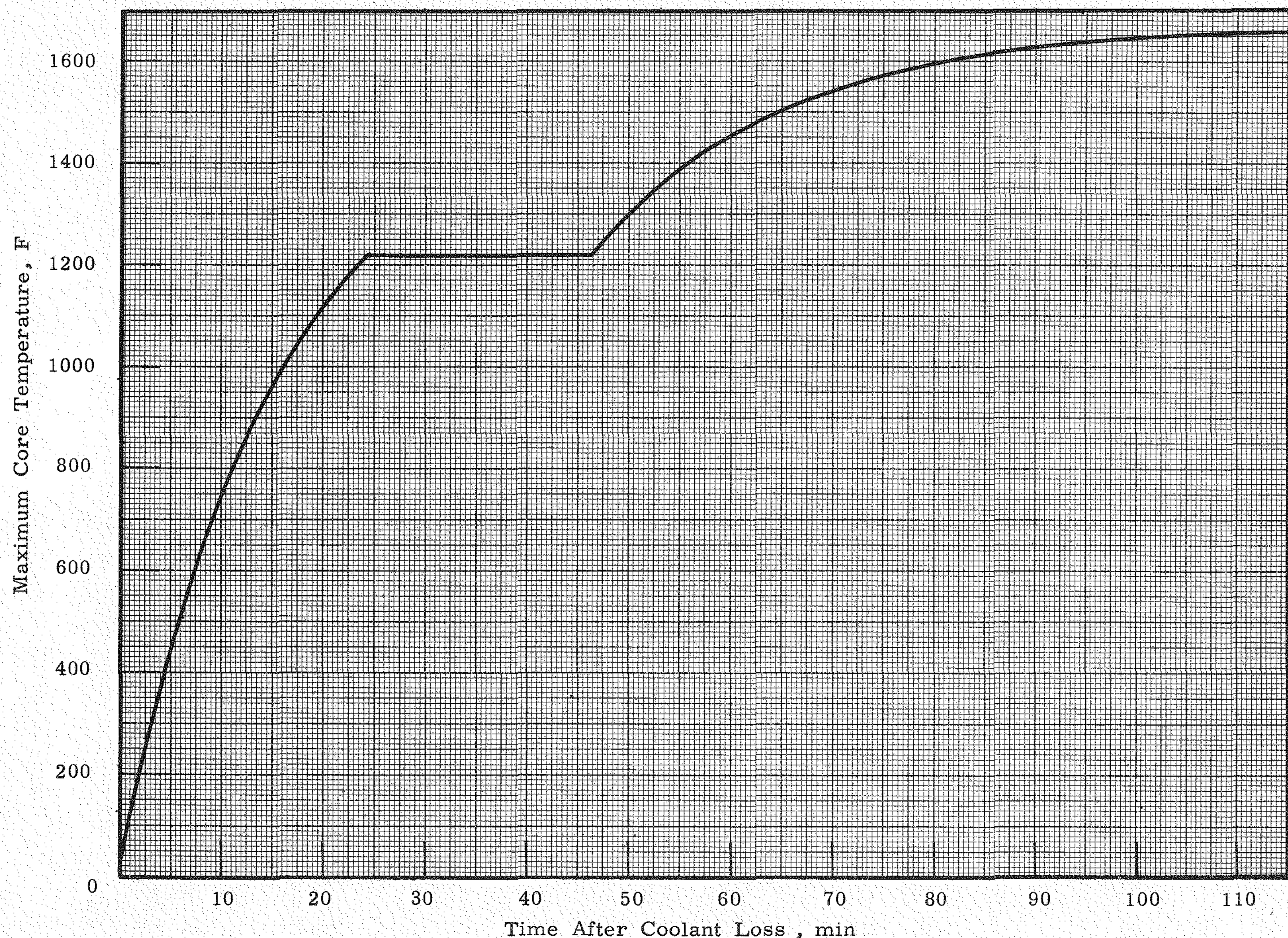

FIGURE 23

Temperature of 19-Rod Pu-Al Fuel Element After Coolant Loss
(Free Convection Cooling in Air)

where they join a common supply line. A flexible hose with a quick-connecting coupling will allow the supply line to be connected to either the PRTR process water or sanitary water systems.

The system is designed to provide flow to both the inlet and outlet sides of the fuel element thimble in the event of a system rupture or in the case of a flow obstruction or pump failure to provide flow through the thimble to a line which drains to the loadout canal. If an abnormal coolant system flow signal were received, the flexible line would be connected to the process water line and the system valved to provide flow to both the inlet and outlet sides of the thimble. If the abnormal flow had resulted from a flow obstruction or pump failure, bursting of a rupture disc would allow water to flow through the thimble to the loadout canal.

The connecting hose and the manually operated valves will be located in the control area where they will be immediately available to the operators. However, some time will be required for the technicians to make decisions and perform the necessary manual operations. Studies of various loss of coolant accidents were made to determine the time in which emergency coolant must be provided to prevent melting. These accidents are discussed below.

Coolant Line Rupture

Normal System Flow

Rupturing of either the inlet or outlet lines to the fuel element thimble or the thimble itself could result in a loss of coolant from the cooling system. If the system had been operating with normal coolant circulation, a low flow alarm would be actuated by the flow monitor located near the discharge of the coolant circulation pump. If the rupture had occurred just downstream of the flow monitor, the low flow alarm would not be activated until the system was almost completely drained. This would occur less than one minute after the rupture. For ruptures in other parts of the system the alarm would occur sooner. The low level alarm in the system surge tank would also be actuated within a few seconds after the rupture occurred.

