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LOS ALAMOS SCIENTIFIC LABORATORY

OF THE UNIVERSITY OF CALIFORNIA o LOS ALAMOS NEW MEXICO

STATUS REPORT ON THE WATER BOILER REACTOR

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OF THE UNIVERSITY OF CALIFORNIA LOS ALAMOS NEW MEXICO

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STATUS REPORT ON THE WATER BOILER REACTOR

by

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ABSTRACT

This report describes the main design features and operating characteristics of the Water Boiler reactor, as of early 1963. Special emphasis is given to reactor safety and to aspects of the over-all operation which have not been well covered in previous reports.

ACKNOWLEDGMENTS

I wish to acknowledge the cooperation and assistance of several staff members of the Los Alamos Scientific Laboratory in the preparation of this report. I am especially indebted to R. E. Carter, J. W. Starner, M. M. Thorpe, and H. T. Motz (all of P-2) and M. Frechette (H-1) for their contributions.

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I. Introduction

The present Los Alamos Water Boiler reactor, nicknamed "SUPO," is a modification of an earlier design referred to as "HYPO." The conversion from HYPO to SUPO, which was accomplished during the period April 1949 to March 1951, is well described in references 1 and 2. There have been no major modifications in SUPO since March 1951.

The status of the Water Boiler has not been reviewed in any formal report since August 1955, at which time L. D. P. King presented a brief paper[3] on the design and operating characteristics of SUPO at a reactor conference. For this and other reasons, it was considered appropriate at this time to prepare an up-to-date description of the reactor. Included in the present report is a discussion of the inherent safety of the Water Boiler operation.

II. General Description

The Water Boiler is an enriched-uranium homogeneous reactor. Basically, the reactor consists of a 1-ft stainless steel sphere filled with a water solution of uranyl nitrate and surrounded with a graphite neutron reflector. Criticality is controlled with a set of four control rods and a rod referred to as the "safety" rod. Other major components include the "biological" radiation shield, two thermal columns, various access ports, and a gas recombination system. Photographs and cross-sectional views of the reactor are shown in Figs. 1-6. Some general-information statistics on the reactor are given in Table I.

Although the Los Alamos Water Boiler was the first reactor of its general type, its main function has been to provide Laboratory personnel with an intense source of neutrons, rather than to serve as an experiment in reactor design. On the other hand, the fact that the design has proved to be successful has led to the construction of a number of other Water-Boiler-type reactors.

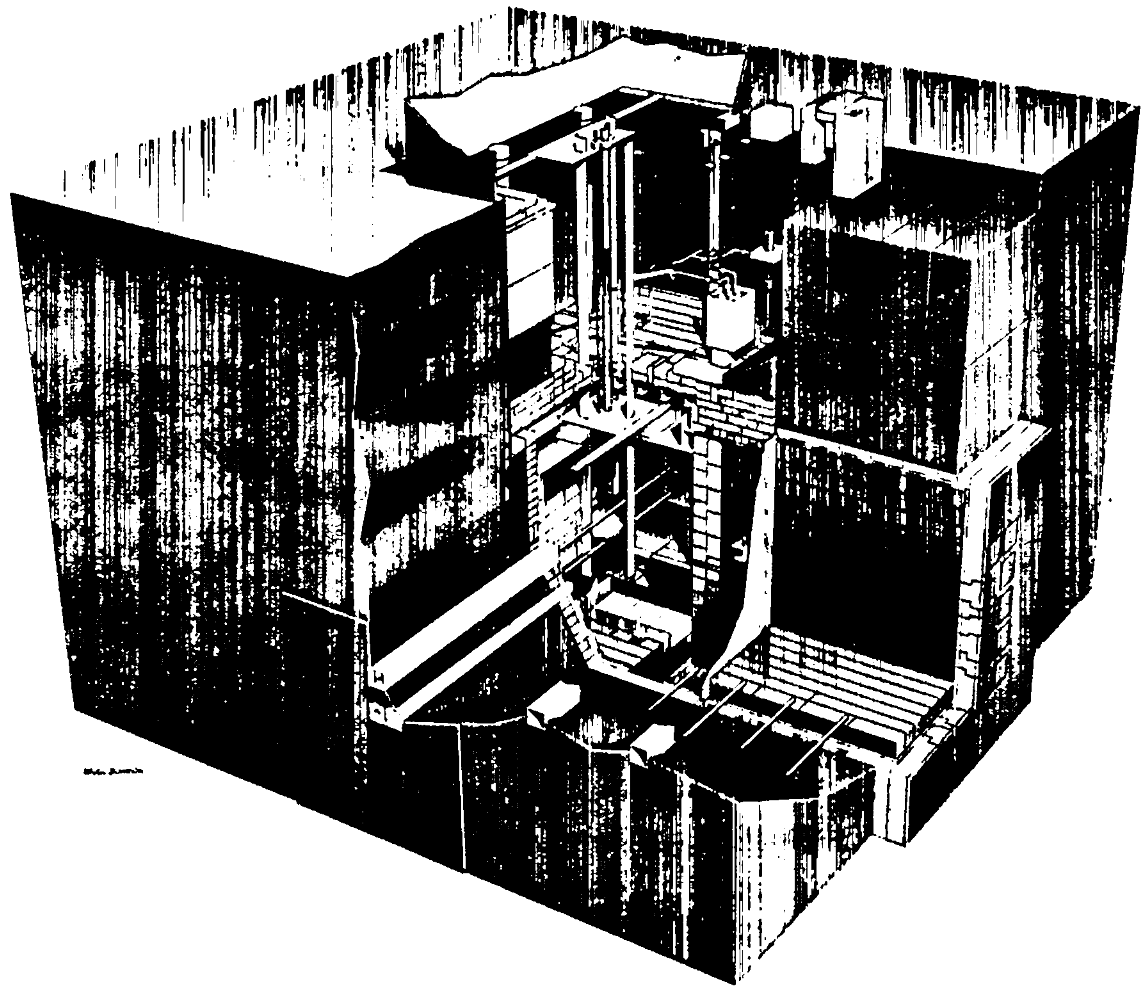


Figure 1. Cutaway view of SUPO, looking northeast.

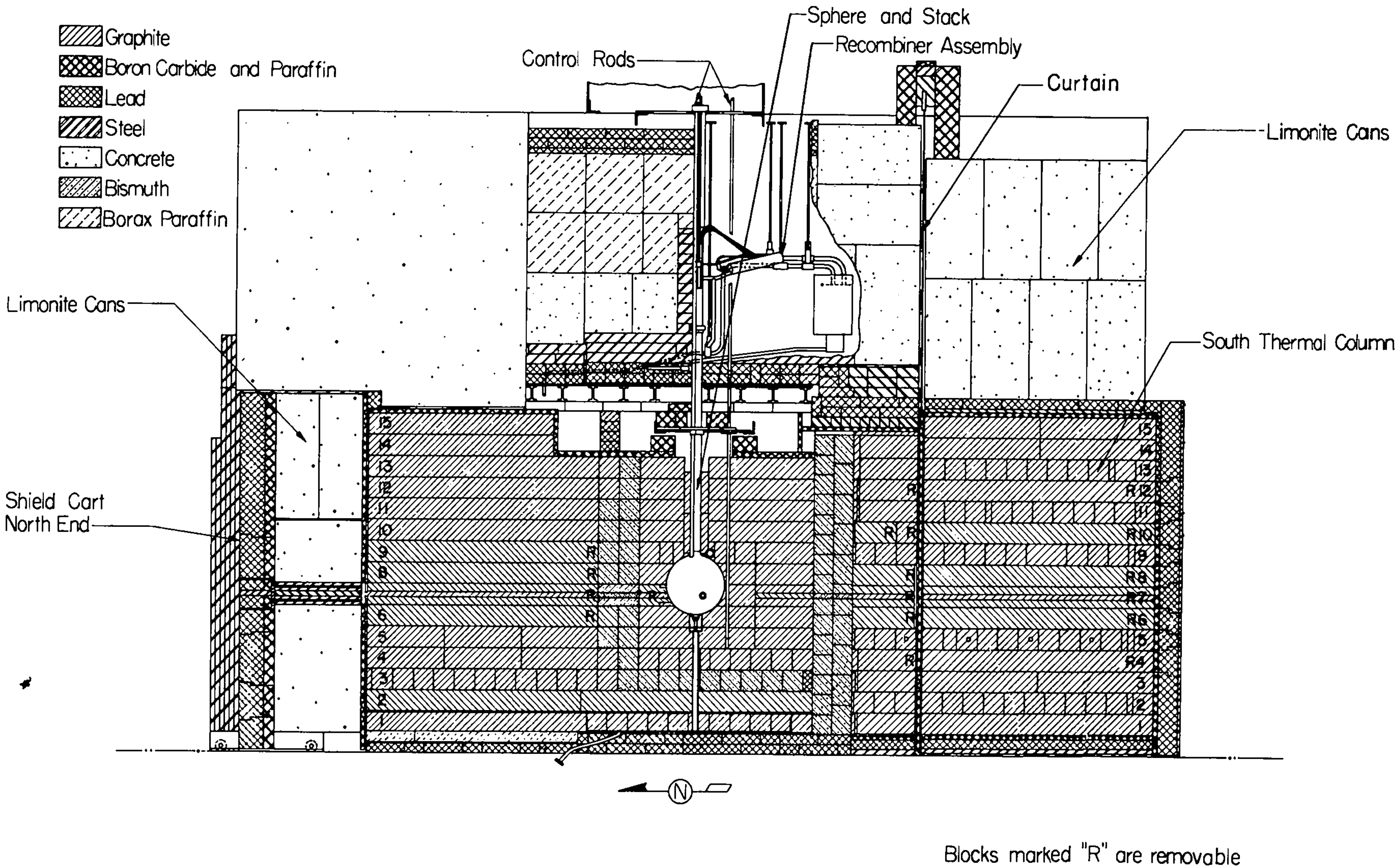


Figure 2. North-south vertical cross section.

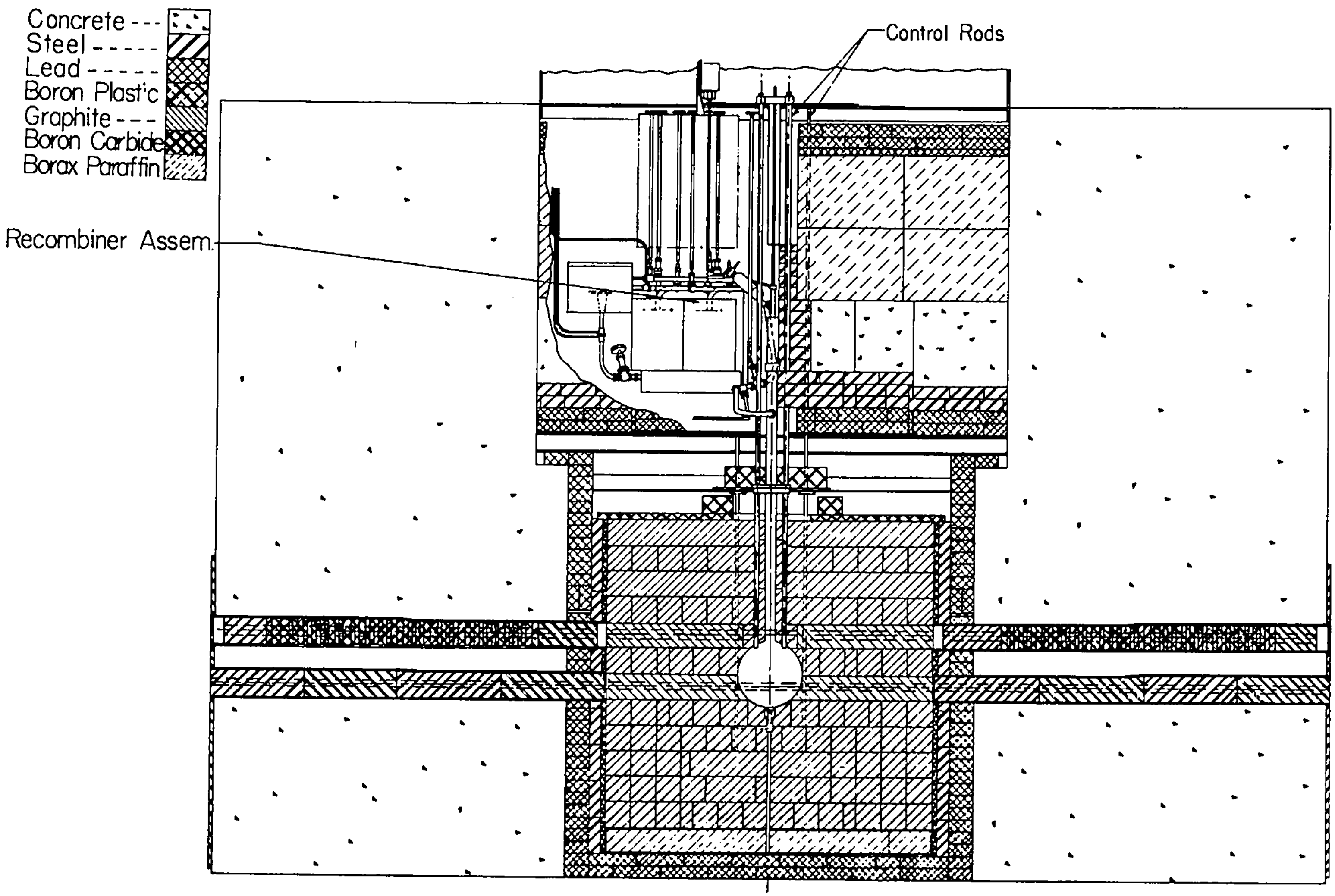


Figure 3. East-west vertical cross section.

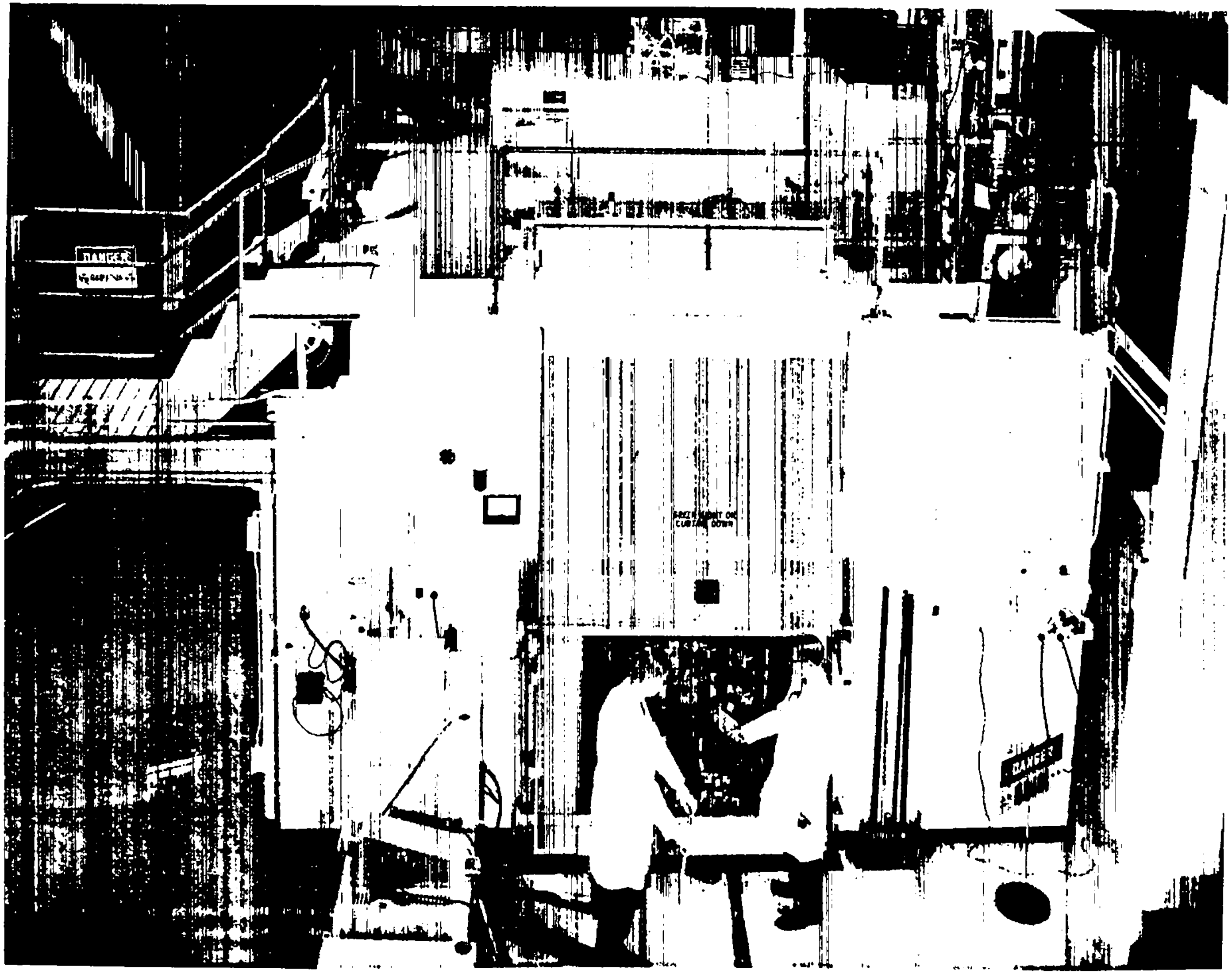


Figure 4. Photograph of south face. A graphite "stringer" is being removed from the South Thermal Column.

