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CRITICALITY ANALYSIS: LWBR ASSISTANCE PROGRAM IN BUILDING 3019

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CRITICALITY ANALYSIS: LWBR ASSISTANCE PROGRAM IN BUILDING 3019

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MARCH 1972

*Y-12 Plant

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CRITICALITY ANALYSIS: LWBR ASSISTANCE PROGRAM IN BUILDING 3019

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ABSTRACT

Equipment items used for handling ^{233}U in the LWBR program assistance effort in Building 3019 at the Oak Ridge National Laboratory were analyzed to define the criticality control methods to be applied. The dimension, function, and location of each item are stated, and the primary criticality control parameter is identified; secondary control methods are also specified for each item. The data found in standard literature references were used as a guide in most cases. Special calculations were made for cases in which the literature does not apply directly or does not define the limits as closely as plant conditions require.

1. INTRODUCTION

Oak Ridge National Laboratory (ORNL) has received and stored several hundred kilograms of ^{233}U that will be converted to ceramic-grade $^{233}UO_2$ powder for use in the Light Water Breeder Reactor Program (LWBR). This work, performed under contract with the Bettis Atomic Power Laboratory (BAPL), will involve, in addition to the receipt and storage of material, the purification of this material by solvent extraction and ion exchange and its subsequent conversion to ceramic-grade oxide at the nominal rate of 20 kg/week. The oxide will be blended with ThO_2, pressed into pellets, and loaded into fuel rods at BAPL. Some types of scrap generated at BAPL will be returned to ORNL for recovery.

This report covers the nuclear safety aspects of the entire ORNL effort. A companion report[1] covers all other safety and hazard aspects associated with the process.

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2. METHOD FOR EVALUATING CRITICALITY SAFETY

An evaluation of the facilities involved in the LWBR support program has demonstrated that the criteria for nuclear criticality safety are satisfied. The evaluation covers the equipment geometry and neutron absorbing materials; for the cases where other primary criticality control is needed, the practices and procedures employed (e.g., mass limit and concentration control) are indicated.

The evaluation is summarized in Table 1, which is the basis for the nuclear safety review. This table lists each uranium vessel having a diameter of 4 in. or more and specifies the operating conditions. The control methods applied to each equipment item are also shown.

The body of this report expands the tabulated information with explanations of function, location, and method of applying controls. Additional supporting calculational results are also included; most of these deal with specific application of the generalized control parameters. Some additional information on equipment relationship is also provided. The process analysis section of the report is organized in the same manner as Table 1; the operating sequence is the guide. A statement of function is presented for each flowsheet in order to clarify the manner in which the controls are applied. The methods of criticality control are in accordance with the general concepts set forth in the Appendix to Chapter 0530 of the AEC Manual.[2] These concepts are embodied in the following three definitions:

(1) Primary Criticality Control: A method of control, necessarily reliable, upon which primary dependence is placed to prevent a nuclear chain reaction.

(2) Secondary Criticality Control: A method of control that is relied upon to prevent a nuclear chain reaction in the event of failure or ineffectiveness of the primary criticality control.

(3) Double Contingency Principle: Process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a nuclear incident is possible.