If the rupture had occurred in the thimble itself, the thimble could drain within a few seconds leaving the fuel element uncovered, and the temperature of the element would begin to rise. Plutonium-aluminum melting would not be complete until about 21 minutes after coolant loss. If the rupture had occurred in any part of the system other than the thimble, the pump would probably vapor lock with the fuel element thimble still full of water. The fuel element would be cooled for some time by boiling the water in the thimble. With no credit taken for cooling by steam sweeping up past the fuel element, and assuming that the break had occurred in a location allowing the coolant to drain down to the junction of the cooling line and the thimble, Pu-Al cores would not reach the melting point of aluminum for more than an hour after the rupture had occurred.

As soon as the flow alarm had been actuated, and the abnormal flow conditions confirmed by observing the flow meter, action would be taken to establish emergency coolant flow. The minimum of 21 minutes before any core melting could be completed under the most unfavorable circumstances should be more than adequate to establish emergency coolant flow.

Coolant System Failure During Coolant Transfer

When an irradiated fuel element is first placed in the coolant thimble, the thimble will be full of light water from the PRTR loadout canal. An air blower will then be used to force this light water from the system, and dry the system before it is refilled with heavy water. During the time that air is being blown through the system, a flow gage measuring the flow rate will be monitored continuously. This gage will be located at the valve control panel in the control area and will measure the flow which has passed over the fuel element. If abnormal flow rates were observed, actions to provide emergency coolant flow would be taken.

The coolant transfer operation will be conducted with the fuel-containing thimble submerged under more than 20 feet of water in the loadout canal. The fuel element will be sealed in the thimble by a quick-disconnect coupling. Disconnecting of this coupling would allow water from the canal to run into the thimble and provide adequate cooling of the fuel element. This action would require hooking on to the coupling release under 20 feet of water. Actions specified in emergency procedures for loss of coolant will require immediate turning on of emergency coolant, then unsealing the thimble and removing the fuel element.

During the coolant transfer operation the fuel element temperature will increase, particularly during the early part of the transfer when water in the lines limits the gas flow rate. This temperature increase would affect the allowable time to restore cooling after a system failure. However, fuel jacket temperatures above 572 F will not be tolerated for elements to be recharged in the PRTR. The system will be adequate to limit jacket temperatures below this value. Assuming a temperature of 572 F at the time of coolant loss, Pu-Al cores could reach their melting point at the maximum power point in about eight minutes. Complete core melting at the maximum power point would not result for 15 minutes. The Zircaloy jackets would not reach their melting point for about 40 minutes. These times would be sufficient to prevent any fission product release before water cooling was restored.

Before any short-cooled elements are transferred, tests will be conducted with unirradiated elements in the thimble to determine the flow transients during the transfer period.

Multiple System Ruptures

Simultaneous rupture or plugging of both the inlet and outlet cooling lines would make it impossible to furnish either normal or emergency coolant to the fuel element thimble. In this case, UO_2 elements could be transferred to the loadout canal and removed from the thimble in the 44 minutes available before jacket melting temperatures were reached.

With a Pu-Al element, if the thimble remained filled with water, there would be adequate time to transfer it to the loadout canal and remove the fuel element. However, there is a possibility that, with both lines parted completely and the broken end of one line fallen below the bottom of the thimble, siphoning would drain most of the water from the thimble. Filling of the lock will require about 9.5 minutes. In this case, about 11.5 minutes would be available for transfer manipulations and unsealing the thimble before complete melting of a core cross section.

Electric Power Failure

The reactor will be shut down immediately upon loss of electrical power. The only possible risk from a power outage would be the melting of an irradiated fuel element. To provide protection against this eventuality, emergency power will be provided to the following:

1. coolant system recirculation pump,
2. coolant system air blower, and
3. reactor cell crane.

Normal electrical services will be supplied by the Bonneville Power Administration system. Upon loss of BPA power, emergency power from the PRTR diesel generator would be available in approximately 15 seconds. Normal coolant circulation would be restored.

The Bonneville Power Administration supply is highly reliable, showing five outages in four years at the 300 Area, the supply area for the Critical Facility. The duration of these outages ranged from momentary to eight minutes, with an average duration of three minutes. Failure of BPA power during transfer or testing of a short-cooled irradiated element, an operation which is planned only a few times per year, and coincident failure of the emergency power supply, is highly improbable. However, even in this case, emergency coolant could be supplied to the fuel element cooling system. The flow alarm would be actuated, and the emergency coolant system would be connected to the sanitary water system which is fed by 300 Area "hi-tanks". The system would be valved to provide emergency coolant flow through the thimble and into the loadout canal.

If the total power outage had occurred with the coolant system filled with water, at least an hour would be available to provide emergency cooling and prevent melting of Pu-Al cores. If the system were air-filled during coolant change-over, a minimum of 15 minutes would be available to prevent melting of Pu-Al cores, and at least 40 minutes would be available to prevent melting of Pu-Al or UO_2 fuel element jackets.

Failure of Blower or Recirculation Pump

The consequences of failure of either the blower or the coolant recirculation pump would be similar to those following a total power outage. However, in this case, the reactor cell crane would remain operable and after emergency coolant flow had been established, the fuel element could be transferred to the loadout canal and removed from the thimble.

Failure of Valves

With either normal coolant system flow or air flow during coolant change-over, failure of valves in the system could block coolant flow. If the system were blocked during normal water circulation, a low flow alarm would be received. Emergency coolant flow would then be provided by valving the emergency system to provide once-through flow. The thimble would remain full and over an hour would be available to restore flow.