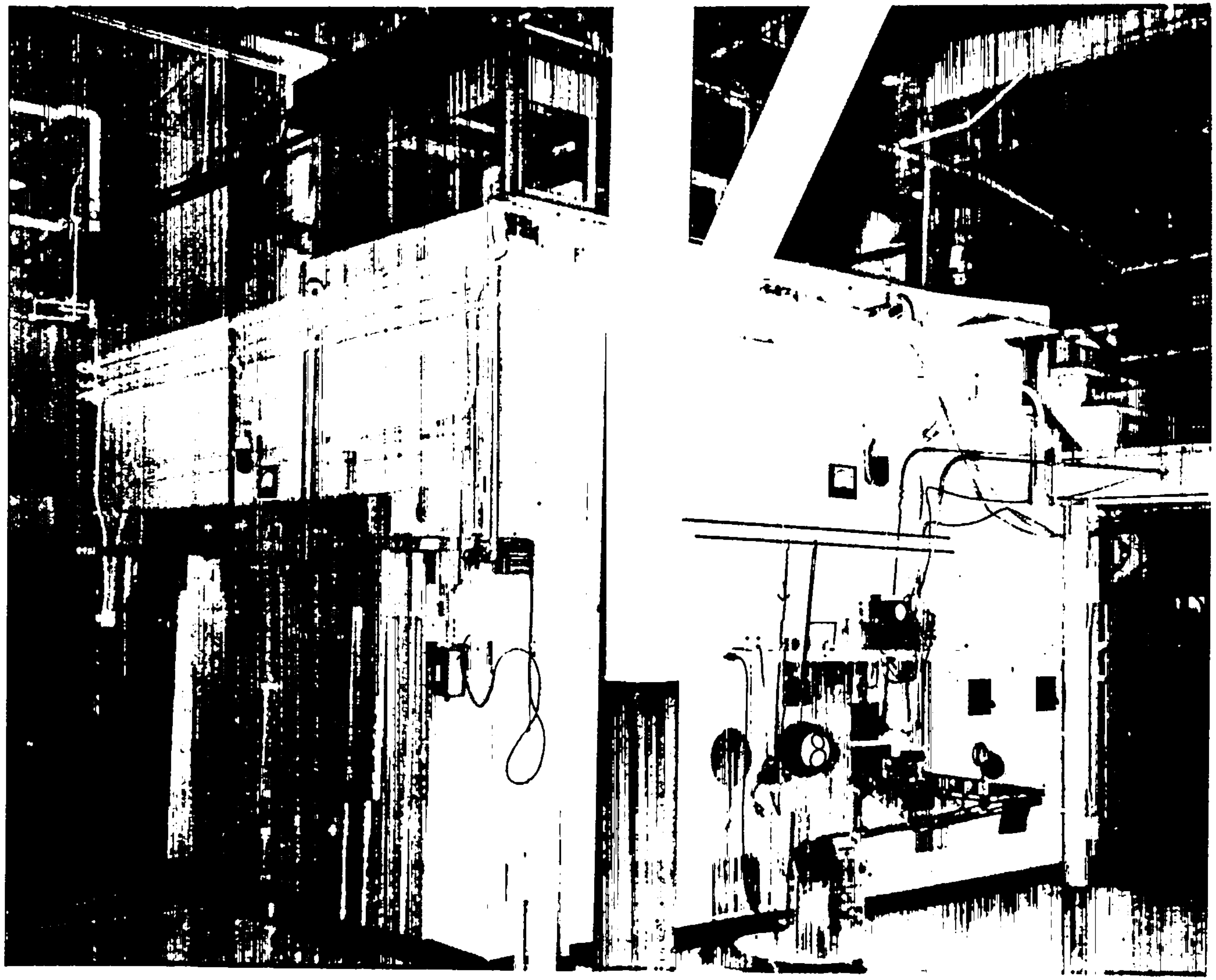


Figure 5. Photograph of the north and west faces.

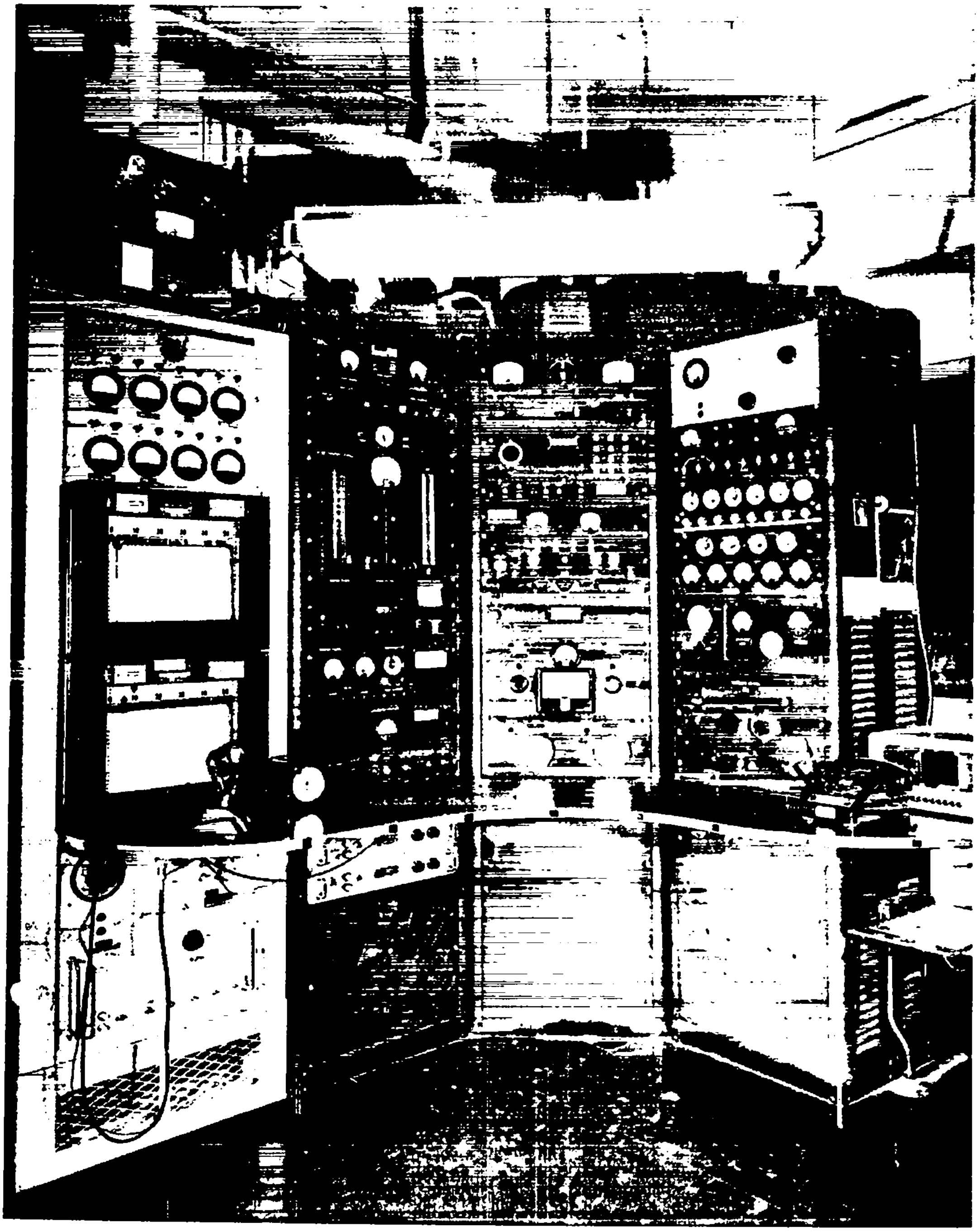


Figure 6. SUPO control panel.

TABLE I

WATER BOILER CHARACTERISTICS

1.	Nominal maximum power	25 kw
2.	Maximum thermal flux (at 25 kw)	9.3×10^{11} n/cm^{2}sec
3.	Critical mass (1949 value)	~780 gm
4.	U^{235} mass (1962)	~950 gm
5.	U^{235} enrichment	88.8%
6.	Uranium solution volume	12.7 liters
7.	Type of solution	$UO_2(NO_3)_2$
8.	Radiolytic hydrogen evolution rate (at 25 kw)	7.3 l/min[a]
9.	Cooling water flow	~3.2 gal/min
10.	Rate of U^{235} burnup	~1 gm/yr
11.	Biological shield dimensions (ft)	15 x 15 x 10½

[a]Measured at a pressure of 580 mm Hg.

III. The Reactor Core

A. Sphere Details

The sphere was formed by welding together two "spun" hemispheres of type-347 stainless steel. Figure 7 shows a bottom view of the sphere assembly just before the welding operation. Inside the sphere are (1) three helical coils of 1/4-in. stainless steel tubing, each 20 ft long, through which cooling water is passed, (2) two re-entrant thimbles which permit the boron control rods to penetrate the sphere volume, (3) a "Glory Hole" port tube, and (4) two "bubbler" tubes (not visible in Fig. 7) used in measuring the solution level. A cross-sectional view of the sphere is shown in Fig. 8, and a list of pertinent data is given in Table II.

Figure 9 shows the sphere assembly in position with the reflector partially constructed. The small tube which comes out of the bottom of the sphere is used for solution addition or removal. In the assembled reactor, the sphere is situated in a 1-ft-diameter spherical cavity in

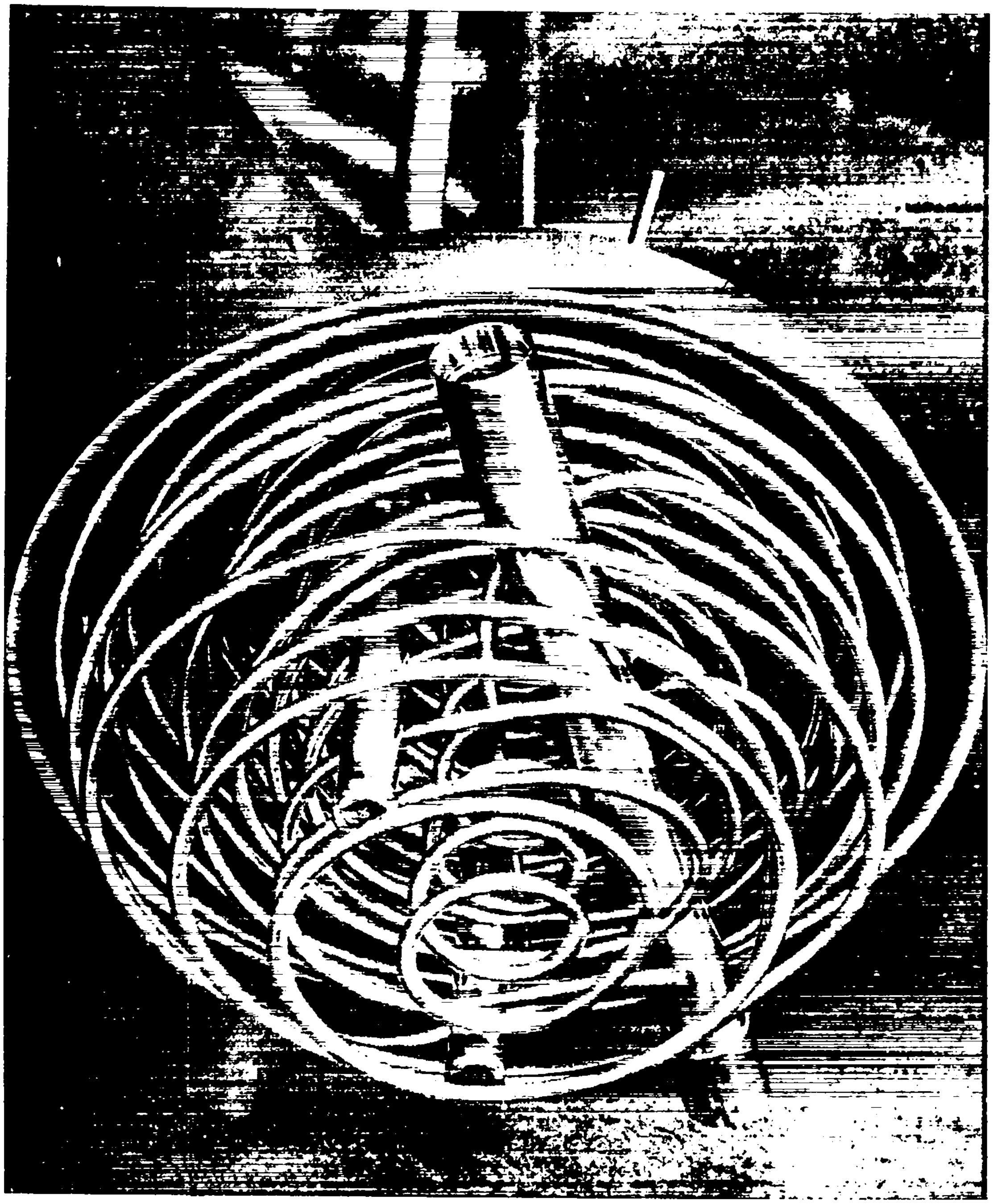


Figure 7. Photograph of sphere interior, with bottom hemisphere removed.

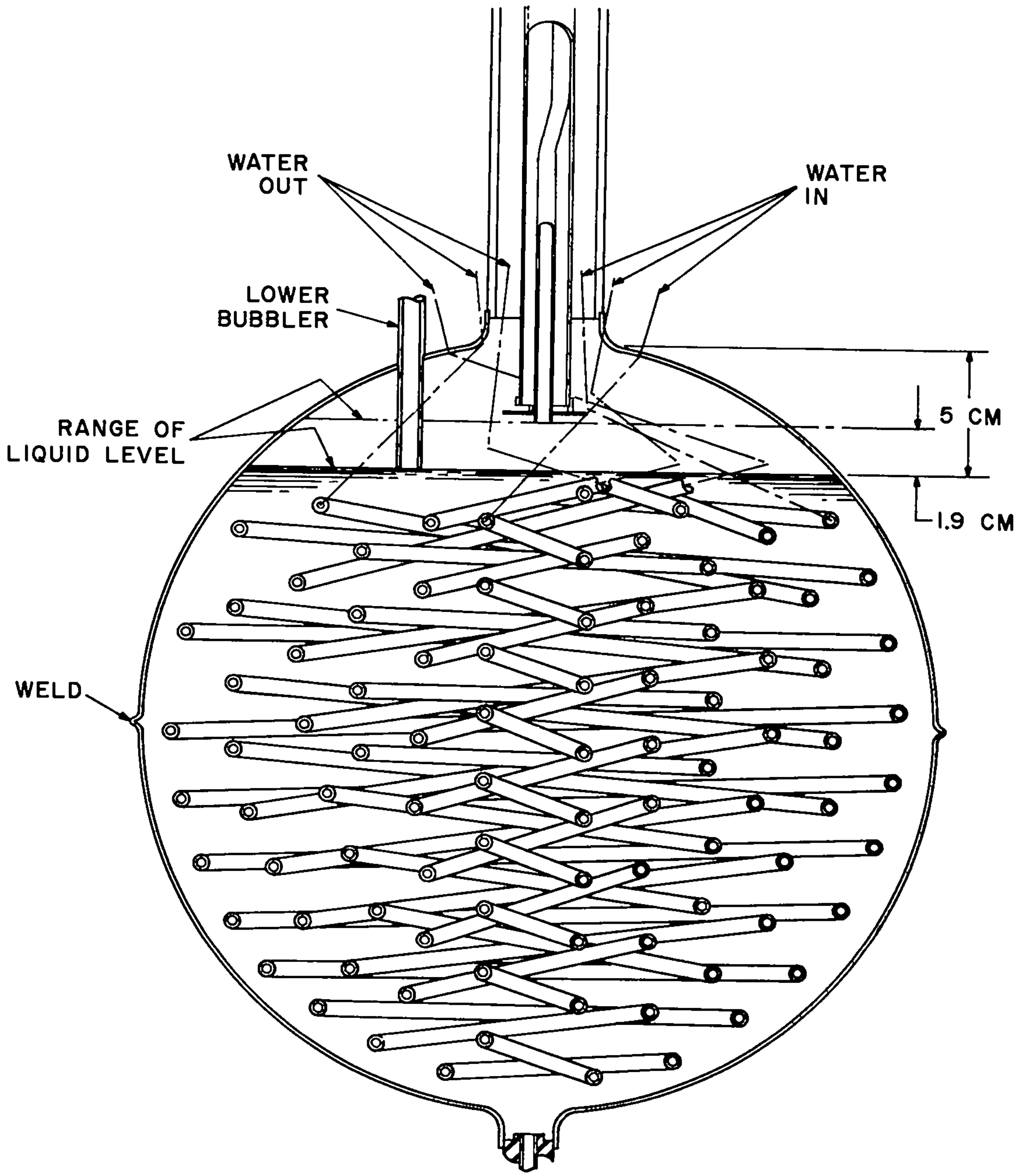


Figure 8. Cross section of sphere assembly.