Table 1. LWBR Program Process Vessel Criticality Evaluation

Part 1. Solution System

Vessel Number	Operation	Flow-sheet Dwg.[a]	Size: Length (ft)	Size: OD (in.)	Volume (liters): Total, Without Rings	Volume (liters): Working	Neutron Absorb.: Glass Rings (% glass)	Neutron Absorb.: Th/U Ratio in $Th(NO_3)_4$	Flowsheet Concen-tration (g U/liter)	Max. Safe Concen-tration (g U/liter)	Dist. (in.) from: Uranium-Containing Vessels	Dist. (in.) from: Reflectors (walls)	Prev. Nuclear Safety Review	Criticality[b] Control Methods: A	B	C	D	E	F	G
U-1 - U-10 †	Liquid storage	062	4	36	900	400	34		≤250	250 *	2	3	527		P			S	S	✓
U-20 and 21	Liquid storage	062	2	4	4.75	2.4			≤250	500	>50	2	527	P				S	S	
S-2 ††	HE feed adjust.	063	5	48	1700	1500		≥17:1	10	10	R-25:54	SJ[c] 12	597		P		S		S	✓
S-4	HE feed tank	063	6.5	36	1230	1150		≥17:1	8	10	N-1:30	SJ 12	597		P		S		S	
S-12	HE lean acid	063	4	36	800	560			<1	7.5			597		P				S	
S-15	HE ox. diss.	063	13.1	4		16			≤400	m	S-16:18	SJ 18	477	P	S				S	✓
S-20 ‡	HE scr. diss.	063	14.6	8		211		≥17:1	≤10	m	N-1:24	SJ 5	488	S	P		S	S	S	✓
R-25 †	HE evap. prod.	063	4.5	18	215	115	24		≤250	250	R-35:27	6	597		S			P	S	✓
R-35	HE ox. diss. prod.	064	11	5NPS	42	32		(Cd)	≤300	400	R-25:27	24	477		P				S	
N-1	SX extr. col.	064	39	5NPS	220	220	24[d]		≤21	250	S-4:30	SJ 15	597	S	P			S	S	
N-2	SX aq. waste	064	5.5	36	1150	800		>>17:1	<0.01	10	N-16:21	SJ 3	597		P		S		S	
N-4	SX aq. waste c.	064	5.5	36	1150	800		>>17:1	<0.01	10	N-1:18	SJ 10	597		P		S		S	
N-23	SX org. accum.	064	2.75	18	130	130 (full)			<0.01	7.5	N-1:30	9	597		P				S	
P-1	SX scrub col.	064	17.5	5NPS	100	100 (full)	24[e]		≤21	250	P-11:24	SJ 12	597	S	P			S	S	
R-1	SX strip col.	064	24	5NPS	121	121 (full)	24[f]		≤31	250	R-11:27	SJ 22	597	S	P			S	S	
R-11	SX org. hd. tank	064	8.25	6NPS	46	46 (full)			≤21	100	R-1:27	SJ 20	597	S	P				S	
P-69	SX org. c. tank	065	3	42	845	450	24		≤0.01	150	P-69FC:51	48	597		S			P	S	
P-69FC	SX org. fl. ck.	065	5	12	90	48	24		≤0.01	180	P-69:51	15	597		S			P	S	
P-70	SX aq. sol. wsh.	065	4.5	42	1050	800			≤0.01	7.5	R-2:18	SJ 15	597		P				S	
R-2 §	IX recovery tank	065	5.5	36	1200	645	24		<100	180	R-3:36	SJ 4	597		S			P	S	✓
R-3 §	SX prod. catch tank	065	5.5	36	1200	645	24		<125	180	R-2:36	SJ 4	597	S	S			P	S	✓
R-22 †	SX evap.	065	11.5	5NPS	45	12			≤250	400	R-32:9	SJ 3	597		P				S	✓
T-20	SX solv. wash	065	4	18	190	100			<0.01	7.5	X-14:48	SJ 12	597		P				S	
X-11ABCD ‡	IX feed tank	066	20	4	200	140			≤125	400	X-15A:8	10	583	P					S	✓
X-15ABCD	IX prod. tank	066	20	4	200	140			≤125	400	X-11D:8	10	583	P					S	✓
X-14	IX int. prod. tank	066	5	18	240	115	34		≤125	250	X-17:6	28	583		S			P	S	✓
X-17	IX elut. tank	066	5	18	240	115	34		≤20	250	X-14:6	4	583		S			P	S	✓
N-7	SX aq. rework	067	6	24	570	305	24		≤10	200	N-8:24	SJ 9	597		S			P	S	✓
N-8	SX rework sep.	067	6	24	545	290	24			200	N-7:24	SJ 9	597		S			P	S	✓
N-16 §	SX VOG drain	067	6	24	575	307	24			200	N-2:21	SJ 9	597		S			P	S	✓
P-6	SX evap. cond.	067	5.5	60	3200		24[g]		<1	7.5	P-1:36	9	597		P			S	S	✓
Dem. H_2O	OX dil. and wash	M03	5	4	12.5	10			0	500	UNH:20	h		P					S	
HNO_3	OX diss. and adj. pH	M03	5	4	12.5	10			0	500	CiSn:21	h		P					S	
UNH	OX feed metering	M03	5	4	12.5	10			<125	500	W.Rec.:18	h		P					S	
Prec.	OX chem. reaction	M03	5.5	4	13.75	10			<125	500	Cent.:17	h		P					S	
Washdown	OX decon.	M03	3	4		4-5			0[i]	500				P					S	
Waste rec.	OX rework	M03	5	4		10			>0.1[i]	500	UNH:18			P					S	
Cent. snat.	OX rework	M03	6	5		15-20			>0.1[i]	400	Rinse #1: 16			P	P				S	
Cent.	OX dewatering	M03		11		j			>125[l]	500	Prec.: 17	h		P	P				S	✓
Waste neut.	OX pH adj.	M03		8		k			>0.01[l]	45	Rinse #2: 10	h			P				S	
Rinse #1	OX encl. cleaning	M03		5		k			>10[i]	400	Cent. snat.: 13	h		S	P				S	
Rinse #2	OX waste acc.	M03		8		k			>0.01[i]	45	W. Neut.: 10	h			P				S	
Iso. dil.	OX final waste tr.	M03		8		k			>0.01[i]	45	None	h			P				S	
Cent. sn. pot	OX catch pot	M03		4		0.5-1.5			>0.1[i]	500	Cent.: 10	h		P					S	