During periods when the blower is providing air cooling, failure of either the valve through which air enters the system or the valve through which it leaves the system could block the air flow. In this case, emergency coolant flow would be established. At least 15 minutes would be available before Pu-Al cores would melt, and at least 40 minutes before jacket melting temperatures were reached.

Failure of Thimble Tube Seal

Accidental release of the latch which seals the fuel element in the thimble tube would, in effect, produce a coolant system leak. Such a leak would present no great risk, as the emergency cooling system could provide water flow through the fuel element thimble. Releasing the latch could allow the thimble to drop away from the fuel element. The components in the reactor cell are arranged such that in many locations the thimble would only fall about one foot. The active section of the fuel element would remain inserted in the water filled thimble. The thimble could fall completely free of the fuel element if the thimble was positioned directly over a reactor opening or was being raised beside the lock door in the cell. Calculations indicate that no jacket melting would occur if the fuel element was left hanging in air.

Releasing of the latch during coolant transfer operation would present no risk. The thimble would be submerged in the loadout canal and if the seal were broken, canal water would flood the thimble, providing adequate cooling.

Procedures

In general, actions specified in emergency procedures for loss of cooling will first require immediate use of the emergency cooling system. The fuel element will then be transferred to the water filled loadout canal. If it is impossible to supply emergency coolant and the thimble remains full of water, both the fuel element and thimble will be transferred to the canal, the quick disconnect thimble cover released, and the fuel element removed. If emergency coolant is unavailable and the thimble is drained by the accident, the fuel element will be removed from the thimble and then transferred to the canal.

Only if an accident of the type that would make it impossible to furnish emergency coolant were compounded by failure of the reactor cell crane in a position where the fuel element could not be removed from the thimble would any fission product release occur. Consequences of such a release are discussed in Section VII (Maximum Credible Accident).*

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* Page 88

Critical Mass Considerations

Fuel element storage facilities were designed to prevent the attainment of a critical array of fuel elements.

Unirradiated Fuel Storage

Racks along one of the walls in the reactor cell will be used to hold unirradiated fuel elements which are awaiting charging in the reactor. These racks will also be used to hold Critical Facility fuel elements during reactor loading changes. The storage racks will accommodate up to 46 fuel elements in a double row. The two rows will be six inches apart with elements in each row separated by 4-3/8 inches. The racks will be fitted with slots to hold fuel elements and maintain this spacing. The fuel elements will be suspended in air. This array would remain subcritical even if the cell were flooded with light water. The inventory of heavy water in the vicinity of the Critical Facility will not be sufficient to cover fuel elements in the storage racks, even if it should all drain into the cell. During reactor operation neutrons from the reactor will reach fuel elements stored in the racks. Resulting fissions would return some neutrons to the reactor. However, the effect on the reactor is expected to be small. Even with the reactor acting as a source, the geometry is such that stored fuel could not achieve criticality.

Irradiated Fuel Elements

Irradiated fuel elements to be tested in the Critical Facility will be transferred from the PRTR. While awaiting transfer into the Critical Facility, these elements will be stored in the PRTR storage basin in a critically safe 8×10-inch array in light water.

Unirradiated Fuel Shipping

Fuel elements being brought into the Critical Facility will be handled by operations personnel. Handling of the fuel elements will be in conformance with Process Specifications and Operating Procedures designed to prevent achievement of a critical array.

Disposal of Wastes

Aqueous Wastes

Waste water from the Critical Facility, both heat exchanger cooling water and water pumped from the sump or the fuel transfer lock, will be transferred to the PRTR process sewer. A valve, which automatically closes upon receipt of a PRTR aqueous effluent trip signal, is installed in the Critical Facility waste line. This system is also constructed such that radioactive wastes can be transferred to hot waste hold-up tanks. This system is described more fully in the PRTR Final Safeguards Analysis.[5] A manually operated valve will allow water from the Critical Facility to be routed to the hold-up tanks. The only PRTR aqueous waste that is routed through the same line as Critical Facility wastes is the storage basin overflow.

Gaseous Wastes

A ventilation air flow of 500 cfm will travel through the reactor cell. This air will be exhausted to the PRTR exhaust system. It will pass through an absolute filter (99.97 percent removal efficiency for particles of 0.3 micron diameter), an activated charcoal filter (98 percent removal efficiency of iodine), and be exhausted through a 150-foot high stack. The Critical Facility exhaust line and the ventilation supply line will be provided with confinement valves. The exhaust air activity will be monitored by an HM chamber. This chamber will monitor the activity from both the PRTR and the Critical Facility. A high activity signal will trip safety circuits in both the PRTR and the Critical Facility, and will initiate PRTR ventilation containment. This signal will also close valves in the Critical Facility ventilation supply and exhaust lines. A check valve in an alternate exhaust line will open if cell pressure increases (fully open at 2 psig). Opening of this valve will allow gases to escape through two activated charcoal filters (each filter having a 95 percent removal efficiency for halogens), then through the PRTR exhaust system filters to the stack.

Sabotage and Nonnuclear Incidents

Sabotage

The facility will be manned with operating personnel during all operating periods. When the facility is unattended, all irradiated fuel elements will be returned to the PRTR storage basin, and the reactor will be deactivated in a manner which minimizes the possibility of unauthorized operation of the equipment. All control and safety rods will be fully inserted, the moderator will be completely drained from the reactor tank, and all power sources for the above will be de-energized and locked out. Access to the doors leading to the control area will be through the PRTR service building which will remain occupied by PRTR Operation personnel.