TABLE II

SUPO SPHERE DATA

1. Sphere		
O.D.	(12 in.)	30.480 cm
I.D.	(11-7/8 in.)	30.163 cm
Thickness of shell	(1/16 in.)	0.15 cm
Inside area		2,858 cm^2
Inside volume		14,368 cc
2. Glory Hole		
Center of sphere to center of Glory Hole		6.43 cm
Length along center line		27.28 cm
O.D.	(1.105 in.)	2.807 cm
I.D.	(1.005 in.)	2.553 cm
Volume of displaced soup		168.8 cc
Mass of stainless steel	(ρ = 7.86)	240 gm
Volume of stainless steel		30.5 cc
Inside volume of Glory Hole		138.8 cc
3. Control rod thimbles		
O.D.	(3/4 in.)	1.905 cm
Length, below soup level	(8-15/16 in.)	22.78 cm
Volume of displaced soup (<u>each</u> thimble)		64.9 cc
4. Cooling coils		
I.D.	(3/16 in.)	0.476 cm
O.D.	(1/4 in.)	0.635 cm
Length (3 coils)	(60 ft)	1,830 cm
Volume of soup displaced	($\sim$57 ft)	$\sim$550 cc
Volume of coils - inside		325 cc
Mass of stainless steel in coils	(ρ = 7.86)	1,995 gm
Volume of stainless steel		254 cc
5. Void-spherical sector		
Height from bubbler level to sphere I.D.		4.84 cm
Volume of void		992 cc
Area of chord plane		388 cm^2
6. Total volume of soup voids in sphere		1,841 cc
		14,368
		- 1,841
7. <u>Calculated</u> volume of soup (at bubbler level)		12,527 cc
8. <u>Measured</u> volume of soup (at bubbler level)		12,696 cc

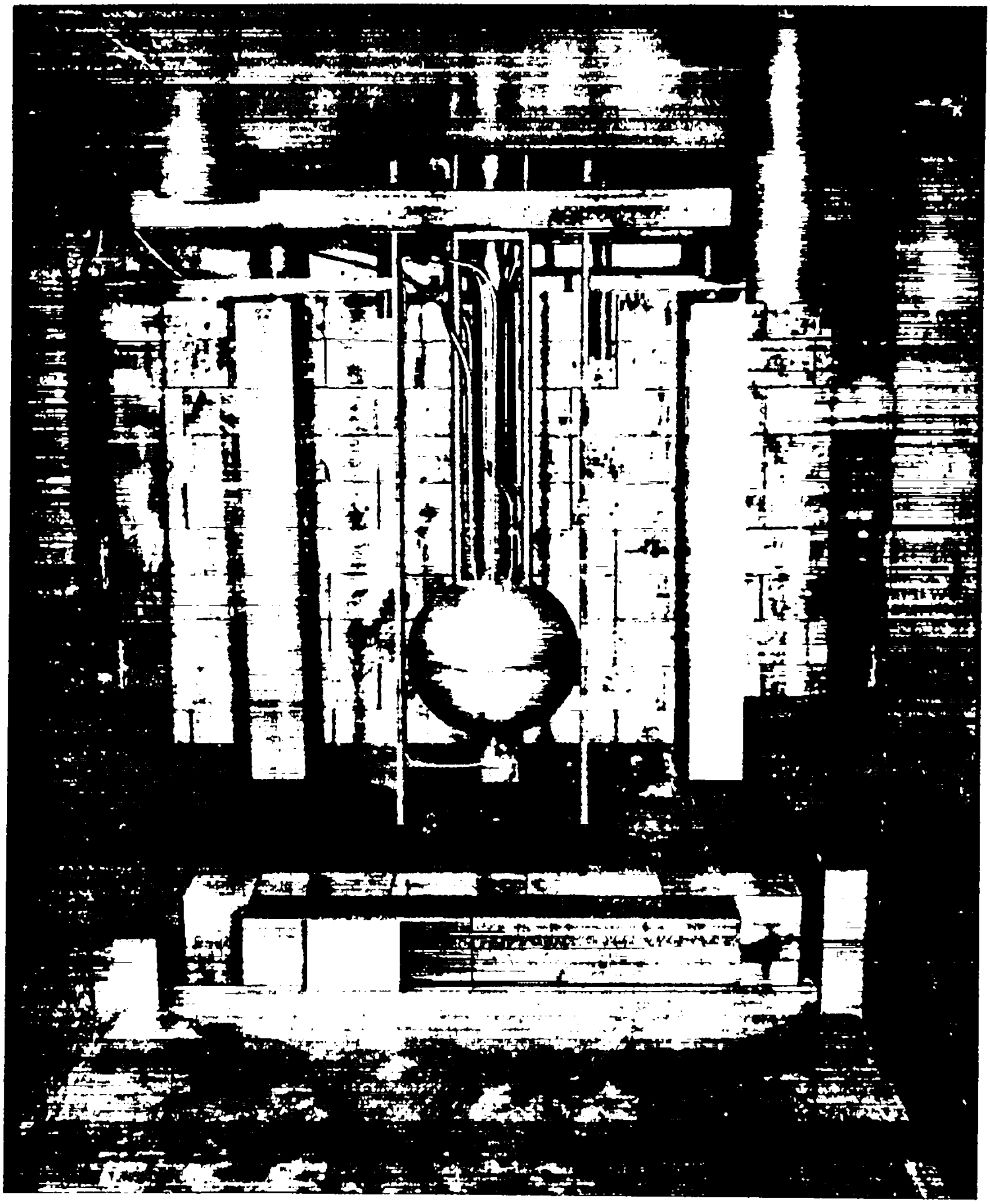


Figure 9. View of SUPO during construction, just after graphite layer #4 had been stacked.

the graphite reflector, i.e., the reflector is essentially in contact with the sphere (see Fig. 2). An Sb-Be start-up neutron source is located in a removable graphite block which contacts the sphere on the north side (layer #7 in Fig. 2).

B. The Uranium Solution (or "Soup")

The Water Boiler "soup" is a solution of uranyl nitrate [$UO_2(NO_3)_2 \cdot 6H_2O$] in dilute nitric acid ($\sim$0.1 N). The approximate chemical composition of the soup in early 1960 is given in Table III. A chemical analysis of the solution is currently made semiannually by group K-2; typical results (July 1960) are shown in Table IV. The Fe concentration, which is determined once a year, is assumed to be an indicator of the corrosion rate of the stainless steel surfaces that are in contact with the soup. A plot of all Fe analyses to date is shown in Fig. 10. There is some evidence that the corrosion rate is slowing down, but further data are needed before one can be confident of this trend. In any case, if the corrosion is occurring uniformly over all exposed surfaces, the 1960 observed Fe concentration of 0.6 mg/cc corresponds to a reduction in wall thickness of only $\sim$0.0001 in. On the other hand, if most of the corrosion has occurred in a localized area, the sphere could conceivably start leaking at any time. The consequences of such a leak are discussed in Section IX.

Because of the radiolytic decomposition of NO_3 ions during operation, it is necessary to add concentrated HNO_3 to the uranium solution periodically. The adopted procedure is to add 185 cc of acid every 1000 kwh. This acid is injected into the sphere through an acid-addition system located on top of the reactor. A full description of the acid-addition system is given in reference 4. The reactor is operated at a power of 1 kw during acid additions; this procedure insures good mixing of the acid with the soup and enables the operator to ascertain, from the change in control-rod position, approximately how much acid has been transferred to the sphere. Addition of 185 cc HNO_3 to the soup induces

TABLE III

APPROXIMATE GRAM WEIGHTS OF SOUP CONSTITUENTS (1960)
FOR A SOLUTION VOLUME OF 12,700 cc

Constituent	Total Grams
U^{235}	950
U^{238}	120
N	190
H	1,410
O	11,960
Stainless steel (in solution)	~10
Fission products	~15

TABLE IV

RESULTS OF ROUTINE SOUP ANALYSES

Values in the top row were obtained from the sample of July 5, 1960, and below these are the observed ranges of values which appear to be acceptable.

	Nitrogen (mg/ml)	pH	U (mg/ml)	Total U (gm)	Fe (mg/ml)
7-5-60	14.99	0.85	82	1037	0.62
Acceptable range	10-17	0.5-1.5	75-85		<1.0

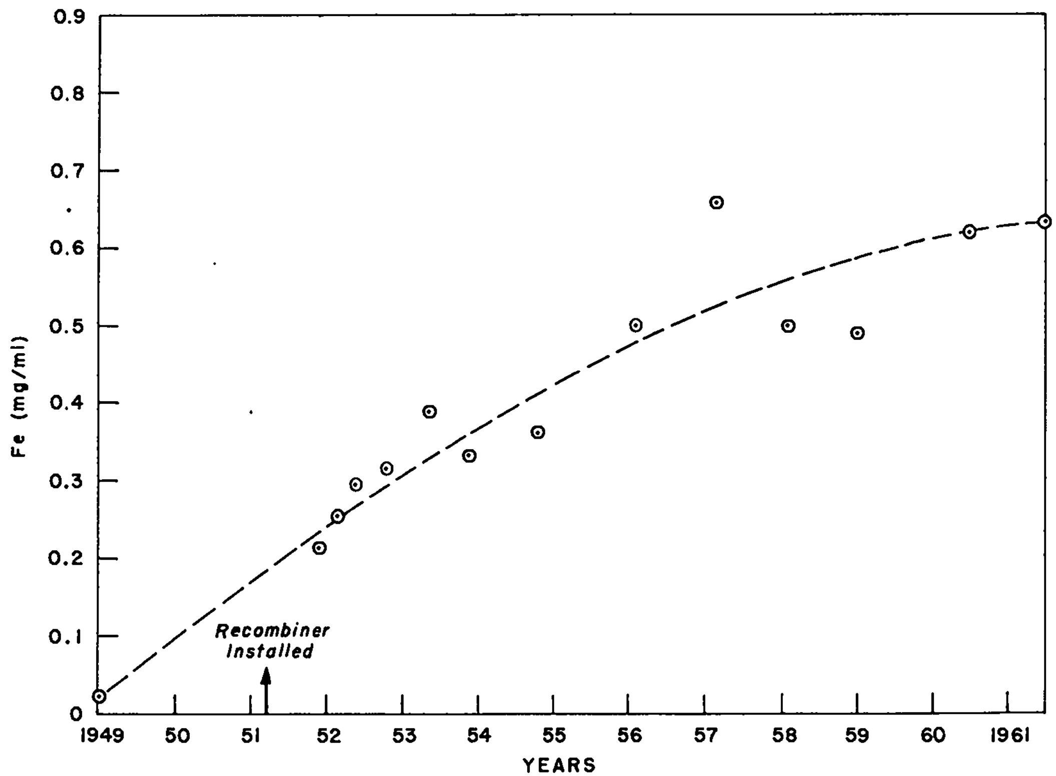


Figure 10. Plot of Fe content of soup vs time.

a negative reactivity change equivalent to ~10 gm U^{235}.

Following each acid addition, a water trap in the recombiner system (labeled "LIQ. TRAP" in Fig. 11) is dumped. The volume of this trap (~82 cc) roughly equals the amount of by-product water formed in the breakdown of 185 cc of HNO_3, the reaction being

$$2HNO_3 \rightarrow H_2O + \underbrace{\text{oxides of nitrogen (or } N_2\text{) + oxygen}}_{\text{e.g., } NO + NO_2 + O_2}$$

Thus, by dumping the trap once every acid cycle, the inventory of water in the sphere is kept approximately constant. In fact, this trap is the primary means of controlling the total solution volume. After the trap has been dumped, about 8 kwh of reactor operation are required to refill the trap with condensate.

On at least two occasions since 1949, a uranium precipitate has formed in the sphere. Chemical tests strongly suggest that this precipitate is uranyl peroxide (trihydrate), $UO_4 \cdot 3H_2O$. Experience has shown that precipitation will not occur as long as the acid concentration is kept above a certain threshold. A standard procedure followed since 1956 is to add an extra 50 cc of HNO_3 to the sphere whenever the soup-addition line becomes partially plugged, which occurs about every 6 months. It is assumed that the observed plugging of this line corresponds to a buildup of precipitate in the bottom of the sphere.

The burnup rate of U^{235} in the Water Boiler, based on an average integrated power of 25,000 kwh/yr, is ~1.1 gm/yr. Thus, many years of operation are possible without fuel addition or fuel processing. In fact, the soup has not been chemically purified since it was transferred to the sphere in 1949.

IV. Gas Recombination System

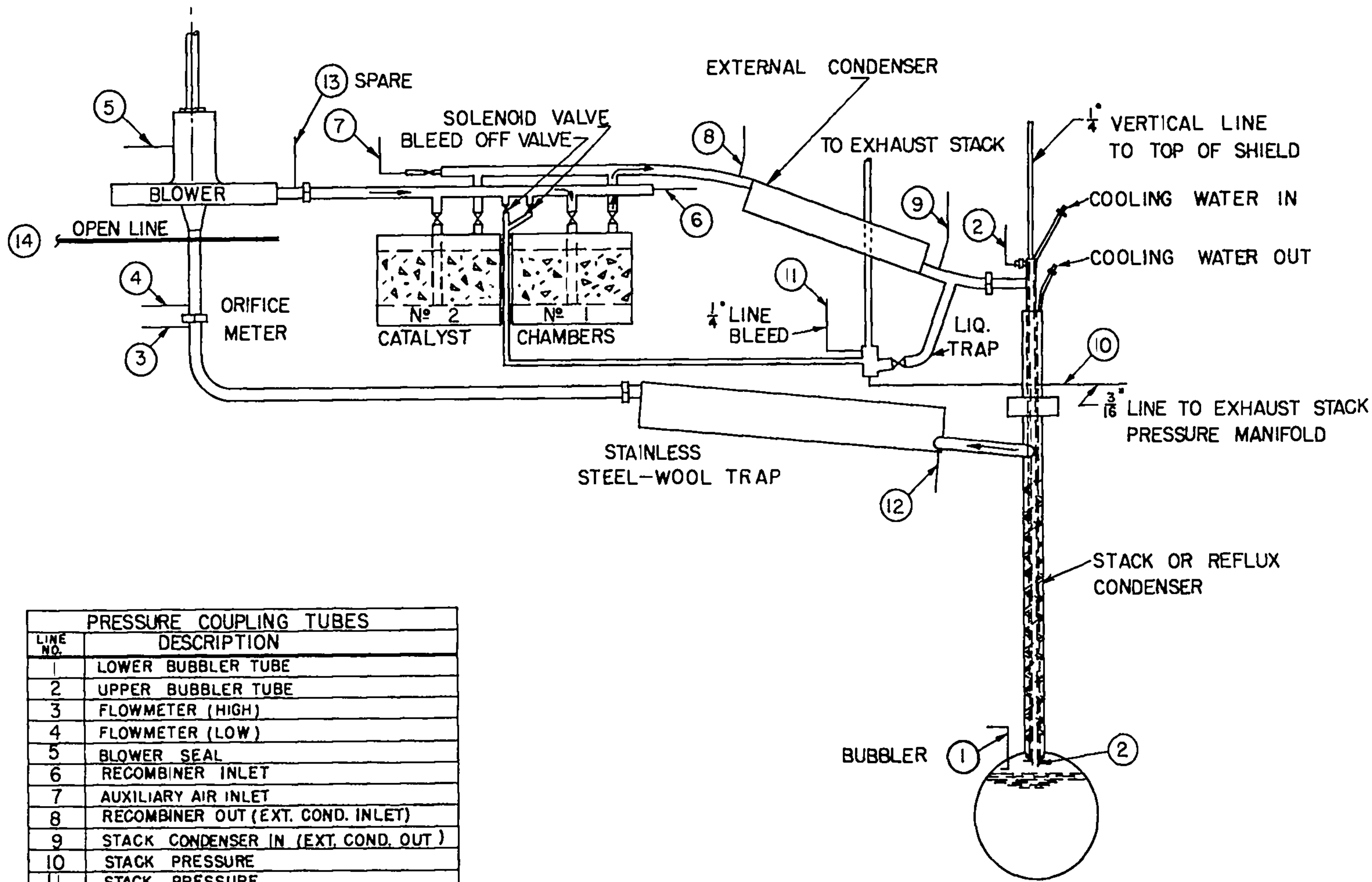
The fission fragments released in the sphere solution during power operation lose part of their energy by dissociating H_2O molecules and

NO_3 ions. At a power level of 25 kw, there are ~7.5 l/min of H_2 and 120 cc/min of gaseous oxides of nitrogen being released from the soup. The gas recombination system, installed in early 1951, was designed to eliminate most of the problems associated with this large gas evolution rate. These problems include:

1. Potential explosion hazard associated with the evolved H_2-O_2 mixture.
2. Necessity of frequent addition of H_2O if the "off" gases were allowed to escape.
3. Undesirability of releasing large quantities of highly radioactive gas to the atmosphere.

The gas recombination (or "recombiner") system is described in considerable detail in reference 2; consequently, only a brief description will be given here. Figure 11 shows a semischematic layout of the assembly. A centrifugal blower forces 110 l/min of sweep gas (primarily air) around the semiclosed system. At this circulation rate, the solution off-gas is immediately diluted a factor of ~9, which keeps the hydrogen concentration below the detonation limit at all points of the system. The diluted gas is passed first through a reflux condenser in the sphere stack, which removes most of the water vapor and acid vapor, as well as solution spray. Next, the gas goes through a stainless-steel-wool trap, which serves to remove entrained liquid and provides a large surface area for capture of fission fragments. After passing through the blower (blower pressure differential = 8 in. of H_2O), the gas enters a catalyst chamber containing platinized alumina pellets. Here, the H_2 and O_2 recombine, with sufficient evolution of heat to raise the exit gas temperature to ~200°C. The so-called "external" condenser reduces the gas temperature to ~20°C, which condenses the water vapor; the condensate then runs back into the sphere, with the result that there is essentially no net loss of water from the sphere.

What was originally believed to be the highest pressure point of



PRESSURE COUPLING TUBES	
LINE NO.	DESCRIPTION
1	LOWER BUBBLER TUBE
2	UPPER BUBBLER TUBE
3	FLOWMETER (HIGH)
4	FLOWMETER (LOW)
5	BLOWER SEAL
6	RECOMBINER INLET
7	AUXILIARY AIR INLET
8	RECOMBINER OUT (EXT. COND. INLET)
9	STACK CONDENSER IN (EXT. COND. OUT)
10	STACK PRESSURE
11	STACK PRESSURE
12	STACK CONDENSER OUT
13	SPARE (PLUGGED)
14	LEAK DETECTOR - COPPER TUBE

Figure 11. Schematic diagram of recombiner system.

the recombiner system, the blower exit, is connected through a needle valve to an exhaust line which has a static subatmospheric pressure of -3 in. H_2O. With this arrangement, it was assumed that any leak which developed in the recombiner system would be a leak <u>in</u> rather than <u>out</u>. There was a flaw in this argument, however, since for several years there has been a small leak of radioactive gas <u>out</u> of the system in the vicinity of the centrifugal blower. The leak originated soon after the blower was repaired in 1953. Presumably, there is a crack in the gasket seal, and the inertial pressure at the periphery of the blower housing is above atmospheric. The small amount of gas which escapes is collected with an exhaust arrangement which discharges the gas into the atmosphere above the reactor building. This leak has not been fixed because of the extremely high radiation levels (>500 r/hr) which would be encountered during the repair operation.