* $k_\infty \sim 0.1$ (p. 44)

†† ANISN $k_\infty \leq 0.5$

† ANISN, SENS

§ ANISN k_∞ $k_{eff} = k_\infty / (1 + M^2 B^2)$ § app A

Table 1. (continued)

Part 2. Solids Accumulation

Location	Operation	Flowsheet Drawings	Normal Loading	Safe Mass (kg)	Remarks	Criticality[b] Control Methods A	B	C	D	E	F	G
Centrifuge bowl	Dewatering	M03	≤1.0 kg	>2	See Sect. A-13.1 Appendix A; must use Teflon bowl.	P		P			S	✓
Microwave oven	Drying	M03	≤1.0 kg	>2	As above.	P		P			S	✓
Furnace	Heat treatment	M03	≤5.0 kg	>>100	Slab geometry; no hazard.	P		P			S	
Blender	Mixing	M03	≤22 kg	>>100	Moderator must be excluded; see Sect. A-13.2, Appendix A.			P			S	✓
Penthouse wells	Oxide storage		<1 kg/ft	5 kg/ft	See A-7, Appendix A; safe mass function of can size and oxide density.			P			S	✓
Charging box	Dissolver loading	063	<1 kg	>9	Must be dry.			P			S	

Part 3. Loss of Solution Containment

Location	Operation	Flowsheet Drawings	Area (ft^2)	Safe Slab Capacity (kg)	Remarks	Criticality[b] Control Methods A	B	C	D	E	F	G
Center pipe, tunnel floor	UNH liquid storage	062, 066	235	>900	6000-liter volume in a 34% glass array.					P	S	
Cell 5	SX	063, 064 065, 067	319	>60	34% glass up to 3-in. depth; safe capacity holds for all concentrations.					P	S	
Cell 6	SX, IX	064, 066 067	277	>60	Same as for Cell 5.					P	S	
Cell 7	SX, IX	064, 065 066	329	>60	Same as for Cell 5.					P	S	
Liquid waste handling encl.	Solutions, oxide system	M03	35	9.9	Box mass limit is 2.0 kg.		P	S			S	
Precipitation enclosure	Chemical reaction	M03	8.25	2.3	Box mass limit is 1.5 kg.		P	S			S	
Isotopic dilution encl.	Waste soln. disposal	M03	12	3.4	Box mass limit is 0.2.		P	S			S	

[a] Dwg. No. applies to F 12229 CP --- series, except for M03, which is in F 12229 CD --- series.

[b] Criticality control methods are identified in this table:
- A- Geometry. Safe for any solution up to limit shown (TID-7016 Rev. 1).
- B- Concentration control.
- C- Mass limit (TID-7016 Rev. 1).
- D- Soluble poison.
- E- Fixed poison. (ANS proposed standard as basis.)
- F- Administrative control.
  - (1) Run sheets (4) Supervisory approval
  - (2) Locked controls (5) Radiation background readings
  - (3) Material balance (6) Box mass limits
- G- Specific case calculation described in Appendix.
- P- Primary.
- S- Secondary.

[c] SJ = Steam jacket.

[d] Borosilicate-glass rings in expanded top section.

[e] Borosilicate-glass rings in expanded top and bottom sections.

[f] Same as (e) except for strip column.

[g] Rings in bottom 6 in. only (300 liters).

[h] Located in glass and sheet-metal box; no reflectors nearby.

[i] Concentrations vary; where numbers are shown, they represent maximum estimates.

[j] Centrifuge feed controlled by withdrawal rate; no fixed liquid level.

[k] Vessels accumulate solutions; level is manually controlled.

[l] Centrifuge feed is a solid-liquid mixture.

[m] This vessel is safe for all mixtures of oxide and solution, provided the stated absorber is present.

2.1 Principles of Criticality Control

Each of the several methods for providing criticality control for a specific vessel or operation is designated in such a way that the concept can be referred to easily in the detailed single-vessel analyses that follow in this report. These methods are listed below.

A. Geometry: Nuclear safety by geometrical means implies the use of vessels that are safe for any ^{233}U concentrations which can be encountered in the vessel.[3] It involves consideration of changes due to corrosion, loss of contents by leakage, reflectors nearby in the process location, and interaction with other vessels.

B. Concentration Control: Concentration control is achieved by the use of multiple, independent administrative procedures. These procedures include the use of safe vessels for concentration adjustment prior to transfer into the vessel subject to concentration control, and analytical results verified by inventory data and process instrumentation. In special cases, the concentration of a solution is limited by process conditions and cannot reach an unsafe level as the result of an operational error or equipment failure.