Bombing

No protection against bomb and external missiles is incorporated in the design of the control area. The reactor cell, which will be constructed below grade level with 2- and 3-foot thick reinforced concrete walls and a 4-foot thick concrete cover, should provide considerable protection for contained equipment.

Earthquake

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

Windstorm

The control area building is designed to resist wind pressures in accordance with the Uniform Building Code, Section 2307. The remainder of the facility will be below grade.

Floods

The floor of the control area will be 35 feet above the estimated 100-year maximum flood stage. The site, therefore, offers safety from natural Columbia River floods.

Interaction of Critical Facility and PRTR

Nuclear Interaction

The PRTR and the Critical Facility Reactor will be separated by more than 100 feet of earth, concrete, and light water. Even if the light water in this path were replaced by air, no nuclear interaction would be possible.

Interaction of Systems

Services

The Critical Facility will be supplied with process water and instrument air from the PRTR systems. Normal usage rates for these services in the Critical Facility will be about 10 cfm of air and up to 50 gpm of process water. Maximum rates of process water use would be about 100 gpm. The PRTR systems are adequately sized to provide the Critical Facility needs and still maintain adequate supplies for the PRTR and planned test loops.

Confinement and Containment Systems

The Critical Facility and the PRTR will share the same activity monitors for both exhaust air and aqueous effluent. Activity released from either facility in sufficient quantities to reach the trip point will trip the safety circuits of both reactors, initiate containment of the PRTR, and trip the confinement circuits of the Critical Facility. An exhaust air trip will scram the PRTR and the Critical Facility, automatically stop the PRTR ventilation exhaust fan and the Critical Facility supply fan, initiate automatic ventilation containment of the PRTR, and automatically close the Critical Facility ventilation system supply and exhaust valves. An aqueous effluent monitor trip will scram both reactors, initiate automatic aqueous PRTR containment, and automatically close the Critical Facility valve in the line leading to the PRTR process sewer.

An exhaust air trip would present no hazard to either reactor. Essential systems would continue to operate.

An aqueous effluent trip would shut off the normal escape path for steam generated in the PRTR primary coolant system heat exchanger. Relief valves would allow the steam to vent through lines submerged in the PRTR discharge water pit. Aqueous confinement in the Critical Facility would present no problems unless a short-cooled irradiated fuel element were in the facility. Closure of the valve in the effluent line would stop the flow of cooling water through the heat exchanger. However, if the cooling system had been operating with circulating coolant, the water contained in the system would not reach the boiling point for over one-half hour. Plutonium-aluminum fuel element cores would not reach their melting point for at least 1-1/2 hours. During this time the source of contamination could be determined through readings of various activity monitors, and emergency cooling provisions could be made.

Radiation Release Consequences

The maximum credible accident in the PRTR [6] would produce radiation dose rates of several hundred r/hr at the Critical Facility. These dose rates, resulting from fission products confined in the PRTR containment vessel would continue to be high for several days, requiring temporary abandonment of the Critical Facility.

Abandoning the facility would present no risk unless a short-cooled irradiated element were present. Tripping of the PRTR exhaust air or aqueous containment circuit would also trip the rod safety circuits in the Critical Facility, insuring that the reactor would remain sub-critical. With no short-cooled elements in the facility, there would be no risk of fuel melting or fission product release.

If a short-cooled irradiated fuel element were being tested in the Critical Facility, melting of the fuel element could occur if the normal cooling system should fail. However, in the immediate vicinity of the facility, the radiological consequences of a fuel element meltdown in the Critical Facility would be far less severe than those of the maximum credible accident in the PRTR. At a distance from the facility, radiological consequences of the release from the Critical Facility would be at most comparable to those of the PRTR incident. Thus, the radiological consequences of a compound incident in inhabited areas would not be more than a factor of two greater than that of the PRTR incident alone. No serious exposure of residents of surrounding areas would result. Release of fission products through melting of a fuel element in the Critical Facility would increase radiation levels at the PRTR. The radiation levels at this location would not be high enough to require evacuation of the PRTR, however, even under the most unfavorable atmospheric conditions.

VII. MAXIMUM CREDIBLE ACCIDENT

General

In Section VI, Safety Analysis, possible nuclear excursions and equipment malfunctions were analyzed. In no credible case did a nuclear excursion lead to melting of the core with consequent release of fission products to the reactor cell. For the loss-of-coolant accidents, no possibility of fuel melting would exist unless a freshly discharged PRTR fuel element were being tested in the facility. If coolant were lost to such an element, times ranging from 14 minutes to more than an hour were available in which to manually provide emergency cooling and prevent jacket melting.

Certain credible combinations of failures could lead to melting of a recently discharged PRTR fuel element. The course of such an accident is discussed below.

Course of Maximum Credible Accident

Events Leading to Accident

The maximum credible accident in the Critical Facility would be the complete loss of coolant to a recently discharged PRTR fuel element being transferred or tested in the facility, and simultaneous failure of the cell crane. The accident could result from a combination of events such as the following:

1. failure of both the normal and emergency cooling systems; or simultaneous rupture or blocking of both the inlet and outlet coolant lines such that it would be impossible to furnish either normal or emergency coolant to the fuel element; and
2. mechanical failure of the cell crane; or failure of both the normal and emergency power systems. Mechanical or power failure when the crane is positioned over the reactor would prevent the removal of the thimble.

Melting of Fuel Element

Following loss of coolant, the fuel element temperature would increase. Assuming that the fuel element remained in the thimble and was heated adiabatically, complete melting of the inner rod of a Pu-Al element at the maximum power point could occur in about 21 minutes. Although the temperature would be well below the melting point of the jacket (1200 F compared to 3314 F), the expansion of the aluminum upon melting might rupture the jacket. If not, the temperature rise would continue until, about 45 minutes after coolant loss, the melting point of the jacket would be attained. Melting of the cores would be gradual, extending over a period of approximately 1 hour.