Excess gas which appears in the system, whether internally generated or externally injected, is automatically bled into the exhaust (or "stack") line through the needle valve mentioned above. The two main sources of excess gas are (1) the liberated oxides of nitrogen, from NO_3^- breakup, which do not recombine in the catalyst chamber, and (2) bleed air forced down the pressure-sensing tubes of the recombiner system to buck diffusion of radioactive gas up these lines to the pressure gauges. During 25 kw operation, about 220 cc/min of excess gas flows into the exhaust line. This line runs underground to the top of an adjacent mesa and terminates in a 150-ft-high stack. A large centrifugal blower near the base of this stack generates the (-3)-in. exhaust pressure mentioned above. It requires ~8 hours for the fission gas to travel from the reactor to the base of the stack; consequently, much of the radioactivity has already decayed before the gas is liberated to the atmosphere. It has been the opinion of the Laboratory Health Division that the liberation of this gas does not constitute a hazard to the nearest populated area.

The catalyst chamber has exhibited no detectable change in operating characteristics since its original installation. Presumably it will last indefinitely, even though its normal operating temperature (at 25 kw) is ~450°C.

Startup procedures specify that the reactor power shall not exceed 10 kw until the recombiner temperature has reached 100°C. This procedure preheats the catalyst bed and tests its operation before the higher oxy-hydrogen production rates characteristic of full power are encountered. After the reactor is turned off, the recombiner blower is operated for a full hour to ensure recombination of all radiolytic gas present in the system at the time of shutdown. At the end of the hour, the blower is turned off by a time-delay circuit.

When the reactor is operated at ≥30 kw, abnormal pressure fluctuations are observed in the recombiner-system pressure gauges. These pressure variations are believed to be caused by unstable burning of the gas in some region preceding the catalyst pellets. This is one of several reasons why the Water Boiler is normally operated at a maximum power of 25 kw (see Section VII).

V. Control Rods

The Water Boiler has four control rods and a safety rod. A scram signal from the control console will cause all five rods to drop (free fall) into the position of minimum reactivity. Rod drop times are of the order of 0.5 sec.

The safety rod and two of the control rods are basically cadmium sheets with dimensions 1/32 x 2-1/2 x 30 in., sandwiched between 1/32-in. sheets of aluminum. These rods operate in vertical slots in the graphite reflector, essentially tangent to the sphere. The other two control rods, called the boron rods, operate in re-entrant thimbles which penetrate the sphere volume. Each boron rod is basically a sintered B^{10} (95.5%) plug, 0.555 in. in diameter and 17.9 in. long, contained in a steel tube which is plated with cadmium, ~3 mils thick. The control rods are checked once a year for corrosion. To date, corrosion has been a minor problem.

The safety rod is motor-driven. At the push-button command of the operator, the drive motor moves the rod all the way out or all the way in. It takes ~12 sec for withdrawal of this rod. The operator cannot stop the rod in an intermediate position. Because of an electrical interlock, none of the other rods can be withdrawn until the safety rod is in its "out" position. This is the only sequencing interlock involved in the control of the five rods. However, the following conditions are necessary before the safety rod can be withdrawn:

1. Control-rod-position indicators set at 000.
2. Scram switch in the "up" position.
3. All scram relays unshorted (see Table VI, Section VI).
4. All (four) safety chassis scram channels untripped.

Each cadmium control rod is actuated by a pair of selsyns, one at the control console and the other at the rod gear box. The operator simply turns the console selsyn to move the control rod. The boron rods, on the other hand, are equipped with motor drives. An electronic automatic control (A.C.) system, provided for holding the reactor at a preselected power level, can be connected to either of the boron rod drives. During steady operation, the A.C. system is normally used to actuate the east boron rod. Some type of automatic control is absolutely essential for water-boiler-type reactors because of (1) instabilities associated with the high radiolytic gas evolution rate, and (2) the large temperature coefficient of reactivity.

The reactivity worth of each of the three cadmium rods is ~\$1, and the two boron rods are each worth ~\$3. Total rod control therefore amounts to ~\$9. Plots of rod worth versus position for the boron rods and west cadmium rod are shown in Figs. 12 and 13. Additional control-rod reactivity data are given in Section IX.

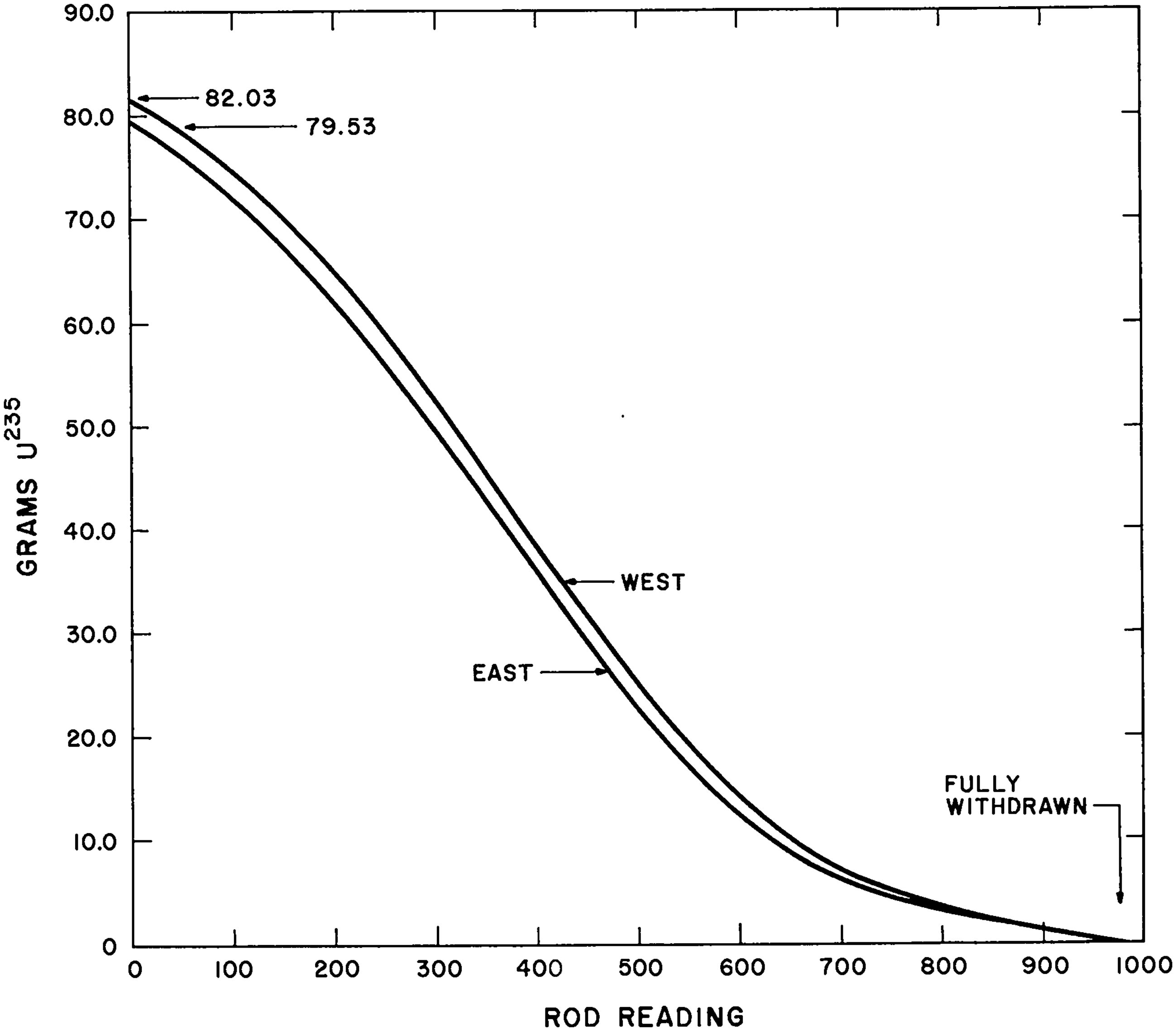


Figure 12. Reactivity worth of boron rods (gm U^{235} vs position); ~120 numbers = 1 in.

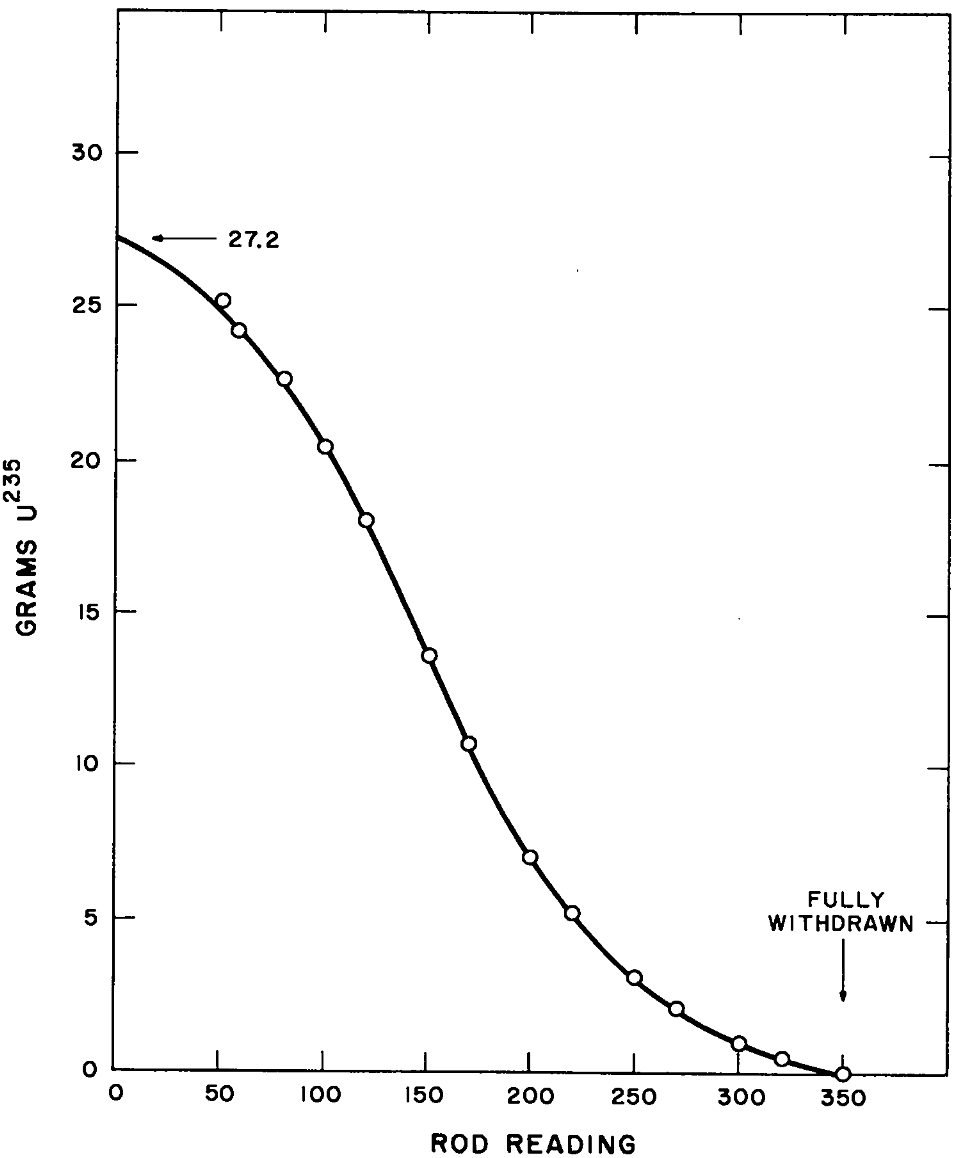


Figure 13. Reactivity worth of west cadmium rod (gm U^{235} vs position); ~15 numbers = 1 in.

VI. Instrumentation

A. General

A block diagram of the Water Boiler control instrumentation is shown in Fig. 14. Several of the indicated components have been described earlier in this report; others are described in the following paragraphs. Although it is not immediately apparent in the block diagram, a sizable fraction of the instrumentation is associated with the recombiner system.

B. Chamber Systems

There are six instrumentation chambers involved in the Water Boiler control system. They are all shown in the block diagram (Fig. 14). Pertinent facts about these chambers are given in Table V. A description of the purpose and functions of each chamber system is presented below.

1. TWEETER System
   a. Basic Purpose: Low-level neutron monitor.
   b. Specific Functions:
      1) Provides an audible signal (broadcast over loud-speakers in both the reactor room and control room) whose frequency is proportional to reactor power. This system gives one of the first indications of neutron multiplication. It is automatically turned off at ~1 kw.
      2) Provides a counting-rate-meter indication of power level in the operating range between source level and ~10 watts.
2. INTEGRATOR System
   a. Basic Purpose: High-level neutron monitor.
   b. Specific Functions:
      1) Provides scaler information from which the total integrated kilowatt-hours may be obtained for any period of reactor operation, regardless of arbitrary changes in power level.

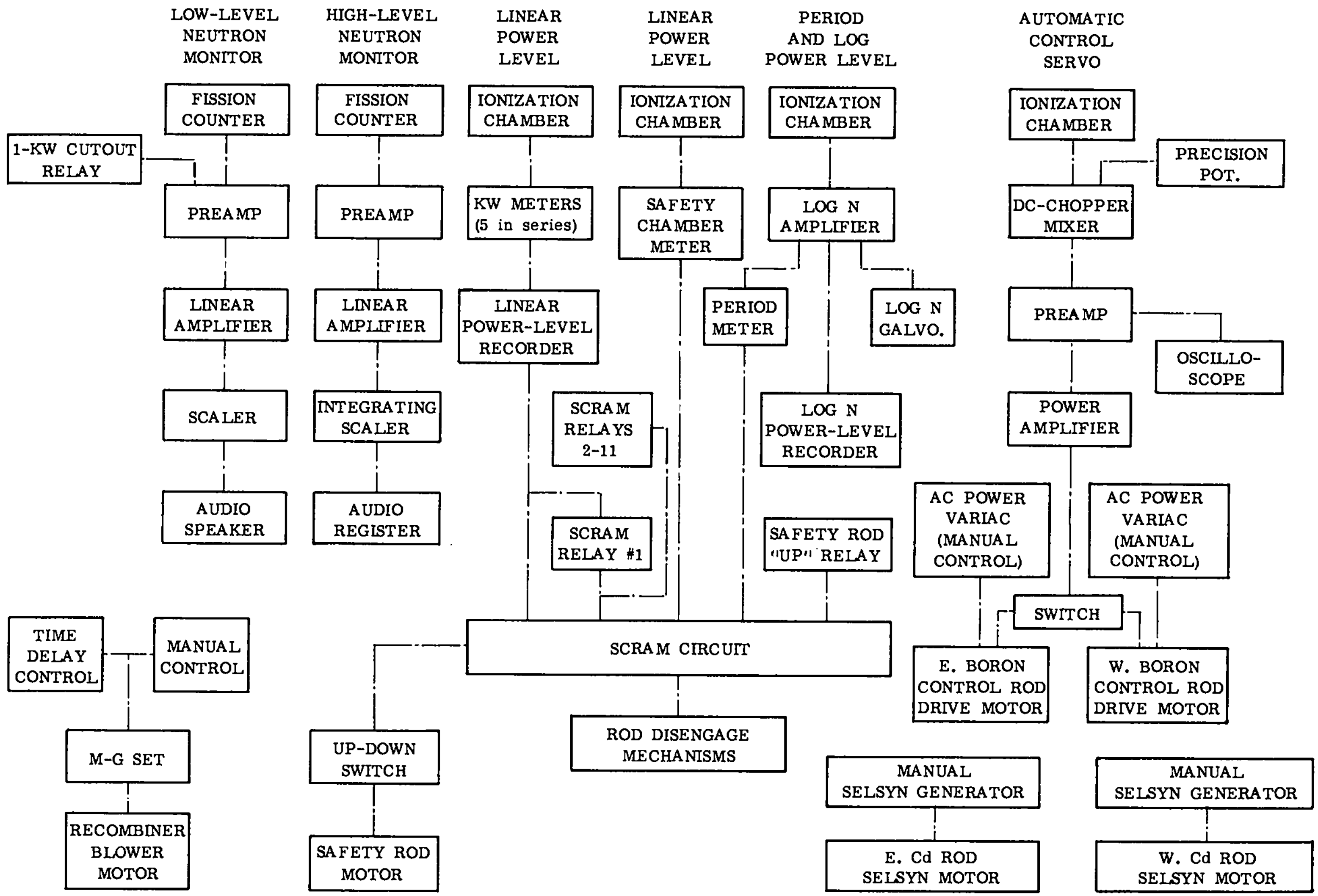


Figure 14. Block diagram of control instrumentation.

TABLE V

WATER BOILER CHAMBER SYSTEMS

Name	Type of Chamber	Usable Range[a]	Scram Point	Chamber Location
TWEETER	Fission (pulse)	Subcritical - 1 kw	None	1-in. SE Vertical Port
INTEGRATOR	Fission (pulse)	~1 kw - 30 kw	None	4-in. NW Vertical Port
LOG N	Boron-lined ionization (compensated)	Power: subcritical - 30 kw Rate: 100-7 sec/decade	None 7 sec/decade	4-in. NE Vertical Port
AUTOMATIC CONTROL	Boron-lined ionization (uncompensated)	~0.1 watt - 30 kw	None	North Thermal Column (Graphite Layer #2)
KILOWATT	Boron-lined ionization (uncompensated)	~0.5 kw - 30 kw	30 kw and 35 kw (Two channels)	North Thermal Column (Graphite Layer #2)
SAFETY	Boron-lined ionization (uncompensated)	~1 kw - 30 kw	30 kw	4-in. SE Vertical Port

[a]Upper limits of 30 kw are indicative only of maximum power at which reactor can be operated without tripping a power-level scram. All systems except the TWEETER would be operative up to a power level of 50 kw.

2) Provides the reactor operator with an audible signal whose frequency is proportional to reactor power (useful only for power levels >1 kw).
3) Provides a highly accurate measure, through scaler counting rate, of relative power level (useful in the power region >100 watts).

3. LOG N System
   a. Basic Purpose: Combination power-level and period monitor, operative over the entire reactor power-level range.
   b. Specific Functions:
      1) Provides the operator with a visual (meter) indication of reactor period.
      2) Provides a period scram (Primary Scram Channel I).
      3) Provides the operator with a visual indication of reactor power level throughout the entire operating range from source level to maximum power. Log N is displayed on a 7-decade galvanometer scale, 14 in. long. This device usually gives the operator his first indication of source multiplication.
      4) Provides a record of Log N as a function of time. The Log N recorder is normally in operation 24 hours a day, 7 days a week.