C. Mass Limit: The total mass in an enclosure or vessel is limited by multiple, independent administrative procedures to a safe level for the type of material being processed.

D. Soluble Neutron Absorber: Thorium nitrate is present in scrap dissolver product and provides an additional margin of safety in the feed make-up vessel.

E. Fixed Neutron Absorbers: Borosilicate-glass raschig rings are used for criticality control in large-diameter vessels that may contain uranium at high concentrations. When used as a primary criticality control, the method (1) provides assurance of the presence and effectiveness of the rings, (2) maintains the concentration below the recommended safe concentration by two or more reliable and independent administrative procedures, and (3) demonstrates safe collection in the event of leakage. Except as noted, borosilicate-glass raschig rings will be used in accord with ANSI N16.4.[4]

In some cases, inspection procedures have been modified and are discussed later in the report. All process vessels except expanded column sections are inspected visually and recalibrated at the start of an operating program. The expanded column sections are radiographed.

F. Administrative Procedures: The administrative procedures that are used to implement these criticality control methods are applicable to all operations.

(1) Run sheets supplemented, as necessary, by additional written procedures are used for each process operation. Each run sheet consists of a set of instructions and a checklist that provides a step-by-step operational guide for the given process operation. For each process operation, the run sheet identifies limiting values of the process conditions, gives a procedure for collecting any quantitative information that is needed, provides for operator signification (by a check) that he has complied with the procedure, and requires verification and signature by the supervisor who has independently assessed the performance of the procedure. Each entry on the checklist implies that the operator has identified and reviewed reliable quantitative information which will permit an assessment of the precision of the criticality control procedures. The checklist (integral with the run sheet), which is completed by operating personnel and initiated by supervisory personnel, constitutes use of the double contingency principle.

(2) All means of transferring fissionable material are controlled by locked systems; locked line valve controls are used for solution movements, and locked doors or other access openings are used for other materials. The keys that are required can be obtained only from supervisory personnel as needed at the appropriate step in the run sheet.

(3) Material balances are calculated every 8 hr on operating units of the plant (more frequently for certain items). The inventory data required are obtained from instrument readings and analyses recorded in the operating log. The material balance provides an independent check on the concentration and mass of ^{233}U contained in various pieces of equipment. The oxide production facility is operated on continuous material balance bases by means of in-process batch inventory.

(4) Radiation background readings are valuable tools for determining accumulations of ^{233}U above expected or desired amounts since the ^{232}U daughters emit high-energy gamma radiation. This affords another method for maintaining inventories.

3. PROCESS ANALYSIS

The total LWBR program includes storage of uranium oxide, storage of uranyl nitrate solutions, dissolution of oxide materials, purification by solvent extraction followed by ion exchange for thorium removal, and a final conversion of nitrate solution into a uranium oxide product. Most of these operations are shown in the seven flowsheets listed in Table 1; however, neither the oxide storage method nor the provisions for handling leakage from vessels are included. The provisions for handling leakage are discussed in a section following the flowsheet items. The complete process is analyzed in the subsequent sections in the order indicated.

3.1 Storage Systems

3.1.1 Storage of Liquids

Figure 1 shows the system used for storing liquids. It includes the ten storage tanks and connecting piping and also the vacuum pot used for sampling and transfers. All solution transfers into the storage facility are made through transfer vessel U-20. An automatic level control on this vessel limits the volume to a maximum of 2.4 liters per vacuum transfer; an alarm sounds at 2.4 liters when pressure transfers are being made. Solutions are brought into U-20 from cell 5 vessels R-25 and R-35, and also from shipping containers by means of a special glove-box connection. A protective reserve capacity for U-20 is provided by U-21, which is identical with U-20. The normal storage concentration in vessels U-1 through U-10 is 225 to 240 g of ^{233}U per liter, with an absolute maximum of 250 g of ^{233}U per liter. These tanks are 36 in. in diameter and are packed with borosilicate-glass raschig rings; glass occupies 34% of the total volume and is uniformly distributed, as shown by calibration curves. Each tank contains 400 liters when filled to maximum operating level (which is 70% of total capacity). All lines enter through the top and are controlled by remotely operated valves. Solution can be removed by vacuum lift into U-20 and via gravity discharge into R-25 or to a shipping bottle. Tanks U-5 through U-10 may also be discharged by pumping to the ion exchange feed tank, using a diaphragm pump located in the shielded pipe tunnel area with the storage tanks.