Metal Water Reaction

Molten aluminum and zirconium could conceivably drop into pools of water on the floor of the reactor cell. Experimental studies of molten-metal-water reactions by Higgins showed that the percentage completion of these reactions was dependent upon the particle size and the metal temperature.[7] It was found that the reactivity of molten aluminum was nil at temperatures up to 1170 C (2140 F) under the conditions of the Aerojet work.

No mechanism for dispersion of the molten metal into fine droplets as it runs or falls into the water in the reactor tank or on the floor has been postulated. However, it is assumed that the molten aluminum contacting the water will react to 10 percent completion. The amount of zirconium cladding is small compared with the amount of aluminum in the fuel cores (about 0.21 lb-mol zirconium compared to 1.21 lb-mol aluminum), and the heat of reaction per lb-mol for zirconium is only about 60 percent that for aluminum. For these reasons, it was assumed that the energy contribution of the zirconium-water reaction would be negligible as compared with that of the aluminum-water reaction. Based on a 10 percent reaction completion for aluminum, the amount of metal reacted would be 3.28 pounds, or 0.121 lb-mol.

The aluminum-water reaction would liberate hydrogen. The reaction would take place in a pool of water and, consequently, the hydrogen would be liberated as part of a steam-water mixture. In mixing with air, some steam would be condensed by the cooler air. The uncondensed water vapor and hydrogen would disperse through the cell atmosphere.

The hydrogen concentration in steam escaping from a 30-gallon pool on the floor of the cell (the volume of water which could drain from the thimble-cooling system) would be about 7 percent by volume. This is below the lower limit for downward propagation of flame, 9.0 percent.[8] No explosion would be possible.

For hydrogen-air-water vapor mixtures containing more than 30 percent water vapor, the lower limit of flamability for hydrogen is greater than 7 percent.[9] The hydrogen would disperse in the air together with the water vapor. After complete dispersion of all hydrogen generated from the reaction, the hydrogen concentration in the air would be less than 1.5 percent, as compared with the lower limit of flamability for near-dry air, about 4.0 percent. At all times during the dispersion, the hydrogen concentration in the water vapor-air mixture should be below the lower limit of flamability. It is concluded that no hydrogen burning would occur.

The heat generated by the aluminum-water reaction was determined from the following relationship:

$$2\,Al + 3\,H_2O \rightarrow Al_2O_3 + 3\,H_2;\ \Delta H = -399,776\ \text{Btu/lb-mol.}$$

The excess energy contribution of the reaction of 0.121 lb-mol of aluminum would be 38,000 Btu.

Air Escape from Cell

At the time that radioactive fission products from the ruptured fuel element reached the exhaust air activity monitors, the PRTR exhaust air activity circuit would trip. Valves would operate to close off the normal exhaust line and route gases escaping from the cell through the activated charcoal filters, and then to the PRTR exhaust system absolute filter. The PRTR exhaust fan and reactor cell air supply fan would stop at the time of the high activity trip.

The heating and evaporation in the reactor cell would force gases, including fission products, through the exhaust line and the filters, where more than 99.5 percent of the halogens and at least 99.95 percent of the particulate fission products would be removed. The gases would then be vented through the 150-foot high PRTR stack. Containment valves in the PRTR exhaust line would have been closed by the high activity trip, preventing diffusion of fission products into the PRTR containment vessel.

At the time that the molten metal reacted with the water in the cell, there would be a rapid increase in the volume of gas and vapor in the cell, resulting in flow rates through the exhaust line of up to 50 scfm. The fission product decay heat in the fuel element would continue to increase the temperature and humidity in the cell, exhausting more gas and entrained fission products, until the cell air temperature reached equilibrium at an estimated temperature of 200 F. After this peak was reached, the temperature would gradually decline as the fission product heat generation rate decreased.

The cell will be sealed to prevent leakage around access opening cover blocks, etc. During the heating of the cell air and the boiling of water, escape of gases through the alternate exhaust line would limit the cell pressure to 2 psig and essentially no leakage through other paths should occur.

Figure 24 shows the rate of gas escape through the exhaust filters as a function of time. If the fission products escaped in the same ratio as the air in the cell which is replaced by water vapor, total escape of fission products from the cell by the time the cell atmosphere reached equilibrium would be about 80 percent of the total released from the fuel, assuming no condensation in the cell.

In these calculations, it was assumed that the meltdown, and metal-water reaction proceeded linearly over a period of 30 minutes, although complete melting would require more than an hour.

AEC-GE RICHLAND, WASH.