4. AUTOMATIC CONTROL System
   a. Basic Purpose: Generation of electronic driving signal for servo system that is used to keep reactor power level at a constant value.
   b. Specific Functions:
      1) Provides electronic feedback signal mentioned in (a). A by-product of this feature is that the operator is provided with a simple method of inducing large-scale movement of the two boron rods.
      2) Provides a method for accurate establishment of a desired

power level. The accuracy of this system decreases with power level, since the A.C. chamber is uncompensated.

3) Provides the operator with an oscilloscope null indication when a preset power level has been reached.

5. KILOWATT CHAMBER System

a. Basic Purpose: Linear power-level indicator.

b. Specific Functions:

1) Provides a visual, linear power-level (meter) reading at the control panel and on all four reactor faces (useful for power levels $\gtrsim$ 0.5 kw).

2) Provides two power-level scrams (Primary Scram Channel II, and a shorting-relay scram).

3) Provides a record of linear power level versus time for power levels in excess of $\sim$0.5 kw.

4) Provides the signal that turns off the TWEETER system at $\sim$1 kw.

6. SAFETY CHAMBER System

a. Basic Purpose: Linear power-level scram.

b. Specific Functions:

1) Provides a power-level scram (Primary Scram Channel IV).

2) Provides a visual, linear power-level (meter) reading at the control panel (useful for power levels $\gtrsim$ 1 kw).

C. Scram System

The nerve center of the scram system is the Safety Chassis (labeled "SCRAM CIRCUIT" in Fig. 14). There are five Schmidt-trigger "flip-flop" circuits in the Safety Chassis, controlled by fourteen separate scram signals (see Table VI). When one of the scram signals reaches its trip point, the associated flip-flop circuit "flips," causing all rods to disengage and drop to their position of minimum reactivity.

Operation of the reactor with any scram bypassed, inoperative, or with the scram trip point set differently from that given in Table VI

TABLE VI

SCRAM CONDITIONS

Scram Channel	Trip Point or Trip Condition	Signal Source
I. Period	7 sec/decade. (Also tripped if Log N amplifier "CALIBRATE-USE" switch is in the "CALIBRATE" position.)	Differentiated signal from Log N amplifier.
II. Linear power level (kw)	35 kw	DC current from kilowatt ionization chamber.
III. Spare		
IV. Linear power level	30 kw	DC current from safety chamber.
V. Shorting relays (all are sensitrol relays except #10 and #11)		
1. Linear power level	35 kw	DC current from kilowatt chamber.
2. Sphere temperature	$\geq 85^{\circ}C$	Thermocouple current.
3. Stack blower motor	±50% change in normal motor current	AC current to one phase of motor.
4. Kilowatt and safety chamber high voltage	Deviation of ±50 volts from normal 800 volts	DC current through H.V. bleeder resistor.
5. Automatic control chamber high voltage	Deviation of ±50 volts from normal 800 volts	DC current through H.V. bleeder resistor.
6. Liquid level	Rise of soup level to $\sim$1 in. above upper bubbler tube.	Upper Bubbler Magnehelic pressure transducer.
7. Recombiner flow	Drop of pressure differential across orifice flow meter to $\sim$0.1 in. H_2O	Magnehelic pressure transducer.
8. System-to-stack pressure	Rise of highest pressure point of recombiner system to 1 in. H_2O above stack pressure	Magnehelic pressure transducer.
9. Stack vacuum	Rise of stack pressure to -1 in. H_2O (normal reading: -3 in. H_2O)	Magnehelic pressure transducer.
10. Air-out-of-sphere temperature	$\geq 31^{\circ}C$	Thermocouple-controlled relay (connected to μ-switch in 12-pt temperature recorder).
11. Sphere water flow	Sphere cooling water must be turned on	Relay controlled by sphere-flow toggle switch.

requires the approval of the Water Boiler Committee. With one exception, scram channels can be bypassed or rendered inoperative only by rewiring or disconnecting appropriate cables. The one exception is the Period scram, which can be bypassed with a spring-loaded push-button switch. It is necessary for the supervisor to use this bypass switch during power calibration of the Log N system; otherwise, electrical transients introduced during this procedure would usually trip the Period scram.

All of the scram circuits are checked periodically. The trip points of the Period, Kilowatt, and Safety scram channels are checked daily. Also, the Recombiner-Flow scram is tested each day by turning off the recombiner blower motor. All sensitrol relays are tested once a week by tripping them with a magnet and checking that they activate the associated Safety Chassis relays. At the end of each quarter year, every scram circuit is thoroughly tested, either directly or indirectly, to insure that it is operating as specified in Table VI.

VII. Operating Characteristics

A. Normal Operation

The Water Boiler can be operated at any steady-state power between $\sim$0.1 watt and 25 kw. The lower limit is imposed by the automatic control system, which starts to be nonlinear at power levels $<$0.1 watt. The nominal upper limit of 25 kw is imposed by conditions outlined in the next subsection.

It takes a minimum time of $\sim$5 min to bring the reactor up to power levels $>$5 kw from subcritical (all rods down). After a scram, the reactor power level essentially follows the decay of the delayed neutrons; the power drops a factor of $\sim$25 in the first 5 sec, and after 1 min the power level is down a factor of $\sim$200. The rapidity with which the reactor can be brought to full power or dropped down to essentially zero power from full power is often a considerable advantage in the design and performance of reactor experiments. The reactor has been brought to power and scrammed as many as 50 times in a single 8-hr period.

Certain operational data are recorded in a log book every two hours during steady operation at any given power level. Typical operating conditions at 25 kw are shown in Table VII. Calibration of the reactor power-level scale is normally derived from thermal data obtained at operational levels of ≥10 kw. The formula used to calculate the thermal power is

$$\text{Thermal Power} = C_1 C_2(\Delta T_s \cdot F_s + \Delta T_c \cdot F_c)$$

where

ΔT_s = (OUT-IN) temperature rise of sphere cooling water ($^\circ$C)
ΔT_c = (OUT-IN) temperature rise of recombiner condenser cooling water ($^\circ$C)
F_s = sphere cooling water flow (gal/min)
F_c = condenser cooling water flow (gal/min)
C_1 = 0.264 = conversion factor: kw/($^\circ$C x gal/min)
C_2 = 1.06, an empirical constant which mainly corrects for conduction heat loss to the reflector

B. Operation at Power Levels >25 kw

Soon after SUPO was completed, trial runs were made at power levels as high as 45 kw, and during most of 1951 and 1952, the reactor was frequently operated at 35 kw for sustained periods of time. However, experience gained during these early years finally led to the decision (September 1953) to limit the power to a nominal 25 kw. The main reasons for this decision follow:

1. At >30 kw, rather large pressure fluctuations are observed at various points around the recombiner-system gas loop. These fluctuations may be associated with unstable burning of hydrogen in the region preceding the catalyst chamber. Consequently, limitation of the reactor power to <30 kw is a safety measure designed to reduce the possibility of a

TABLE VII

TYPICAL EQUILIBRIUM OPERATING CONDITIONS AT 25 kw (December 1961)

East boron control rod position	525
Sphere H_2O flow	3.43 gal/min
Condenser H_2O flow	0.20 gal/min
Integrating counter	112.8×10^3 cts/min
Temperatures (^oC):	
Sphere solution	60.8
Sphere water (In)	5.0
Sphere water (Out)	30.8
Reflux condenser water (In)	3.5
Reflux condenser water (Out)	10.8
External condenser water (Out)	15.7
Air into sphere	26.1
Air out of sphere	11.5
Air into recombiner	34.2
B_4C shield	21.8
Blower bearing	34.0
Steel (east side)	22.1
Recombiner bed (bottom)	434
Recombiner bed (middle)	405
Recombiner bed (top)	366
Air out of recombiner	180
Catalyst pot	122
Pressures (in. H_2O):	
Catalyst chamber Δp	3.1
Reflux condenser Δp	5.7
Orifice flow meter Δp	0.61
Stack vacuum	3.1
System-to-stack Δp	0.18
Bleed-in air	5.2
Sphere inlet water pressure	80 psi
Bleed-in air flow	130 cc/min
Blower-generator frequency	85 cycles/sec
Blower-generator power	160 watts

H_2-O_2 explosion.

2. At ~35 kw, power-level transients start to appear which are either too large or too rapid to be completely cancelled by operation of the automatic control system. The precise cause of these transients is not known. Some of them are definitely correlated with the phenomenon discussed above in (1). However, there is basis for postulating that many of the transients are caused by sporadic local boiling of the core solution in the spaces between cooling-coil loops.

3. Sustained operation of the reactor at power levels ≥30 kw sufficiently raises the temperature of the edge of the reflector region to cause boron-loaded paraffin to melt and slowly ooze into port 4W.

C. True Boiling Mode

During certain studies of reactor behavior, SUPO has been operated at boiling temperature (~93°C). To achieve this condition, the cooling water is turned off during operation at a power level in the range 10-25 kw. As the sphere temperature rises, the control rod is steadily withdrawn by the automatic control because of the large negative temperature coefficient of reactivity. When the boiling temperature is reached, a large increase in void volume occurs because of steam bubble formation. This causes the reactor power to decrease to a quasi-equilibrium value of 5 or 6 kw, regardless of further control rod withdrawal. In fact, all control rods can be completely withdrawn without affecting the reactor behavior; the automatic control system is therefore completely useless under these conditions. During the boiling mode of operation, the power level is almost continuously varying, but the magnitude of the fluctuations is only 10-15%.

VIII. Experimental Facilities

There are over thirty different ports in the Water Boiler where samples may be irradiated. In addition, external neutron beams can be established at many of these ports.

Small samples requiring high flux are usually irradiated in one of the ports which enter the graphite reflector region. These ports run in an east-west direction and either penetrate, are tangent to, or end adjacent to the sphere. Larger samples or samples that require an almost pure thermal flux are irradiated in the thermal columns. A brief description of the more frequently used sample facilities follows.

A. Graphite Reflector Ports

1. Glory Hole (Ports 3E and 3W)

Samples may be placed in a 1-in.-I.D. tube, approximately 10-3/4 in. long, which passes through the sphere in an east-west direction, 2-1/8 in. below and 1-3/8 in. south of the sphere center. This is the highest flux position available in the reactor ($\sim 10^{12}$ n/cm^2 sec at 25 kw). The Cd ratio, measured with 5-mil indium foils, is ~ 3 (see Fig. 15).

2. Ports 4E and 4W

Ports 4E and 4W terminate, respectively, 6 in. east and 6 in. west of the reactor north-south centerline. The centerline of these "collinear" ports is located 2-1/8 in. below (in the same horizontal plane as the Glory Hole) and 5-5/8 in. south of the sphere center. The normal sample positions in each of the ports are two 1-3/4-in.-I.D. x 4-in.-deep cylindrical holes in the graphite stringer. The vertical axis of both holes is 10 in. from the sphere vertical centerline. These positions are the ones most frequently used for sample irradiations. The neutron flux at 25 kw is $\sim 6 \times 10^{11}$ n/cm^2 sec. If required, the inner removable 4-1/4-in. x 4-1/4-in. graphite stringer may be modified to accommodate larger samples.

3. Ports 1E and 1W

Ports 1E and 1W terminate, respectively, 4-5/8 in. east and 4-5/8 in. west of the north-south centerline. The centerline of these collinear ports is located 6-3/8 in. above and 7/16 in. south of the sphere center. The normal sample positions in each of the ports are two 1-3/4-in.-I.D. x 4-in.-deep cylindrical holes in the graphite stringer. The width of the inner graphite stringer for each port is 2-1/8 in. The two sample holes

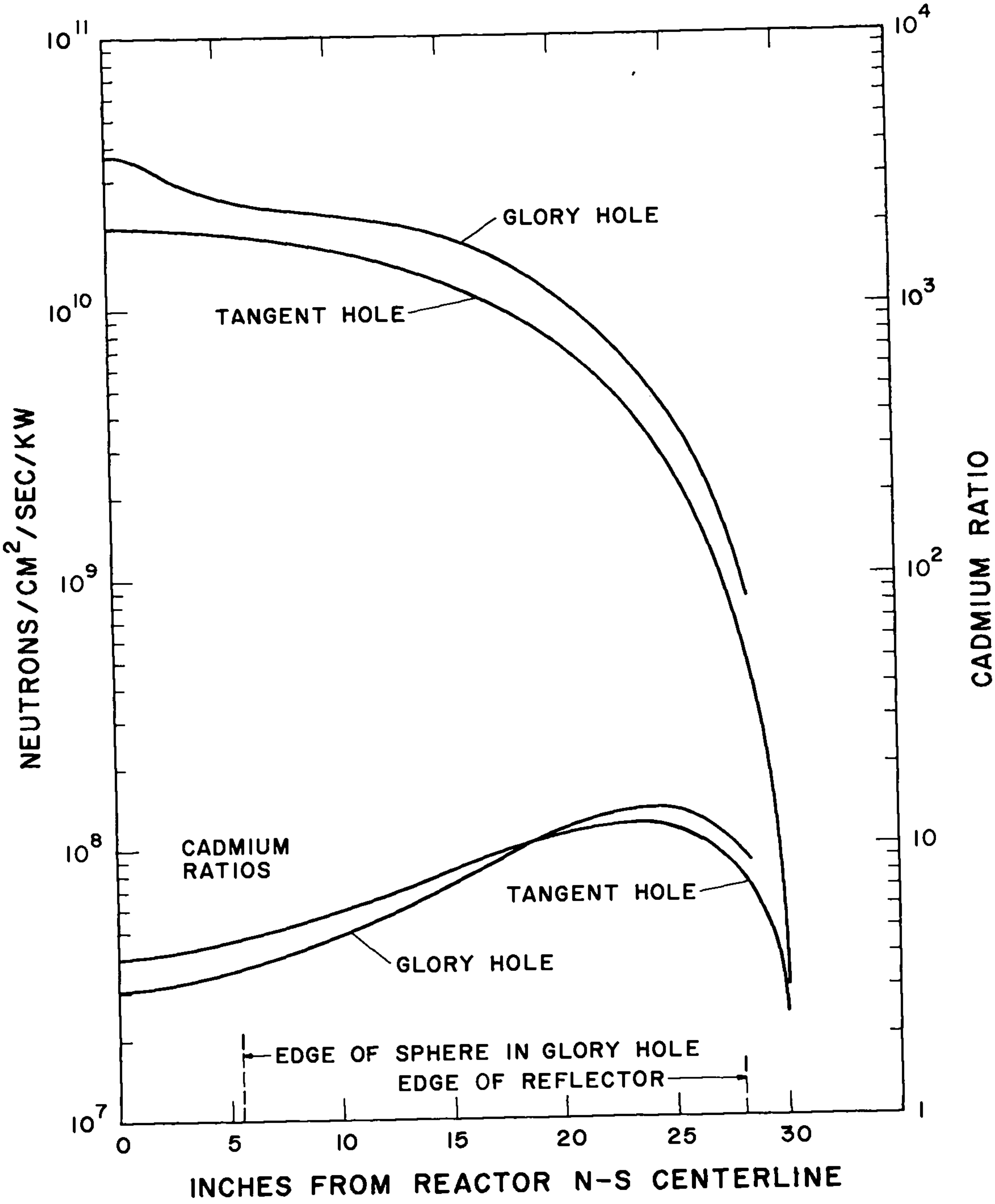


Figure 15. Flux and Cd ratio in Glory Hole and Tangent Hole.

in each stringer are in line and are located about 6 in. and 8 in., respectively, from the sphere vertical centerline. The neutron flux in the inner sample hole is $\sim 4 \times 10^{11}$ n/cm^2 sec at 25 kw.

4. Tangent Hole (Ports 2E and 2W)

Samples may be placed in a 1.4-in.-I.D. tube, approximately 10 in. long, which is almost tangent to the sphere surface. The centerline of the port is 6-3/8 in. above and 2-9/16 in. south of the sphere center. At 25 kw, the flux is $\sim 5 \times 10^{11}$ n/cm^2 sec (see Fig. 15). Currently, a facility for pneumatically inserting samples is mounted in the west half of this port (2W). Samples are contained in nylon or aluminum tubes (rabbits) which will accommodate cylindrical samples about 1.7 in. long and 0.35 in. O.D. Rabbit-tube extensions are available which allow rapid delivery of the sample to the chemistry building or to the nuclear spectroscopy laboratory.

5. Vertical Ports

There are six vertical cylindrical ports; two are of 1 in. diameter and four are of 4 in. diameter. They are designated as 4-in.-SW, 1-in.-SW, 4-in.-SE, 1-in.-SE, 4-in.-NW, and 4-in.-NE. The four "S" ports penetrate the reflector $\sim$6 in. north of the south bismuth wall; the two "N" ports penetrate the North Thermal Column $\sim$2 in. north of the first 4-1/4-in. layer of the north bismuth wall. The vertical centerlines of the 4-in. ports are $\sim$25 in. from the sphere vertical centerline, whereas the 1-in. ports are located $\sim$20 in. from the sphere vertical centerline. At present (1962), the two "SW" ports are the only ones available for experiments; the other four ports are occupied with instrumentation chambers (see Table V).

B. Thermal Columns

1. North Thermal Column

The North Thermal Column is a region of high purity graphite approximately 42 in. wide and 63 in. high, extending 4 ft north of the graphite reflector region. The thermal column is separated from the reflector region by a bismuth wall 8-1/2 in. thick. Access to the column is normally gained by rolling back the 26-in.-thick main shielding door. There are other smaller doors which may be opened if it is desired to bring out neutron beams or electrical leads from experiments. A cavity 4 ft long by 12-3/4 in. wide by 17 in. high can be produced by removing the twelve central 4-1/4-in. x 4-1/4-in. x 4-ft graphite stringers. Smaller cavities can be made with combinations of available graphite stringers. The flux distribution in the central port is shown in Fig. 16.