Nuclear Safety, Vessels U-1 through U-10.—The primary control method is use of a fixed neutron absorber in accordance with the proposed ANSI standard,[4] with two exceptions. In the first exception, the maximum solution concentration of 250 g of ^{233}U per liter is based on calculations made for this specific situation (see Sects. A-1, A-2, and A-3 of the Appendix). The second exception relates to details of inspection. Each of the ten tanks is inspected annually for ring settling and for solids deposition (by inventory balance only). The removal of rings for analysis, determination of solids deposition, and evaluation of physical condition are done annually on one tank from each group of five (U-1 - U-5 and U-6 - U-10). The same schedule is in effect for recalibration of tanks. Concentration is controlled predominantly by procedures F(1) and F(2); no solution is added to these tanks without a prior concentration measurement and adjustment in another vessel. All operations are effected by means of written procedures, and all transfers are made through lines having locked valves. All operations require control and the issue of keys by the supervisory staff. Tank leakage onto the floor is controlled by glass raschig rings (see Sect. 3.5).

Nuclear Safety, Vessels U-20 and U-21.—Geometry is the primary means of nuclear safety control for transfer vessel U-20; the 4-in.-diameter vessel is safe for a ^{233}U concentration of up to 550 g/liter when fully reflected. Tank leakage onto the floor is controlled by glass raschig rings.

3.1.2 Oxide Storage Facility

There is no flowsheet for oxide storage since unloading and loading are manual operations and storage, per se, is a static situation.

The oxide storage facility is located in Building 3019 and utilizes the off-gas system, the monitoring systems, and the material handling equipment already present in the building. The facility consists of a total of 26 holes drilled in the concrete walls separating cells 2, 3, 4, and 5. The locations, depths, and other details of installation are shown in ORNL Dwgs. 10943-L-001, -002, -003, and 12229-CP-001, -002, -003.

Each hole is lined with stainless steel pipe and vented to the off-gas system. A plug seals the top of the pipe after the containers of oxide have been deposited in the pipe; this seal is intended to reduce the inflow of air to a minimum to avoid overloading the off-gas system.

Holes drilled into massive concrete form a fixed matrix for spacing the stored material and also provide some shielding against the gamma radiation associated with the decay of ^{232}U daughters. Lead shielding is used on the top surface of the storage area, and lead is used as a lining in nine of the storage wells to reduce dose rates in work areas in the penthouse and in adjoining cells. Two of the three sets of storage wells are drilled in a linear array; a third set of nine holes between cells 2 and 3 is drilled as a triangular matrix with 18-in. center-to-center spacing.

The packages stored in these wells previously have varied in size and in content. For the purpose of criticality evaluation, calculations of k_{eff} were made for a range of can diameters, oxide densities, storage array, and water content. These calculations are reported in Sect. A-7 of the Appendix. The results have been reduced to the following set of restrictions:

(1) The maximum inside diameter for a can shall be 3.25 in., and the maximum density of $^{233}UO_2$ shall be 3.5 g/cm^3, thereby resulting in a maximum mass of 5.7 kg of $^{233}UO_2$ per foot.

(2) No significant quantities of liquid water shall be permitted in the storage holes.

(3) The H/U ratio in the oxide shall be less than 0.4, or less than 1.3 wt % H_2O.

The values being used currently for the storage system are much lower than those just given. The diameter of each can is 2.75 in. (ID), and the density of the $^{233}UO_2$ is less than 1.5 g/cm^3; the quantity of moisture present is low (averaging 1.2%), and no liquid water is present. (There is no credible means for water to enter the storage holes.) Some $^{233}UF_4$ is still stored in the system under these rules; it is more dilute than either $^{233}UO_3$ or $^{233}UO_2$.

The storage facility will also be used to store thoria-urania scrap pellets that have been returned for recycling. With a density of 10 g/cm^3 and a maximum ^{233}U content of 5.5%, these pellets are equivalent, with regard to uranium concentration, to 1.021 kg of $^{233}UO_2$ per foot of depth when packaged in 3.25-in.-ID cans. This is well within the safe limits noted.

Nuclear Safety.—Criticality control is achieved by limiting the mass for the unmoderated oxide (see Sect. A.7 of the Appendix). This is accomplished by an administrative procedure that specifies can size and provides locked-cover control. The wells are protected against water entry as follows: (1) the tops of the wells are elevated above the surrounding floor, (2) the wells are plugged and sealed, and (3) the off-gas line is protected against backflow of condensed moisture by the reverse slope of the takeoff line.

3.2 Purification Systems

All of the uranium to be used in the conversion system must be purified from ionic contaminants and also from ^{232}U decay daughters. A low level of ionic contaminants is specified for the final oxide product. Most of the purification is done in the solvent extraction system. The resulting product stream, after concentration to 120 g of ^{233}U per liter, is further purified by ion exchange. The system used was designed to reduce thorium concentration in the product to a very low level. In this way, ^{228}Th (from ^{232}U decay) as well as ^{232}Th is removed, and the decay chain is broken. On decay of the daughters of ^{228}Th, which are short-lived, the gamma activity of the product reaches a very low level.