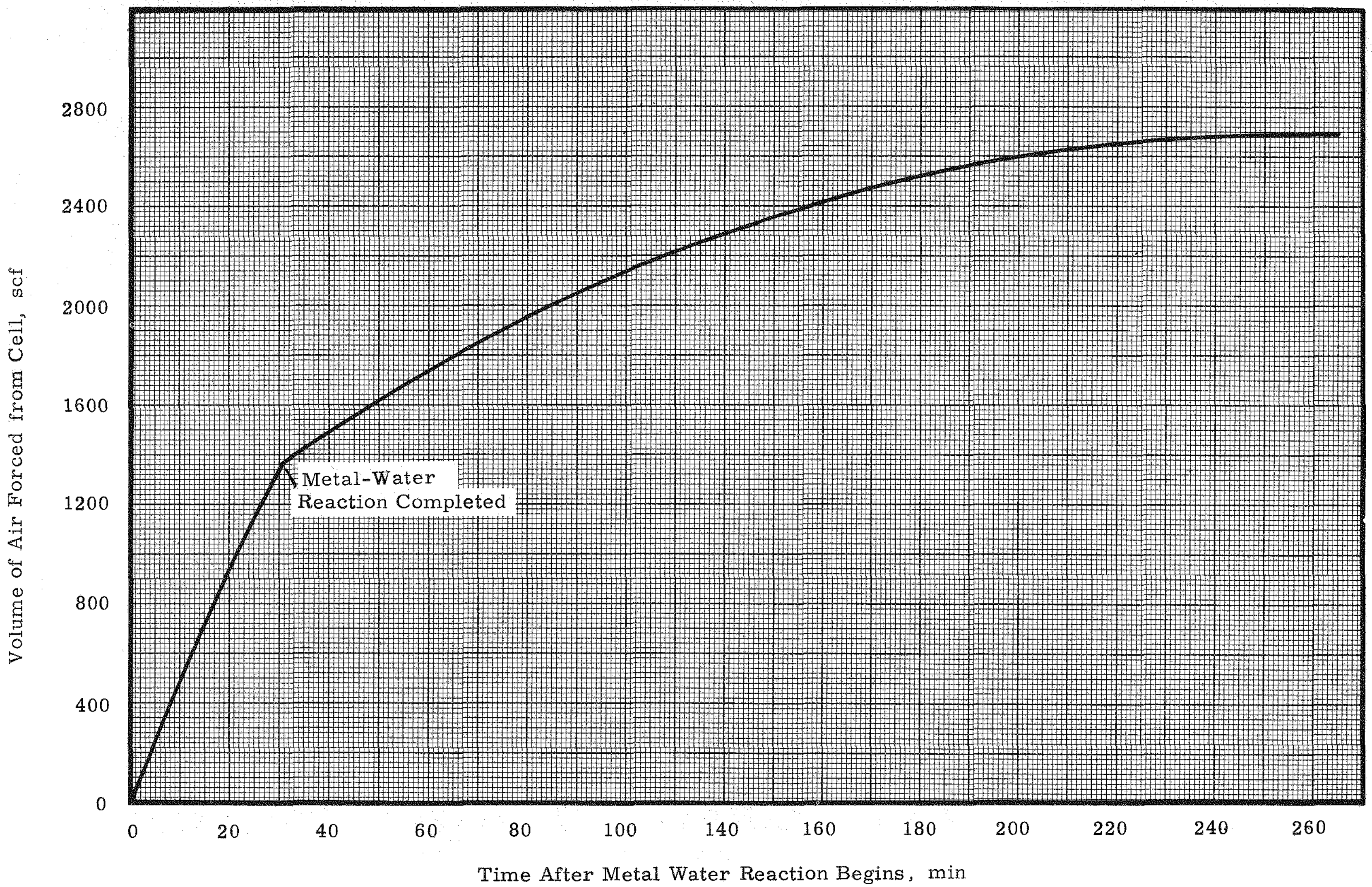

FIGURE 24

Escape of Reactor Cell Atmosphere Following Maximum Credible Accident

VIII. RADIOLOGICAL CONSEQUENCES OF MAXIMUM CREDIBLE ACCIDENT

The accident chosen for analysis of the radiological effects consisted of the gradual meltdown of an irradiated PRTR fuel element in the Critical Facility reactor cell, with release of a portion of the fission products to the cell atmosphere. As heating of the cell atmosphere and vaporization of water produced a slight positive pressure in the cell, fission products would be carried with escaping gases through the exhaust lines to the PRTR 150-foot stack. The activated charcoal filters and absolute filter would remove most of the fission products. The concrete walls of the reactor cell and the earth covering the exhaust line would provide adequate shielding against direct gamma radiation from fission products contained in these locations. The bulk of the hazard would be from distribution of fission products which escaped the filters.

Inventory of Radioisotopes

Radioisotopes considered to be of importance are the fission products, Np^{239}, and the isotopes of plutonium. Fission product inventories were calculated for UO_2 and Pu-Al fuel elements, irradiated to 5000 MWD/T and 50 percent burnup, respectively. Fuel element powers while operating in the PRTR were assumed to be 1200 kw. These are predicted maximum values for PRTR fuel elements. The quantities of fission products in each of the four categories were obtained from previous computations[10, 11] to allow estimation of the quantities released. These categories are:

1. noble gases - krypton and xenon
2. halogens - iodine
3. volatile solids - cesium, tellurium,selenium, ruthenium, and
4. nonvolatile solids - all others

With the fuel element melting not less than three hours after reactor shutdown, radioactive krypton isotopes of mass number 89 and higher and xenon isotopes of mass number 137 and greater would have decayed to almost nonexistent quantities. This is of importance in a confinement system where such noble gases could pass through filters and, subsequently, decay into radioactive isotopes of strontium and cesium.

The estimated quantities of radioisotopes present in a Pu-Al and UO_2 fuel element are presented in Table IV.

Volatilization of Radioisotopes

In the meltdown of a Pu-Al fuel element, it is possible that the entire fuel element core would melt. The melting point of the core is 1220 F as compared with 3314 F for the jacket. For a UO_2 fuel element, the core melting point is about 4990 F, well above the jacket melting temperature. It is doubtful that any UO_2 cores would melt following loss of coolant. As shown in Table IV, the total quantities of radioactive isotopes contained in the two types of elements would be comparable. The quantities of isotopes released will be proportional to the fraction of fuel melted. As complete Pu-Al core melting is possible, and as little or no melting of UO_2 cores is expected, release studies were based on the Pu-Al element meltdown as a worst case.