2. South Thermal Column

The South Thermal Column is a region of high purity graphite, approximately 60 in. wide by 63 in. high which extends 5 ft south of the graphite reflector region. This column is separated from the reflector region by a bismuth wall, 8 in. thick. A neutron-absorbing curtain made of Boral, which can be quickly raised and lowered, is installed 1 ft south of the bismuth wall. This curtain will prevent thermal neutrons from entering the remaining 4 ft of thermal column. Access to the South Thermal Column is gained through thirty ports (see Fig. 4). A cavity as large as 12-3/4 in. wide by 12-3/4 in. high by 5 ft long is obtainable. Since the position of the thermal columns is not symmetric with respect to the sphere, the maximum flux in the south column (0.9×10^{11} n/cm^2 sec) is not as high as in the north column (2.3×10^{11} n/cm^2 sec). The flux distribution in the central port of the South Thermal Column is shown in Fig. 16.

IX. Hazards Evaluation

A. Reactivity Behavior of the Core Solution

The Water Boiler solution has a large negative temperature coefficient of reactivity, with the result that transient power excursions tend to be self-correcting. The fuel loading required for reactor criticality as a function of core temperature, at various steady-state powers, is shown in Fig. 17. These data give an average temperature coefficient of reactivity in the region between 30° and 60°C of $\sim$0.72 gm U^{235}/°C.

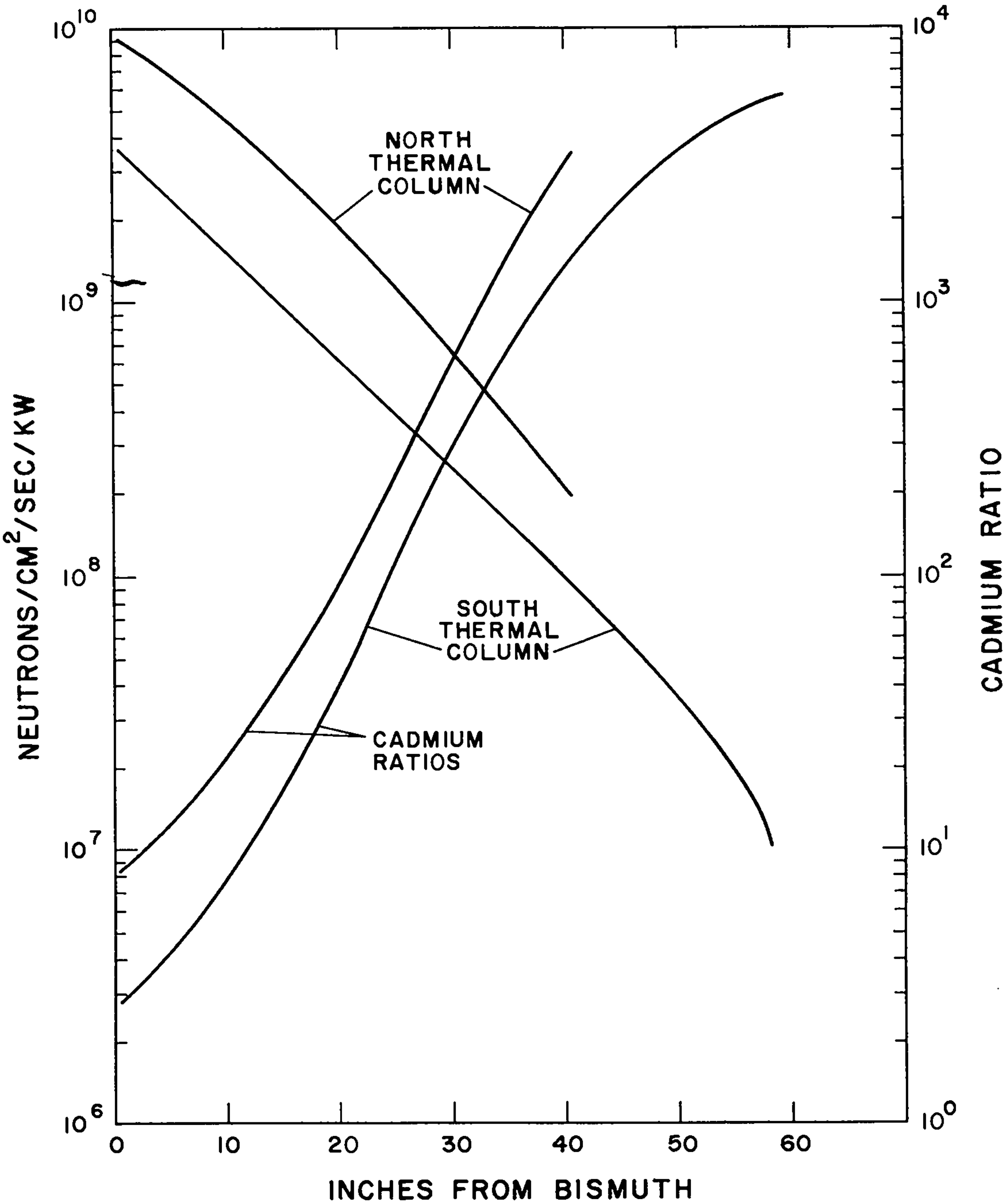


Figure 16. Flux and Cd ratio in the central ports of the thermal columns.

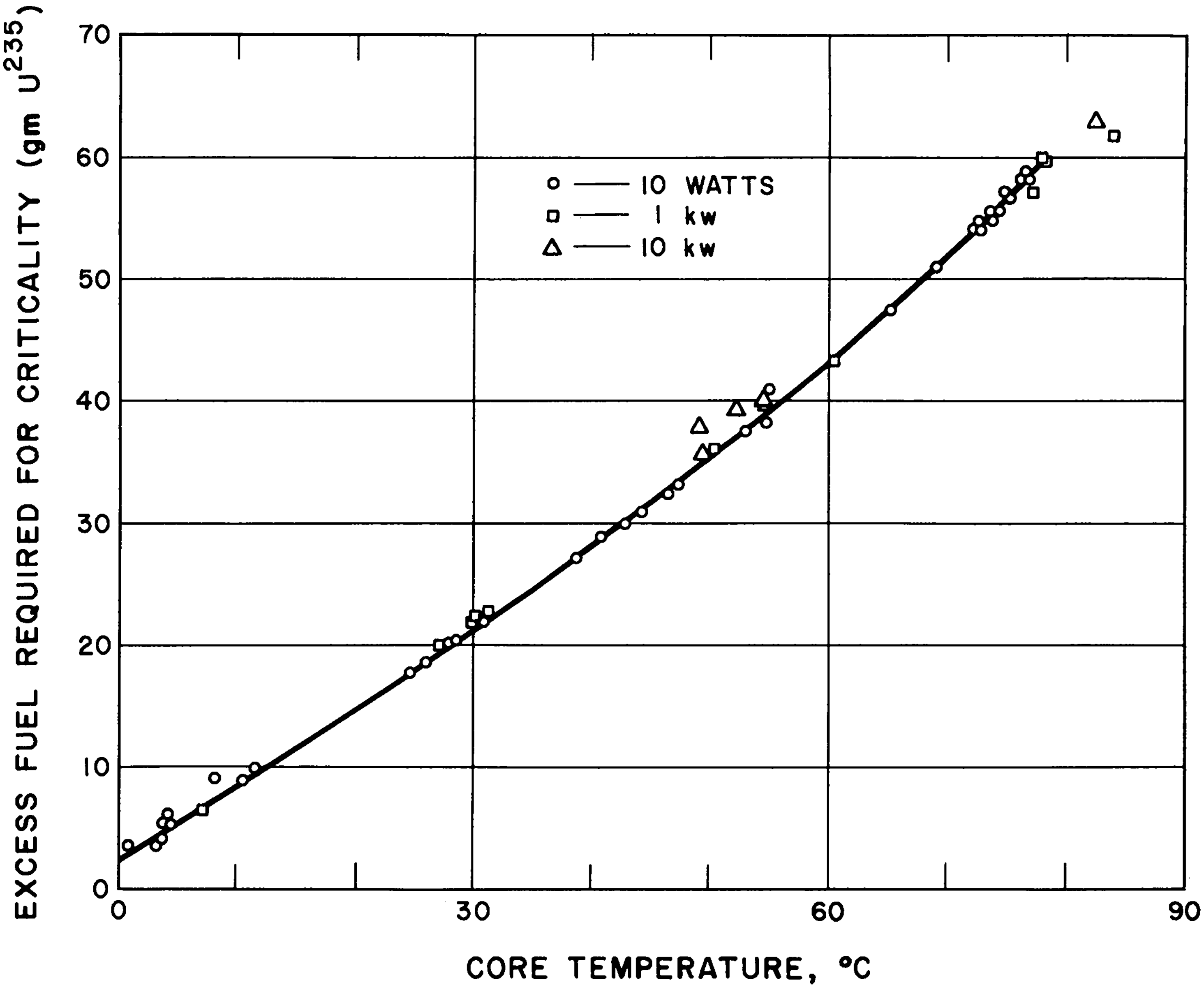


Figure 17. Temperature coefficient of reactivity (SUPO).

The decomposition gases also have a strong influence upon reactor control. Their effect is determined by the instantaneous rate of gas formation combined with the rate at which the decomposition gases leave the reactor core. Kasten[5] made a study of the response of the Water Boiler to sudden reactivity changes, and was able to separate the effects of core temperature rise and decomposition-gas formation upon reactor behavior. In his experiments, about 0.4% Δk was added to the reactor in $\sim$0.1 sec. Initial power levels used were 1 kw and 10 kw, and the initial sphere temperature ranged from 30° to $82^{\circ}C$. The minimum reactor periods observed were of the order of 4 sec. His most significant observation was that immediately following a reactivity addition, reactivity decrease was due primarily to decomposition-gas formation, at least 10% of the potential gases being evolved within 0.2 sec. In all cases the initial rate of reactivity decrease was estimated to be five or six times the rate which could be attributed to core-temperature rise. Kasten concluded that at power levels $\geq$1 kw, the gas formation effect contributes much more to the intrinsic safety of SUPO than does the negative temperature coefficient.

B. The KEWB Experiments

A group of reactor experiments which have direct bearing on the inherent safety of SUPO have been performed by Atomics International Division of North American Aviation, Inc.[6-13] These experiments were carried out under an AEC Reactor Safety program entitled "Kinetic Experiments on Water Boilers" (abbreviated KEWB). The program was initiated in 1954 and is still in progress. Its main objective has been to establish a firm basis for the evaluation of the safety of aqueous homogeneous reactor designs with regard to an accidental addition of large quantities of reactivity. The core design used in the first series of KEWB experiments (1956-1959) was patterned very closely after the Los Alamos design (see Table VIII). Consequently, the results obtained should be applicable to SUPO.

In the KEWB experiments, reactivity additions up to \$5 were introduced, both in step and ramp inputs, without damaging the core. Step

TABLE VIII

COMPARISON OF CHARACTERISTICS OF KEWB SPHERICAL-CORE REACTOR AND SUPO

	Items	KEWB (Full Core)	SUPO
1.	Sphere diameter, inside (in.)	12.3	11.9
2.	Nominal thickness of core vessel (in.)	1/4	1/16
3.	Chemical form of fuel solution	UO_2SO_4	$UO_2(NO_3)_2$
4.	Mass of U^{235} in core for critical (gm)	1240	777
5.	Mass of U^{235} for power operation (gm)	1450	950
6.	Fuel enrichment (%)	93.2	88.8
7.	Volume of fuel solution (liters)	13.7	12.7
8.	Fuel concentration (gm U^{235}/l)	106	75
9.	Total length of 1/4-in.-O.D. cooling coils (ft)	90	60
10.	Mass of stainless steel in cooling coils (gm)	2880	1996
11.	Void volume in control-rod thimbles and horizontal through-tube (liter)	0.93	0.30
12.	Mass coefficient of reactivity (\$/gm U^{235})	0.016	0.037
13.	Temperature coefficient of reactivity at $30^{\circ}C$ (\$/$^{\circ}C$)	-0.041	-0.025
14.	H/U^{235} ratio	260	350
15.	Gas production coefficient (stoichiometric H_2-O_2 STP l/kwh)	18.1	20.2
16.	Sweep-gas flow rate (l/min)	198	110
17.	Control rod worth (\$)	9.16	9.1
18.	Graphite reflector dimensions	56-in. cube	55-in. cube

inputs were added in ~20 msec; ramp inputs were introduced at a maximum rate of $0.16/sec. Among the parameters studied were (1) total energy release, (2) maximum period, (3) peak power, (4) maximum inertial and expansion pressures, and (5) fuel temperature rise. In addition, the effectiveness of void formation and temperature rise as shutdown mechanisms was studied in the entire region of reactor periods down to 2 msec. The results of this latter study are summarized in Fig. 18. The behavior exhibited in Fig. 18 was found to depend very little on the initial temperature or power level of the reactor. It is noteworthy that for a period of ~4 sec, the KEWB data indicate that void formation and core temperature rise contribute nearly equal amounts of reactivity compensation, a result which deviates considerably from Kasten's[5] analysis of SUPO data. Although this difference is not understood in complete detail, it seems probable that the combination of the gas and temperature shutdown phenomena at SUPO would limit reactor power excursions to at least as low a level as indicated by the KEWB results.

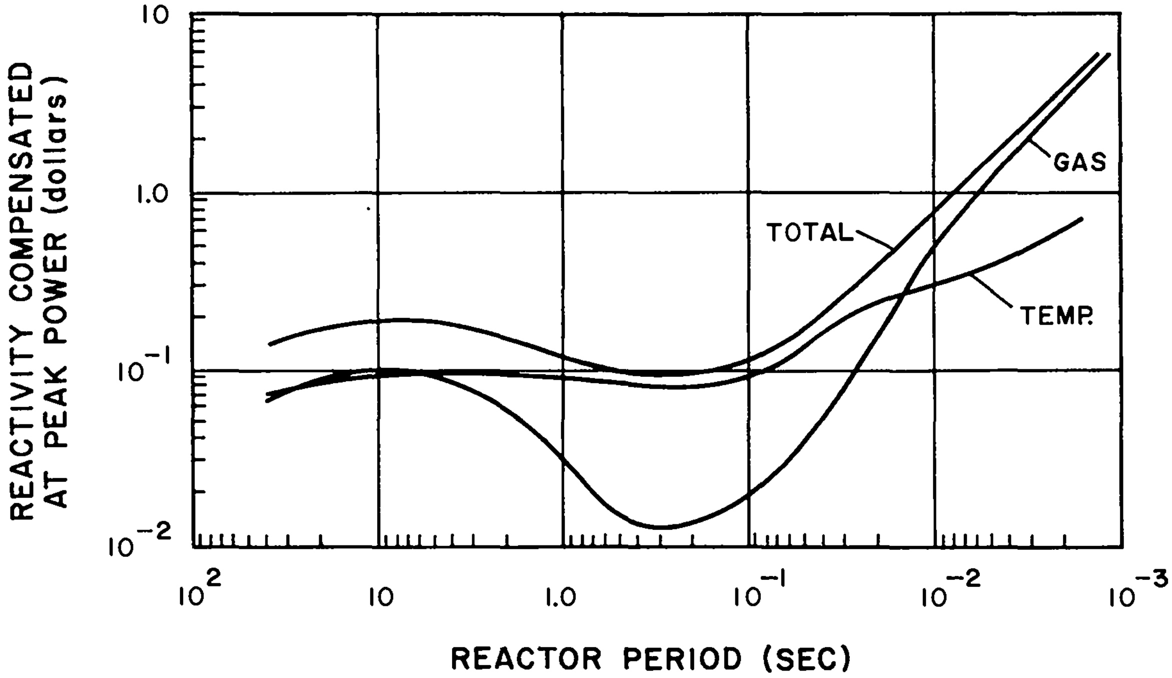


Figure 18. Compensated reactivity at time of power peak vs reactor period (KEWB). Adapted from reference 12.

The KEWB data indicate that the magnitude of a water boiler excursion, measured in terms of pressure generated and energy released, depends primarily on the amount and rate of reactivity addition, and is relatively insensitive to variation of any of the initial system conditions, such as fuel temperature, core pressure, reactor power level, and void volume. Step-input transients, which can be thought of as infinite ramp rates, always provide the maximum burst for a given total reactivity release. Thus, the step-input data can be used to estimate a reasonable upper limit on the magnitude of the excursion that would result from addition of a specified amount of reactivity.

Some of the most significant step-input results, from a safety point of view, are shown in Figs. 19-21. In regard to Fig. 19, it is assumed that the SUPO curve of reactivity versus reciprocal period would fall somewhere between the two curves shown, since the normal solution level corresponds to a 93% filling of the sphere. The peak power and peak pressures associated with a given reactor period are shown in Fig. 20, and the total energy release is shown in Fig. 21. It is noteworthy that the peak powers observed over most of the period range investigated are 50-100 times less than those detected in heterogeneous reactors of the pool type, a direct consequence of the inherent shutdown mechanisms of the aqueous homogeneous reactor.

The maximum expansion pressure observed in any of the KEWB runs was 620 psi, recorded at the bottom of the sphere after a $5 insertion (100% full core). The maximum impact pressure detected was 680 psi at the top of the sphere (85% full core).† These maximum pressures are comparable to the so-called "design" pressure of the SUPO sphere, based on conventional strength-of-materials calculations. These calculations indicate that the static internal pressure required to produce yielding is ~620 psi, and the static pressure required to produce rupture is ~1500 psi. However, from studies of the deformation of Type-347

†This impact pressure pulse is believed to be caused by the surface of the fuel solution striking the pressure transducer.