3.2.1 Head-End System

The uranium-containing equipment items included in the head-end system (see Fig. 2) are: (1) dissolver charging box; (2) oxide dissolver, S-15; (3) scrap dissolver, S-20; (4) inventory tank, R-35; (5) solution storage tank, R-25; (6) feed adjustment tank, S-2; and (7) feed tank, S-4. Two separate functions are served by these pieces of equipment;

the first is dissolution of high-purity uranium oxide feed material to prepare a uranyl nitrate feed solution for the ion exchange purification step, and the second is dissolution of thoria-urania scrap material to prepare feed for the solvent extraction system. Solvent extraction is used to separate the thorium and uranium in the scrap dissolver product.

The Charging Box. — This box is common to both dissolver systems. It is a shielded, manipulator-equipped alpha enclosure and has two dissolver charging chutes, one for S-15 and one for S-20. Both chutes are double-valved. To ensure that no pure uranium oxide is charged into the S-20, the cover plate over the scrap chute is kept locked and the proper key is issued by the supervisor when needed. Oxide of either type is brought into the charging box by passing the shipping can through a sphincter valve. A can opener in the box opens the can; the empty cans are bagged out through a separate port.

Nuclear Safety. -- Nuclear safety control is achieved by limiting the mass; 1 kg of ^{233}U is the maximum inventory for the charging box. Administrative procedures are F(1) and F(2).

S-15 Dissolver. — The dissolver is a thermosyphon type. The major components are fabricated from 4-in. sched. 40 pipe; the steam jacket on the heated leg is made of 5-in. sched. 40 pipe. Acid and uranium oxide are charged to the dissolver in quantities calculated to produce a solution containing 400 g of ^{233}U per liter. The charging method requires that one can of oxide at a time be poured through the charging chute and into the hot acid. The double valve arrangement is used to avoid exposure of the charging box to the dissolver atmosphere. When the full uranium charge, $\leq$5.5 kg of ^{233}U, has been added and instrumentation verifies complete dissolution, sufficient water (already in place in R-35) dilutes the dissolver solution to a final ^{233}U concentration of 300 g/liter. This solution is transferred by gravity to tank R-35, where the final concentration adjustment is made.

Nuclear Safety. -- Dissolver geometry is used to achieve criticality control; its 4-in. diameter is a safe size at a concentration of 550 g of

^{233}U per liter when fully reflected.[3] Administrative procedures F(1) and F(2) apply.

R-35 Vessel. — This vessel receives solution only from dissolver S-15; transfer to storage via the U-20 transfer vessel is described in Sect. 2.1.

Nuclear Safety. -- Concentration is used as the principal means of control; the 5-in.-diam. tank is safe at a ^{233}U concentration of 400 g/liter when nominally reflected.[3] This is the maximum dissolver concentration that could be transferred to R-35. Administrative procedures F(1) and F(2) apply.

R-25 Vessel. — This vessel is an 18-in.-diam tank packed with glass raschig rings. The total glass content is 24% of the total free volume, and the working volume (115 liters) is 70% of the void volume of the packed tank. This tank receives solution from (1) transfer tank U-20 (Sect. 2.1), and (2) from evaporator R-22. It has a common connection with R-35 to R-26 since both vessels use R-26 for batch transfer to U-20. This interconnection is a potential route for solution with a concentration of 300 g of ^{233}U per liter to flow into R-25 when dissolver S-15 is being discharged into the R-35 vessel; procedures call for water to be kept in R-25 in sufficient quantity to dilute one dissolver batch when dissolver S-15 is operated. There is no hazard in R-26 because it is geometrically safe (i.e., it is made of 3-in. sched. 40 pipe).

Nuclear Safety. -- A fixed neutron absorber is used to achieve principal control in accordance with the ANSI standard.[4] The allowable concentration of 250 g of ^{233}U per liter is based on a calculation made for this specific case, as detailed in Sects. A-1, A-2, and A-4 in the Appendix. The control provided by F(1) and F(2) is administrative, as defined in Sect. 2. Numerous locked valves are used (see Sect. 3.1).

S-20 Dissolver. — This dissolver is of the thermosyphon type designed to dissolve thoria-urania pellets returned from BAPL for recycle. The solution legs are fabricated from 8-in. sched. 40 pipe, and a 10-in.

sched. 40 steam jacket is included on the hot leg. A de-entrainer section made from 16-in. sched. 40 pipe is packed with glass raschig rings containing 40% glass.

The dissolver is charged from the charging box described above. A small crusher in the charging box reduces the size of the pellets to improve dissolution rate. The dissolver basket must be reloaded two or three times in order to keep the dissolver 70 to 100% full of crushed pellets during a dissolution. A measured volume of nitric acid solution will be added initially and allowed to react until the desired end point is reached, leaving a heel of about 40 kg of scrap (2.5 kg of uranium). The batch of solution will then be transferred via gravity to adjustment tank S-2. The maximum uranium concentration attained in the S-20 dissolver is 10 g of ^{233}U per liter, and the minimum Th/U ratio is 17/1. When scrap pellets having a lower uranium content are dissolved, the uranium concentration will be lower and the Th/U ratio will be higher.