In order to permit calculations on the consequences of accidents, a general formula has been devised by the General Electric Technological Hazards Council utilizing experimental laboratory data from Oak Ridge National Laboratory, MSA Research Corporation, and Hanford. Using this formula, and assuming 100 percent melting of the Pu-Al core, the following fractions for release from the fuel were assumed:

Noble gases	1.0
Halogens	0.5
Volatile solids	0.5
Nonvolatile solids	0.01

Total quantities of radionuclides released to the reactor cell are presented in Table V.

TABLE IV

QUANTITIES OF RADIONUCLIDES IN Pu-Al AND UO_2 FUEL ELEMENTS AT VARIOUS DECAY TIMES

(curies)

Radionuclides	Type of Fuel Element	Decay Time 3 hr	10 hr	1 day	10 days	30 days	100 days
Noble gases	UO_2, Pu-Al	1.8×10^5	1.4×10^5	1.0×10^5	2.8×10^4	3.2×10^2	1.2×10^2
Halogens	UO_2	2.0×10^5	1.5×10^5	1.1×10^5	1.9×10^4	2.6×10^3	-
	Pu-Al	2.0×10^5	1.5×10^5	1.1×10^5	1.9×10^4	2.5×10^3	-
Volatile solids	UO_2	1.1×10^5	8.8×10^4	7.8×10^4	3.8×10^4	2.5×10^4	9.5×10^3
	Pu-Al	1.0×10^5	7.7×10^4	6.6×10^4	3.1×10^4	1.7×10^4	1.0×10^4
Nonvolatile solids	UO_2	1.2×10^6	8.8×10^5	7.3×10^5	4.5×10^5	3.0×10^5	1.3×10^5
	Pu-Al	1.1×10^6	7.5×10^5	5.9×10^5	3.3×10^5	1.9×10^5	7.8×10^4
Fission products	UO_2	1.5×10^6	1.1×10^6	9.0×10^5	5.2×10^5	2.3×10^5	9.1×10^4
	Pu-Al	1.7×10^6	1.3×10^6	1.0×10^6	5.3×10^5	3.2×10^5	1.4×10^5
Np^{239}	UO_2	7.5×10^5	7.0×10^5	6.4×10^5	5.6×10^5	3.7×10^4	75
Pu^{239}	UO_2	9	9	9	9	9	9
	Pu-Al	6	6	6	6	6	6
Pu^{240}	UO_2	10	10	10	10	10	10
	Pu-Al	8	8	8	8	8	8
Pu^{241}	UO_2	500	500	500	500	500	500
	Pu-Al	800	800	800	800	800	800
Pu^{242}	UO_2	0.003	0.003	0.003	0.003	0.003	0.003
	Pu-Al	0.004	0.004	0.004	0.004	0.004	0.004

TABLE V

QUANTITIES OF RADIONUCLIDES RELEASED TO CELL FROM MELTING Pu-Al FUEL ELEMENT

(curies)

Radionuclides	Decay Time 3 hr	24 hr
Noble gases	1.8×10^5	1.0×10^5
Halogens	1.0×10^5	5.3×10^4
Volatile solids	5.0×10^4	3.3×10^4
Nonvolatile solids	1.1×10^4	5.9×10^3
Total fission products	3.4×10^5	1.9×10^5
Pu^{239}	0.06	0.06
Pu^{240}	0.08	0.08
Pu^{241}	8	8
Pu^{242}	-	-

Escape from Confinement Structure

As the air in the cell is heated and as water in the cell is vaporized, the pressure would be relieved by air and steam escaping through the alternate exhaust line. Fission products would travel out with the escaping gases. The activated charcoal filters and the absolute filter in the exhaust system would remove at least 99.95 percent of the particulate fission products and 99.5 percent of the iodine. The fission products which escaped the filters would be exhausted through a 150-foot high stack.

An evaluation of the whole-body and thyroid doses for an adult at the centerline of the cloud was made for two different atmospheric conditions, strong inversion and neutral. The evaluations were based on a wind speed of 1 meter per second, an effective height of release of 50 meters, with 99.95 percent removal of particulates and 99.5 percent removal of iodine. Whole-body and thyroid doses at various distances from the stack are presented in Table VI. These are total doses assuming that an adult remained at the centerline of the cloud as it passed.

TABLE V

QUANTITIES OF RADIONUCLIDES RELEASED TO CELL FROM MELTING Pu-Al FUEL ELEMENT
(curies)

Radionuclides	Decay Time: 3 hr	Decay Time: 24 hr
Noble gases	1.8×10^5	1.0×10^5
Halogens	1.0×10^5	5.3×10^4
Volatile solids	5.0×10^4	3.3×10^4
Nonvolatile solids	1.1×10^4	5.9×10^3
Total fission products	3.4×10^5	1.9×10^5
Pu^{239}	0.06	0.06
Pu^{240}	0.08	0.08
Pu^{241}	8	8
Pu^{242}	-	-

Escape from Confinement Structure

As the air in the cell is heated and as water in the cell is vaporized, the pressure would be relieved by air and steam escaping through the alternate exhaust line. Fission products would travel out with the escaping gases. The activated charcoal filters and the absolute filter in the exhaust system would remove at least 99.95 percent of the particulate fission products and 99.5 percent of the iodine. The fission products which escaped the filters would be exhausted through a 150-foot high stack.