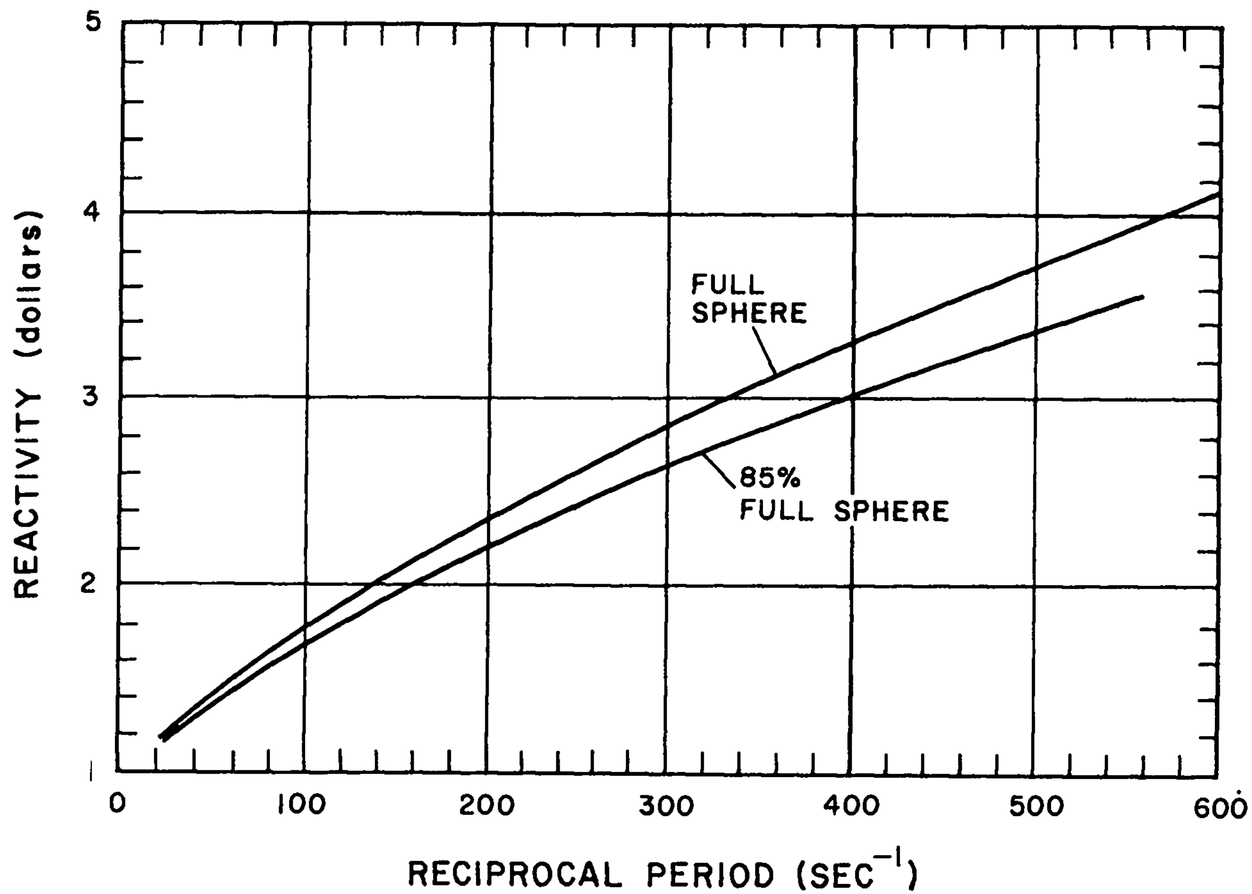


Figure 19. Reciprocal period vs reactivity insertion (KEWB). Adapted from reference 9.

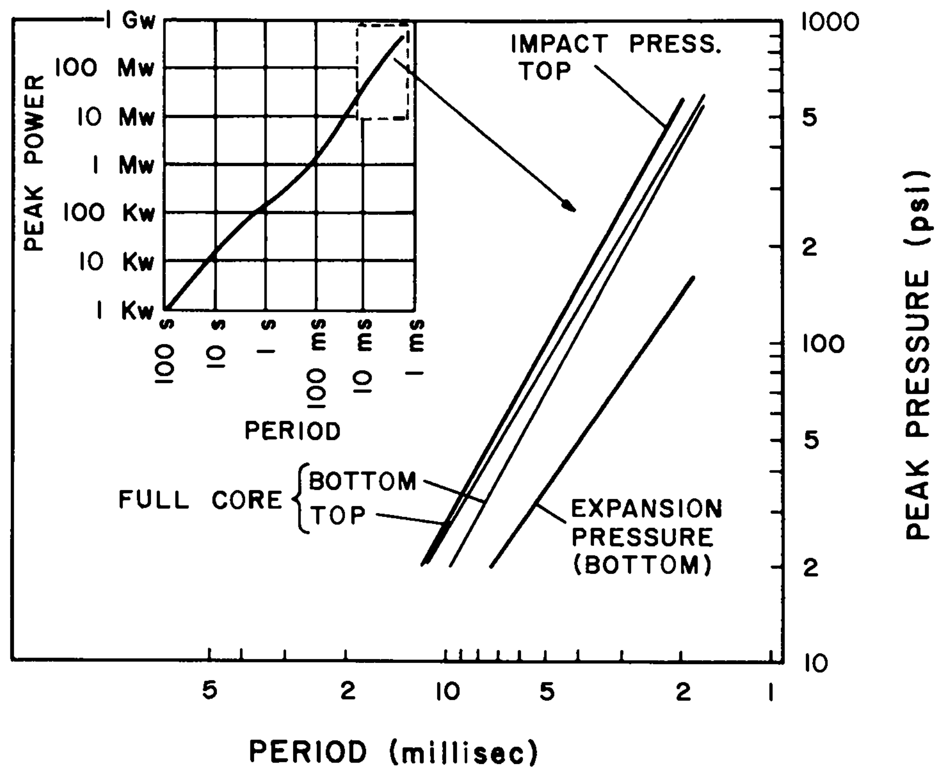


Figure 20. Peak power and peak pressures vs reactor period (KEWB). Adapted from reference 8.

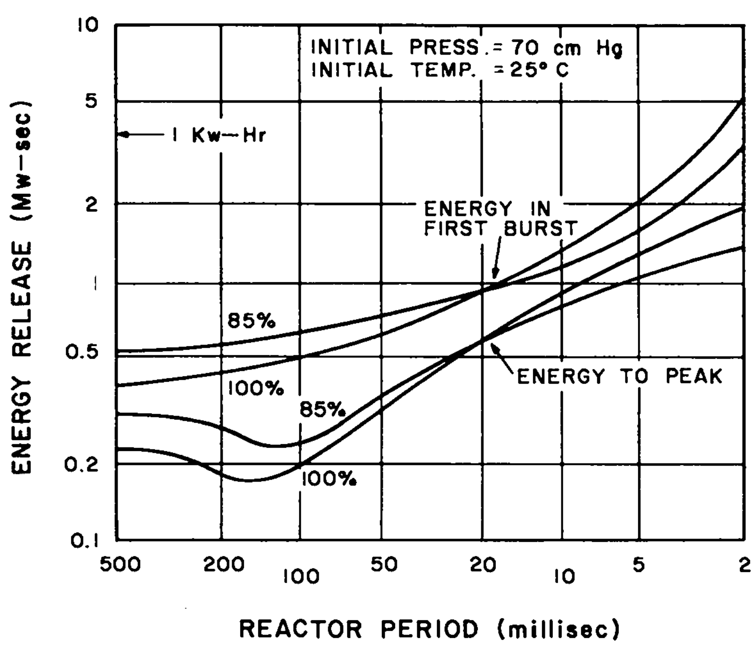


Figure 21. Total energy release vs reactor period, for both the 85% and 100% full sphere (KEWB). Adapted from reference 8.

stainless steel diaphragms under <u>dynamic</u> loading from H_2-O_2 explosions, the KEWB investigators estimate that the peak pressure required to produce rupture of a SUPO-type sphere may be >3000 psi.[11] In view of these results and the fact that the actual sphere wall is rigidly supported by the reflector graphite, it appears highly improbable that damage could result to the SUPO core vessel from either a nuclear excursion or a H_2-O_2 explosion (which can produce peak pressures[8] up to 450 psi).

C. SUPO Reactivity Data and Excursion Possibilities

With the present fuel loading, the maximum excess reactivity of SUPO is ~\$2.7. This value is based on a reactivity equivalent of \$1 = 27 gm U^{235}† and on the following assumed conditions: (1) sphere temperature of 4°C, (2) 1000 kwh after a normal acid addition, and (3) a solution volume of 12,620 cc. At a steady power level of 25 kw, excess reactivity is only ~80¢.

The reactivity worth of the control rods is as follows:

	Grams U^{235}	%ΔK/K	Dollars
Safety rod	33.2	1.0	1.2
E. boron	79.5	2.4	2.9
W. boron	81.7	2.5	3.0
E. cadmium	26.5	0.8	1.0
W. cadmium	27.2	0.8	1.0
		Total	9.1

According to the above data, the minimum amount by which the present reactor is subcritical when all control rods are down is (9.1 - 2.7) = \$6.4. It is evident that the shut down reactor cannot be made critical except through the withdrawal of more than two rods.

†Reactivity values for water boiler reactors are commonly expressed in the units "grams of U^{235}." This terminology expresses the equivalence of a given reactivity change to the addition of N gm U^{235} to the fuel solution.

There are several ways to add significant amounts of reactivity to SUPO. These are described in the following paragraphs, and the hazard associated with each operation is assessed.

1. Movement or Removal of Control Rods by Hand from the Reactor Top

In this case, the most hazardous situation exists during control rod maintenance or inspection. It is standard procedure to have only one rod completely out of its scabbard at any given time. If a rod is removed, a safety bar is immediately locked in place which allows the lifting of only one additional rod at a time. If for some reason the safety bar were ignored, it is conceivable that someone might lift two rods simultaneously. It would appear that the maximum reactivity change would be introduced if one of the cadmium rods was out of the reactor, and both boron rods were then quickly raised. This action would make the present assembly supercritical by a maximum of ~70¢, and no significant excursion would occur.

At some future date, the \$2.7 maximum excess reactivity may be raised, either through addition of U^{235} to the soup or through positive-reactivity changes external to the sphere. If such changes are restricted to give a maximum excess reactivity of \$4, as has been proposed, then the type of accident described above would never involve a reactivity input of more than \$2. The KEWB data indicate that a \$2 step input would put the reactor on a 7 msec period and would produce a maximum inertial pressure of ~60 psi. The SUPO core vessel should easily withstand such an excursion. Soup would undoubtedly be regurgitated up into various sections of the recombiner system, but this alone would not put the reactor out of operation. The principal danger associated with a power transient of this magnitude is the possibility of a H_2-O_2 explosion. There would probably be no explosion if the rods were dropped immediately after the power excursion or if the recombiner system were in operation. It is the KEWB experience that during the power burst, the large quantities of H_2-O_2 generated do not explode, presumably because of the lack of an ignition source. However, after a sufficient amount of gas has diffused to the recombiner, an explosion is usually ignited by the catalyst. In a typical $2 KEWB excursion, the time delay between peak power and the explosion was ~2 min. The maximum peak pressure in the core generated by gas explosions was found to be ~450 psi. SUPO has a safety feature which, if functional, might prevent an explosion of this type. The assumed power transient would certainly produce a large pressure surge throughout the recombiner system, which would trip the System-to-Stack sensitrol and open a 3/4-in. valve that connects the gas loop directly to the stack exhaust line. It is possible that a sufficient amount of the excess gas would escape into the stack line so that the catalyst would not get hot enough to trigger an explosion.

An excursion of the type outlined above actually occurred at SUPO in December 1949, during measurements of rod drop times of the present boron rods.[14] There was no rod safety bar at that time. The rods had just been installed, and one of the cadmium rods was out of the reactor. The boron rods were first lifted individually, a safe procedure, since the removal of one rod was insufficient to make the reactor critical. Later, both rods were pulled, held for ~5 sec, and then dropped simultaneously. This procedure may have been repeated a second time. Radiation detectors indicated that an excursion had taken place, and it was observed that the soup temperature had risen ~22°C. There was no H_2-O_2 explosion. The staff member involved received 2.5 r of γ radiation, but no damage was done to the reactor. The total energy release was estimated to be ~1.2 Mw sec, which is somewhat larger than that accounted for by the estimated reactivity input of $1.15. However, not enough detailed information is available to permit a reliable analysis of this excursion.

2. Movement of Control Rods by Remote Control from the Reactor Console

Ramp reactivity insertions have been shown[8] to give rise to the following sequence of events: (1) A power transient occurs early in the ramp input, soon after $1 has been inserted; (2) The power falls to a low level (20-50 kw) within 1 sec after the initial transient, and then recovers slightly if rod withdrawal continues; and (3) After rod withdrawal has ceased, the power level will slowly decrease as the fuel temperature increases.

The KEWB investigators have concluded the following:[8] "Accidental rod withdrawal is of little concern with regard to hazard to the installation, personnel, or environs, in the case of a water boiler reactor of the KEWB type. Reactivity worth $5 can be added to the machine, by single or multiple rod withdrawal, without pressures being produced, instability resulting, or manual correction being required."

Regardless of the above statement, it is of interest to define the maximum accident attainable at SUPO through this means, since reactivity inputs that are the result of withdrawal of control rods from the console are perhaps more probable than large step additions.

The speed of the boron-rod drive mechanisms permit a maximum rate of reactivity addition, for each rod, of $\sim$7¢/sec. The selsyn-operated cadmium rods can be withdrawn more rapidly, 20¢/sec being considered the normal maximum rate. Although it is strictly against the operating rules to withdraw more than one rod at a time, it would be possible for the operator, having previously made the reactor critical, to withdraw both cadmium rods and the east boron rod simultaneously. In this way, it is humanly possible to add $2.3 in $\sim$5 sec, although it is well established that the Period scram would drop the rods before $1 could be inserted. Manipulation of the rods in this manner obviously cannot happen inadvertently. Such an extreme action on the part of the operator would be interpreted as a deliberate attempt to damage the reactor, and perhaps should not be the concern of the present report. Briefly, however, if the Period scram failed to operate, the excursion would presumably follow the pattern outlined at the beginning of this subsection. The minimum period attained would probably be >10 msec, and the maximum expansion pressure developed would be <30 psi. The principal hazard would be the possibility of a subsequent H_2-O_2 explosion. In this case, however, the recombiner system would necessarily be in operation, and it is the KEWB experience[6,8] that (1) ignition of radiolytic gas following a ramp-induced power transient is improbable when the gas circulation pump is operating, and (2) in those instances when an explosion does occur, it is a relatively low-grade combustion which produces only minor pressures in the core region. The sudden release of several liters of H_2-O_2 would undoubtedly transfer soup from the sphere to other parts of the recombiner system and possibly up to certain unshielded flow meters and pressure gauges on the reactor top. This would put the reactor out of commission for a short time, but there would be no serious damage.

Any future increase in the total excess reactivity of the assembly would not change the hazards associated with accidental control-rod withdrawal.

3. Changes in Sphere Environment, Including the Introduction of Experiments to the Reflector Region

Some of the known reactivity values which belong in this category are as follows:

	Reactivity Worth (gm U^{235})
Glory Hole full of graphite (vs air)	+6.15
Tangent Hole full of graphite (vs air)	+3.31
Glory Hole full of Lucite (vs air)	+8.0
Port 4W graphite (vs air)	+10.2
Cadmium rabbit in Glory Hole	-10.7
Fast-neutron source tube in Glory Hole	+38

The only significant value in this list is the 38 gm U^{235} ($1.4) for the fast-neutron source tube. This assembly has not been put in the Glory Hole for several years, and may never be used again. During those periods when it has been in use, the mechanical stop above the west boron rod has been lowered to effectively cancel the positive reactivity value of the source.

4. <u>Changes in Fuel Solution</u>

The primary reactivity values involved here are:

	Reactivity Worth (gm U^{235})
Addition of 185 cc HNO_3	-10
Addition of H_2O to fill sphere from bubbler level (~1000 cc)	+10
Water in cooling coils (vs air)	+21.5

In regard to the last item, the cooling coils are always full of water since the exit piping is vented to the atmosphere 10 ft above the sphere, and there is no way for the water to siphon out.

Since each of the above items is worth <$1, none of them can affect the over-all hazard unless the change is introduced at the same time that a significant amount of reactivity is being added by some other means.

D. <u>Radiation Hazards</u>

After a 40-hr shutdown, the Water Boiler solution has an activity of ~0.2 curie/cc. During 25 kw operation, the estimated activity of the solution is ~10 curies/cc, and that of the sweep gas in the recombiner loop is ~1 curie/cc. It is evident that any significant leak in the sphere or in the gas system could represent a serious health hazard. These hazards are discussed in the following paragraphs and in Section IX-E.

If the sphere wall were penetrated in such a way that soup could actually <u>flow</u> out of the core, most of the solution would descend through a slot in the graphite reflector to a stainless steel catch pan which underlies the entire reflector region. This pan drains through an underground line to a large stainless steel tank located in a concrete pit (10 ft underground) east of the reactor building. A vacuum pump maintains the outer tank at subatmospheric pressure, with the result that there is gas flow from the reflector region to the vacuum pump at all times. The activity of the vacuum system outlet is continuously monitored, and if this activity should rise to >5 times the normal maximum value, an alarm would be tripped in the reactor control room. In certain types of accidents, this alarm would probably give the first indication of the existence of a leak in either the sphere or the recombiner system. Another gas-activity monitor continuously samples the reactor room air, and will trip an alarm if the activity rises a factor of $\sim$10 above the normal maximum.

If a very slow soup leak were to develop, the reflector vacuum system should prevent dangerous amounts of radioactive gas from escaping into the reactor room air. In this case, it would be possible for the operating personnel to proceed with transfer of the soup from the sphere to a portable set of 1-liter Pb-shielded storage containers. The procedure for this transfer is outlined in reference 4.

In addition to the radiation detectors mentioned above, there are three γ-sensitive monitors located in the reactor room, one in the reactor control room, one in the valve house outside the reactor building, and one at the base of the South Mesa stack. Each of these monitors is connected to two recording panels: one located in the reactor control room and one just outside of the Omega Site main office. If the radiation level at one of the monitor stations exceeds the trip point, a bell rings at both recording panels and a pilot light comes on which identifies the responsible station.