If the dissolver solution concentration exceeds 10 g of ^{233}U per liter, transfers to S-2 (feed adjustment tank) must be made in aliquots of ≤500 g of ^{233}U. The proper amount of diluting solution must be added to S-2 <u>before</u> the aliquot is transferred, and S-2 must be sparged for 30 min between transfers of 500 g of uranium.

Nuclear Safety. -- Concentration is used as the principal means of control. The solution composition reached by all scrap pellet mixtures to be received as safe in S-20 geometry is reported in Sect. A-8 of the Appendix. Pure ^{233}U oxide must be excluded. Procedures F(1) and F(2) apply. Secondary control by fixed neutron absorber applies to the large diameter de-entrainer section only.

<u>S-2 Feed Adjustment Tank</u>. — This tank is the vessel in which all solutions to be purified by solvent extraction are collected and made ready to process. It is 48 in. in diameter and has a working volume (70% of total) of 1500 liters. In addition to liquid level and density probes, it is equipped with a sampler and a sparge ring. The bottom section of the tank can be heated by means of an external steam jacket.

Solutions are received from three major sources and three occasional supplies. The major elements are the uranium storage tanks via U-20, the S-20 dissolver, and the R-2 recycle vessel; the ^{233}U concentrations in the latter two are limited to a maximum of 10 g of ^{233}U per liter. Control of concentration in S-20 by selecting the proper dissolving conditions has already been discussed. If a concentration greater than 10 g of U/liter is reached inadvertently, transfer will be made in 500-g aliquots, with sparging. The solutions collected in R-2 are from ion exchange bed elutriation and from uranium recycle from the oxide line. Vessel R-2 is 36 in. in diameter and is filled with neutron-absorbing glass raschig rings. The total tank volume is 1200 liters; of this volume, 24% is occupied by glass, leaving 915 liters void. The working volume is considered to be 70% of the void, or 645 liters. The normal inputs to this tank will be 300 liters per week or less, leaving adequate volume for dilution and mixing in order to reduce the ^{233}U concentration of the tank to 10 g of ^{233}U per liter prior to transfer to S-2. Solutions transferred from the storage tanks via U-20 are physically limited to 2 liters (500 g of uranium, maximum) by the size of U-20. The feed adjustment tank, S-2, is sparged 30 min for mixing after each transfer (from U-20).

The solutions that are occasionally collected in S-2 come from N-7, P-6, and S-12. Tank N-7 has a working volume of 305 liters and is protected by glass raschig rings. There will be sufficient capacity and holding time to ensure that the concentration is 10 g of ^{233}U per liter or less before transfer to S-2. The other two vessels, P-6 and S-12, are condensate receivers having the characteristics listed in Table 1. They normally contain no uranium; however, if analysis of a sample taken before transfer to waste shows that recoverable uranium is present, the solution is recycled to S-2. In each case, time and volume are adequate to ensure that the condensate contains less than 10 g of ^{233}U per liter prior to transfer to S-2.

Required feed adjustments made in S-2 are adjustment of acid concentration and addition of thorium nitrate. The first adjustment requires addition of water or dilute acid to maintain the HNO_3 concentration at 4 $\underline{M}$ or

less. Thorium nitrate is added as required to keep the Th/U ratio at 17/1 or greater; this addition must be made before the uranium solution requiring thorium adjustment is brought into S-2. Concentration adjustments are also made in S-2. In the case of a very low uranium concentration resulting from collection of waste and rinse solutions, it is necessary to boil the feed down to increase the uranium concentration to a maximum of 5 g of ^{233}U per liter. During the boiling procedure, the volume of condensate is accurately measured in a small collection vessel (S-12). This provides good concentration control in S-2.

The most common concentration adjustment in S-2 is a final dilution from 10 g of ^{233}U per liter to 8 g of ^{233}U per liter, which is the feed concentration desired in S-4 in order to satisfy conditions for the operating flowsheet. If it is desirable to operate at a higher feed concentration, the solution containing 10 g of ^{233}U per liter can be transferred without dilution.

Nuclear Safety. -- Nuclear safety is achieved by controlling the concentration in S-2 to $\leq$10 g of ^{233}U per liter; secondary control is achieved by including a soluble neutron absorber (see Sect. A-10, Appendix A). Concentration control is accomplished by administrative procedures F(1), F(2), and F(3), which limit all inputs to a concentration of $\leq$10 g of uranium per liter, or $\leq$500 g of uranium per batch transfer. The soluble neutron absorber, $Th(NO_3)_4$, will always be present in the tank to provide a Th/U ratio of 17/1 or greater.