An evaluation of the whole-body and thyroid doses for an adult at the centerline of the cloud was made for two different atmospheric conditions, strong inversion and neutral. The evaluations were based on a wind speed of 1 meter per second, an effective height of release of 50 meters, with 99.95 percent removal of particulates and 99.5 percent removal of iodine. Whole-body and thyroid doses at various distances from the stack are presented in Table VI. These are total doses assuming that an adult remained at the centerline of the cloud as it passed.

TABLE VI

ESTIMATED DOSES ON CENTERLINE OF CLOUD DOWNWIND
(Wind speed - 1 meter/sec; effective height of release - 50 meters)

	Atmospheric Stability			
	Inversion		Neutral	
Distance, meters	Whole Body, r	Thyroid, rads	Whole Body, r	Thyroid, rads
500	--	--	1	6
1,000	--	--	3	10
5,000	0.001	0.06	0.4	2
10,000	0.02	0.8	0.1	0.6
50,000	0.07	4	0.008	0.04
100,000	0.04	2	0.002	0.01
500,000	0.007	0.2	<0.001	<0.001

Off-site personnel would be restricted to a distance of at least 4000 feet from the facility by an exclusion area.* Table VI shows that for the most severe meteorological conditions, whole body doses at the edge of the exclusion area would be less than 3 roentgen and thyroid doses less than 10 rads. The doses would decrease with distance from the area. Although technical overexposures to off-site persons could result, no serious biological effects would be expected.

The assumptions used in the release calculations are conservative. It was assumed that all fission products released to the cell atmosphere were vented to the exhaust line. No credit was taken for retention of fission products in the cell and exhaust line from condensation and collision with equipment and passage walls. In an actual release case, these factors could reduce the off-site exposures by perhaps an order of magnitude or more. Also, wind shifts or greater wind velocities would further reduce the doses.

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* Exclusion area is defined as an area with restrictions which would prevent use of the land for homesites, industrial sites, public parks, etc.

Based on the data in Table VI, areas requiring possible decontamination, temporary evacuation, and possible milk confiscation were calculated. These areas are shown in Table VII. As this table shows, the only control which might be required is milk confiscation for a maximum distance of 1 mile. Again, the factors of conservatism discussed previously make these conclusions pessimistic. With the 4000-foot exclusion area around the facility, it is very doubtful that any controls would be required for an actual release case.

TABLE VII

ESTIMATED AREAS OF RESTRICTION
FOR TWO METEOROLOGICAL CONDITIONS

	Inversion	Neutral
Temporary evacuation	None required	None required
Decontamination	None required	None required
Milk confiscation	None required	0.2 sq miles

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Critical Facility physics calculations were performed by J. J. Regimbal, Reactor Engineering Development Operation, Hanford Laboratories Operation.

The circuits used for the conduct of the analog computer studies were developed by G. R. Taylor, Instrument Research and Development Operation, Hanford Laboratories Operation.

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APPENDIX

ANALYTICAL FORMULATION OF NEUTRON KINETICS

The time dependent neutron flux is given by:

$$\frac{dn}{dt} = \frac{(\Delta k - B)}{\ell^*} n - \Sigma_i \lambda_i C_i + S$$

$$\frac{dC_i}{dt} = \frac{B_i}{\ell^*} n - \lambda_i C_i$$

$$\Delta k = k_{eff} - \Delta k_f - 1$$

where C_i = concentration of i th group of delayed neutron emitters,

λ_i = decay constant of i th group,

n = thermal neutron flux,

B_i = fractional yield of the i th group,

$B = \Sigma B_i$ = total delayed fraction,

ℓ^* = effective lifetime of neutrons in reactor, sec,

Δk = reactivity disturbance,

S = source density, and

k_{eff} = effective neutron multiplication constant.

The delayed neutron data used was that reported by D. Houser and M. V. Davis in HW-48907. Analog computations were conducted for two different reactor loadings, corresponding to critical moderator levels of 9 and 5 feet. Values used are as follows:

λ_i	$Bi \times 10^5$ 9-foot level	$Bi \times 10^5$ 5-foot level
7.7	14.5	11.6
2.2	87.2	73.6
0.465	245	204
0.400	15.7	13.6
0.170	114	98
0.0458	125	110

Values of other parameters for the two cases are:

	9-foot level	5-foot level
B	6.014×10^{-3}	5.108×10^{-3}
ℓ^*	7×10^{-4} sec	5.5×10^{-4} sec
S	5×10^{4} n/cm^2-sec	7×10^{4} n/cm^2-sec

The fuel temperature coefficient is given by:

$$\frac{\Delta k}{k} = -\alpha(\sqrt{T} - \sqrt{T_o}) \ln p_o$$

where p_o = cold resonance escape probability, 0.868 for Mark I UO_2 fuel elements,

T = volume average fuel temperature, °K,

T_o = initial temperature, °K, and

α = constant, 0.74×10^{-2}.

Fuel temperatures were calculated from the following equations:

$$\frac{dT_f}{dt} = A_1 n$$

$$\frac{dT_m}{dt} = A_2 n$$

$$\frac{dT_a}{dt} = A_3 n$$

where T_f = volume average UO_2 fuel temperature,

T_m = maximum UO_2 fuel temperature, and

T_a = maximum Pu-Al fuel temperature.

A_1, A_2, and A_3 are constants including conversion from neutron flux to specific fuel power, and the specific heats of the fuel materials. The values used were:

	9-foot level case	5-foot level case
A_1	0.985×10^{-12}	1.029×10^{-12}
A_2	1.415×10^{-12}	1.430×10^{-12}
A_3	1.477×10^{-12}	1.488×10^{-12}

Heat transfer from the fuel cores to the moderator during the excursions presented in this report was disregarded.

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