Several years ago, there was concern about the possibility of a power failure occurring at the same time as a radiation accident, with the consequence that ventilation of the reactor room and control room would be inadequate, the stack exhaust system would cease functioning, and circulation of gas through the recombiner system would stop. This potential hazard has now been greatly reduced through the installation of an auxiliary gasoline-powered motor-generator system which is capable of supplying all of the reactor electrical demands. The time required to put this system into operation is 2 or 3 min.

E. Maximum Credible Accident

1. Description of Accident

In terms of danger to human life, the maximum credible accident at SUPO is believed to be the release into the atmosphere of all recombiner-system sweep gas after sustained operation at full power (25 kw). An accident of this type could conceivably result from a H_2-O_2 explosion, an event which the KEWB experiments have shown can generate peak pressures of several hundred psi, although these same experiments indicate that the possibility of a violent radiolytic gas explosion during steady-state power operation is extremely remote. It is probable that the sphere and all major units of the recombiner system would remain undamaged in such an explosion. However, some of the Magnehelic pressure sensors, which are connected to the recombiner system through 10-ft lengths of 1/8-in.-I.D. metal tubing, might receive pressure pulses large enough to blow off their plastic front covers, which would immediately release large quantities of radioactive gas into the reactor room.

The activity of the SUPO recombiner-system sweep gas (at 25-kw equilibrium) has been estimated by Busey[15] as $\sim$1 curie/cc. Since the gas volume in the entire system is $\sim$6 liters, the potential total activity release in the maximum credible accident (MCA) is roughly 6000 curies. The fission products responsible for this activity are largely the short-lived xenon and krypton isotopes. Iodine fission products are also known to be present, but their fractional contributions to the total activity is negligible (see Table IX and Appendix). Presumably, both iodine and bromine are strongly inhibited from leaving the fuel solution because of their solubility and highly reactive chemical nature.

All activities released into the sweep gas are subject to depletion by the flow ($\sim$200 cc/min) of gas into the stack exhaust line. Consequently, none of the gaseous activities build up to the equilibrium concentrations that one would estimate from the fission production rate.

TABLE IX

CALCULATED[a] MAXIMUM TOTAL ACTIVITIES OF IODINE ISOTOPES IN SUPO SWEEP GAS, AND ASSOCIATED "INFINITE" THYROID DOSE WHICH WOULD BE RECEIVED IN THE MAXIMUM CREDIBLE ACCIDENT

Isotope of Iodine	Half-life	Maximum Curie Strength in Sweep Gas[b]	Concentration in Reactor Room after MCA[c] ($\mu c/m^3$)	"Infinite" Thyroid Dose (rem/min)[d]
I^{131} + d[e]	8 day	0.27	180	4.9
I^{132}	2.3 hr	0.37	250	0.3
I^{133} + d	21 hr	0.60	400	3.4
I^{134}	53 min	0.54	360	0.2
I^{135} + d	6.7 hr	0.54	360	0.9
I^{136}	86 sec	0.02	14	negl.
I^{137}	24 sec	0.01	7	negl.
			Total	9.7

[a]See Appendix.

[b]After several months of constant operation at 25 kw.

[c]Assuming uniform dispersal in a volume of 1500 m^3.

[d]Total rem received during each minute spent in the reactor room.

[e]The "d" indicates that daughter activities have been considered in the dosage calculation.

In fact, the mean "residence" lifetime of any gas atom released into the circulating system is $\sim$30 min. This phenomenon preferentially suppresses the long-lived activities, so that radioactive gas extracted from SUPO decays more rapidly than gross equilibrium fission products.

2. Hazard to On-site Personnel

The instantaneous dispersion of 6000 curies of fission-product activity in the reactor room is obviously of grave concern to the occupants, and the room would be evacuated as quickly as possible. Warning of the high radiation field would be given at once by the three γ-sensitive monitors located in the reactor room, each of which produces a loud audible signal at a threshold of $\sim$20 mr/hr. Also, evacuation instructions would be immediately announced over the public address system. The total radiation dose received by an individual during evacuation is highly dependent on his exit path and on the dispersion pattern of the released gas. For simplicity of calculation, it will be assumed that the 6000 curies are uniformly dispersed throughout the reactor room, that the room volume can be represented by a hemisphere with 9-meter radius, and that the ventilation system has been shut off. On this basis, an individual would receive a whole-body exposure of approximately 25 rem ($\beta + \gamma$) if it took him 1 min to leave the room.

In regard to inhalation of the gaseous activity, the principal hazard is from the iodine isotopes, which collect in the thyroid gland once they have been absorbed into the body. On the basis of the calculated equilibrium concentrations of iodine isotopes given in Table IX, the assumed dispersion of this activity into a volume of 1500 m^3, and the ICRP thyroid dose values given in reference 16, it is estimated that a man with normal respiration rate would receive an "infinite" thyroid dose of $\sim$10 rem during each minute spent in the reactor room following the MCA.

The current accidental or emergency exposure limits are 150 rem to the thyroid and 25 rem whole body. Thus, whole-body exposure is the principal hazard in the MCA. It appears, however, that there is an excellent chance that no one would receive an exposure greater than the allowable emergency value. Something not considered in the above discussion is the fact that the large ventilating fan above the reactor would certainly remove some of the gas from the room before the fan motor could be shut off. This could reduce the potential exposure hazard by a large factor.

3. Hazard to Off-site Population.

Regarding the hazard to the nearby population in the MCA, it will be assumed that all of the gas activity (6000 curies) is instantaneously released into the atmosphere above the reactor building. The housing area closest to SUPO is a trailer park which borders the 300-ft-deep canyon in which the reactor is located. The actual distance to the edge of the trailer area is $\sim$600 ft. The assumptions made in computing the potential radiation dose to a person in the trailer area are (1) that the wind velocity is 1 mph, (2) that the radioactive cloud spreads by one-seventh of its downwind travel,[17] (3) that the initial radius of the cloud is 100 ft, and (4) that the total gas activity decays a factor of two in the $\sim$5 min required for the cloud to drift to the trailer area. Under these conditions, the whole-body dose would be $\sim$1 rem, and the "infinite" thyroid dose would be $\sim$0.2 rem. The current exposure limits for off-site personnel are 3 rem to the thyroid and 0.5 rem whole body.

X. Administration

The operations performed and the specific procedures employed in operating the reactor are subject to the review and approval of the LASL Reactor Safety Committee (RSC). This committee was formed in 1961 by the Director of the Laboratory. All matters relating to the over-all safety of the reactor such as changes in design, operation, or procedures are submitted to this committee.

The day-to-day responsibility for reactor safety obviously rests with the operating group. The Water Boiler Committee, composed of the

Group Leader and two senior group staff members, establishes the general policies for the operation of the reactor (subject to approval by the RSC). It also functions as a board of review and approval for all experiments, procedures, or reactor design changes which might significantly affect the safety of the reactor or of personnel on the site.

The reactor supervisor is responsible to the Water Boiler Committee for the safe operation and maintenance of the reactor and its subsystems. His specific responsibilities include:

1. Direct supervision of the reactor operator, making sure that the approved operating procedures are being followed.

2. Constant review of operating practices to determine whether or not the safety aspects can be improved.

3. Scheduling of reactor operation, experiments, and maintenance.

4. Discussion of proposed experiments and irradiations with potential users of the reactor.

5. The recording of details of all maintenance operations in the reactor log book.

6. The preparation, insertion, and removal of samples in accordance with established procedures. This includes the inspection of samples submitted by others to see that the materials, packaging, and containers conform to the accepted standards of safety.

7. Maintenance of adequate records on all activated materials, and proper storage of radioactive samples.

8. Surveillance of reactor area radiation levels whenever changes are made in experimental setups. The supervisor is to see that adequate warning signs are placed at locations where radiation levels are above tolerance. Assistance of the health monitor assigned to the site will normally be obtained in accomplishing these tasks.

XI. Future of the Water Boiler

In spite of the higher neutron flux locally available at the Omega West Reactor (OWR), SUPO continues to be in demand for various research activities. This is largely due to certain unique advantages offered by a reactor of the water boiler type. These include the following:

1. Operationally, the reactor is highly flexible. The power level can be changed rapidly, constant-power operation is easily established anywhere between 0.1 watt and 25 kw, and there is no xenon poisoning problem.

2. Excellent flux reproducibility. Depending somewhat on the time interval between measurements, a flux reproducibility of $\sim$1% is attainable at all power levels from 10 watts to 25 kw. This results largely from the fact that the flux distribution does not change with time.

3. Low ratio of γ rays/neutron. In the immediate vicinity of the sphere, this is largely a consequence of the low critical mass. Furthermore, the thermal-column regions are shielded by an 8-in. bismuth wall. The relatively low γ-ray background is a great advantage in those experiments where γ rays and neutrons induce similar effects, such as in biological studies. Also, γ heating is no problem.

4. Operation of the facility is not a large effort in terms of either manpower or money. There are two full-time people assigned to the reactor: an operator and a supervisor. The cooling water and power consumption requirements are completely negligible, there is no fuel-handling problem, and the total U^{235} inventory is only $\sim$1 kg.

In many respects, SUPO and the OWR tend to complement one another. Also, SUPO serves as a backup facility in case the OWR has to be shut down for any reason.

SUPO has now been operated for over 11 years, with no forced shutdowns of >2 weeks duration -- a record which few other reactors can match. Among the difficulties that would clearly result in a long-term shutdown of the reactor are development of a leak in either the sphere or the cooling coils, development of a leak in certain critical areas of the recombiner system, or excessive growth of the reflector graphite. The future is of course uncertain, but it may be many more years before these or other serious problems arise.

APPENDIX

CONCENTRATION OF IODINE FISSION PRODUCTS IN THE WATER BOILER SWEEP GAS†

In January 1963, a sample of SUPO sweep gas was examined over a period of several days with a NaI(Tl) scintillation spectrometer. From the observed γ-ray spectra, it was possible to determine the concentrations of certain iodine fission products relative to those of krypton. These experiments are briefly described and interpreted in the following paragraphs.

The gas sample was obtained by drawing gas up pressure line No. 9 (see Fig. 11). Transit time of the gas from the circulating system to the collection cell was $\sim$2 sec. The γ-ray spectrum was first examined at T_0 + 45 min, at which time 32-min Cs^{138} (daughter of 17-min Xe^{138}) was the dominant activity. After 5 hr of decay, 2.8-hr Kr^{88} and its daughter, 18-min Rb^{88}, were major contributors to the observed γ-ray spectrum. After 30 hr, 21-hr I^{133}, 8-day I^{131}, and 12.8-day Ba^{140} were the principal γ-ray emitters. From the calculated atom ratios of I^{131}/Kr^{88} and I^{133}/Kr^{88} which existed at T_0, it was evident that iodine is almost completely inhibited from leaving the fuel solution. In order to express this inhibition mathematically, it was necessary to develop a model for the evolution of iodine from the soup. In this model, it is assumed that iodine is evolved only when the reactor is at power, and that the evolution rate is proportional to the number of iodine atoms present in the solution. The differential equation which describes the rate of change of the total number of atoms, N_I, of a particular iodine isotope <u>in the sweep gas</u> is then

†The work described in the appendix was performed by M. E. Bunker and J. W. Starner.

$$\frac{dN_I}{dt} = f_I S_I - \lambda_I N_I - \lambda_{Leak} N_I \tag{1}$$

where S_I = number of atoms of the iodine isotope in the soup, and is given by

$$S_I = \frac{kY_I}{(\lambda_I + f_I)} \left[1 - e^{-(\lambda_I + f_I)t}\right] + S_I^0 \, e^{-(\lambda_I + f_I)t} \tag{2}$$

where S_I^0 = value of S_I at $t = 0$

f_I = probability per unit time that an iodine atom will escape from the fuel solution (sec^{-1})

λ_I = radioactive decay constant of iodine isotope (sec^{-1})

λ_{Leak} = probability per unit time that an atom in the sweep gas will leak out of the recombiner system (sec^{-1}) (see text, Section IX-E1).

k = the fission rate (sec^{-1})

Y_I = fission yield of the iodine isotope.

The solution to equation (1), for the case of $f_I \ll \lambda_{Leak}$, is

$$N_I = \frac{f_I a_I}{\lambda} \left\{ (1 - e^{-\lambda t}) - \frac{\lambda}{\lambda_{Leak}} \left(1 - \frac{S_I^0}{a}\right) \cdot \left[e^{-(\lambda_I + f_I)t} - e^{-\lambda t}\right] \right\} \tag{3}$$

where $\lambda = \lambda_I + \lambda_{Leak}$

and $a_I = \dfrac{kY_I}{\lambda_I + f_I}$

If $e^{-\lambda t} \approx 0$, which in the case of SUPO means $t > 2$ hr, the total <u>activity</u> of the iodine radioisotope in the sweep gas becomes

$$A_I = \frac{f_I a_I \lambda_I}{\lambda}\left[1 - \frac{\lambda}{\lambda_{Leak}}\left(1 - \frac{S_I^0}{a_I}\right) e^{-(\lambda_I + f_I)t}\right] \tag{4}$$

The krypton isotopes are assumed to be released instantaneously from the solution, with 100% evolution. The differential equation which describes the time variation of the number of atoms of a particular krypton isotope in the sweep gas is

$$\frac{dN_{Kr}}{dt} = kY_{Kr} - \lambda_{Kr} N_{Kr} - \lambda_{Leak} N_{Kr} \tag{5}$$

which gives, for the total activity (A_{Kr}) of the krypton isotope, the relation

$$A_{Kr} = \lambda_{Kr} N_{Kr} = \frac{\lambda_{Kr} k Y_{Kr}}{\lambda} \left(1 - e^{-\lambda t}\right) \tag{6}$$

where $\lambda = \lambda_{Kr} + \lambda_{Leak}$

By making use of the above equations, the measured activity ratios A_{I-133}/A_{Kr-88} and A_{I-131}/A_{Kr-88}, and the operational history of the reactor prior to the sample extraction, it was possible to solve for the iodine "escape" constant, f_I. For both I^{133} and I^{131}, the value for f_I was found to be $\sim 2.7 \times 10^{-7} (\text{sec}^{-1})$. This means that during 1 hr of operation at 25 kw, the probability that a particular iodine atom will escape from the solution is $\sim 10^{-3}$. Thus, short-lived iodine isotopes are less populous in the sweep gas than the long-lived isotopes, because most of the short-lived nuclei undergo radioactive decay before they are able to escape from the soup.

The calculated activities of iodine isotopes given in Table IX of the text were obtained using equation (4) and the following values for the constants: $\lambda_{Leak} = 5.6 \times 10^{-4}$ sec^{-1}, $f_I = 2.7 \times 10^{-7}$ sec^{-1}, and $k = 7.5 \times 10^{14}$ fissions/sec (at 25 kw). Values of Y_I were obtained from reference 18.

REFERENCES

1. L. D. P. King, "The Los Alamos Homogeneous Reactor, SUPO Model," Los Alamos Scientific Laboratory Report LA-1301 (1952).

2. L. D. P. King, R. P. Hammond, J. A. Leary, M. E. Bunker, and W. R. Wykoff, Nucleonics 11 (No. 9), 25 (1953); also "Gas Recombination System of the Los Alamos Homogeneous Reactor," Los Alamos Scientific Laboratory Report LA-1337 (1952).

3. L. D. P. King, "Design and Description of Water Boiler Reactors," Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Vol. 2, p. 372 (United Nations, New York, 1956).

4. J. E. Evans, L. D. P. King, and J. W. Schulte, Nucleonics 11 (No. 1), 67 (1953).

5. P. R. Kasten, Nuclear Engineering (C. E. P. Symposium Series), Part 1, 50, 229 (1954); also R. N. Lyon, "Preliminary Report on the 1953 Los Alamos Boiling Reactor Experiments," Oak Ridge National Laboratory Report CF-53-11-210 (1953).

6. J. W. Flora, "Status Review of the KEWB Program," Atomics International Division of North American Aviation, Inc., Report NAA-SR-MEMO 3569 (1959).

7. D. P. Gamble, "A Model of Bubble Growth in the KEWB Reactor," Atomics International Division of North American Aviation, Inc., Report NAA-SR-MEMO 3320 (1959).

8. J. W. Flora et al., "Kinetic Experiments on Water Boilers, 'A'-Core Report, Part I, Program History, Facility Description, and Experimental Results," Atomics International Division of North American Aviation, Inc., Report NAA-SR-5415 (1962).

9. M. Dunenfeld et al., "Kinetic Experiments on Water Boilers, 'A'-Core Report, Part II, Analysis of Results," Atomics International Division of North American Aviation, Inc., Report NAA-SR-5416 (1962).

10. P. Spiegler, C. F. Bumpus, Jr., and A. Norman, "Production of Void and Pressure by Fission Track Nucleation of Radiolytic Gas Bubbles During Power Bursts in a Solution Reactor," Atomics International Division of North American Aviation, Inc., Report NAA-SR-7086 (1962).

11. M. A. Greenfield, "Kinetic Experiments on Water Boilers, 'A'-Core Report, Part IV, Containment Aspects of Hydrogen-Oxygen Explosions," Atomics International Division of North American Aviation, Inc., Report NAA-SR-5418 (1960).

12. F. Schroeder, "Power Excursion Studies," Proceedings of the 1960 Idaho Conference on Reactor Kinetics, AEC Research and Development Report IDO-16791, p. 37 (1962).

13. M. E. Remley, "Presentation of the KEWB Program," Proceedings of the 1960 Idaho Conference on Reactor Kinetics, AEC Research and Development Report IDO-16791, p. 57 (1962).

14. W. R. Stratton, "A Review of Criticality Accidents," Technology, Engineering, and Safety, Progress in Nuclear Energy, Series IV, Vol. 3, p. 163 (Pergamon Press, New York, 1960).

15. H. M. Busey, Nucleonics 12 (No. 5), 9 (1954).

16. "Recommendations of the International Commission on Radiological Protection, ICRP Publication 2, Report of Committee II on Permissible Dose for Internal Radiation" (Pergamon Press, New York, 1959).

17. Summary Report of Reactor Safeguard Committee, WASH-3 Rev. (1950).

18. J. O. Blomeke and M. F. Todd, "Uranium-235 Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time," Oak Ridge National Laboratory Report ORNL-2127 (1957).