S-4 Feed Tank. — The S-4 feed tank receives the adjusted feed solution from S-2 and supplies the column feed pump. The purpose of the tank is to maintain a uniform feed composition for several hours of operation while additional feed is made up. S-4 is a 36-in. diameter vessel having a total volume of 1230 liters. Transfer into S-4 from S-2 is made via steam jet, which dilutes the solution slightly below the maximum (10 g of ^{233}U per liter) allowed in S-2. The present flowsheet calls for 8 g of ^{233}U per liter in the feed; the concentration adjustment is made in S-2 before transfer.

Nuclear Safety. -- Concentration is used as the means of criticality control. The only source of solution for S-4 is S-2, which is also limited to ≤10 g of ^{233}U per liter. Administrative procedures F(1) and F(2) apply.

3.2.2 Solvent Extraction - (A)

This system (see Fig. 3) accomplishes uranium purification by selectively extracting uranium into a solvent phase, which is 5% di-*sec*-butylphenylphosphonate (DSBPP) dissolved in diethylbenzene (DEB). Extraction is accomplished by countercurrent contact of the aqueous feed stream and the immiscible organic solution in pulsed columns. The ^{233}U concentration of the entering aqueous stream is ≤8 g of ^{233}U per liter; the concentration of the organic stream is dependent on the relative stream volume rates, with a maximum at 21 g of ^{233}U per liter (which is the saturation concentration for 5% DSBPP).

All of the columns are fabricated from 5-in. sched. 40 pipe and contain enlarged sections at the top and bottom for phase separation and flow smoothing. The enlarged sections are made of 8 in. sched. 40 pipe, except at the top of N-1 column where a 16-in.-diam section is used; all enlarged sections except the one at the bottom of N-1 column are packed with glass raschig rings. The packing density is 24% of the total volume. Packed sections are shown in Fig. 4.

Solution concentrations in the columns are ≤8 g of ^{233}U per liter for all aqueous streams, except in R-1 column, and ≤21 g of ^{233}U per liter for all organic streams. The exception in R-1 column is the stripping solution that results from the transfer of uranium from organic solution. The aqueous-phase concentration increases when the flow of stripping solution through the column is reduced. Process procedures and monitors are devised to detect this condition when it occurs and to give immediate warning.

N-1 Extraction Column. — This extraction column receives three input streams: (1) aqueous feed at a ^{233}U concentration of ≤8 g/liter, (2) the clean solvent stream at a ^{233}U concentration of <1 g/liter, and (3) the scrub stream from P-1 column at a ^{233}U concentration of <1 g/liter. The

two aqueous streams combine in the aqueous waste leaving the bottom of N-1 and entering waste tanks N-4 and N-2; this stream is normally very dilute (i.e., containing <<0.1 g of ^{233}U per liter) and, even during a misoperation, can never exceed the feed concentration of ≤8 g of ^{233}U per liter. The organic stream leaving the top of N-1 column will normally have a ^{233}U concentration of 8 to 10 g/liter and cannot exceed 21 g/liter, which is the saturation concentration for 5% DSBPP in diethylbenzene. This stream proceeds to scrub column P-1.

Nuclear Safety. -- The principal means for achieving criticality control in the column are geometry and limited concentration; the 5-in.-diam vessel is safe at a ^{233}U concentration[3] of 400 g/liter in the nominally reflected condition. Lower values result from close control of the concentration and flow rates of the feed stream and of the incoming organic stream. Secondary control in the upper end section is accomplished by using a glass raschig ring fixed absorber.[4] Based on data for R-25 (see Sect. A.4, Appendix A), the 16-in.-diam section with 24% glass will be safe at least to a ^{233}U concentration of 250 g/liter and the maximum aqueous concentration in the column top would be about 160 g of ^{233}U/liter when the scrub stream consists of water instead of aluminum nitrate solution. This value is calculated from the saturation concentration of organic (21 g of ^{233}U per liter) times the ratio of organic/scrub flow, which never exceeds 8/1 in any flowsheet. Administrative procedures F(1), F(2), and F(3) apply. All column operations are continuously monitored during a processing run.

N-4 and N-2 Waste Tanks. — These tanks are 36-in.-diam vessels which contain 800 liters at 70% of their total volume. No glass rings are used in them since they have no input except from column N-1, and the aqueous waste from that column can never be more concentrated than the feed (≤8 g of ^{233}U per liter). If organic extractant is allowed to flow into N-4 as the result of equipment failure, the maximum concentration possible in the organic phase could not be greater than the aqueous concentration since only a single stage of contacting is available. The waste is caught in

N-4 and periodically transferred to N-2, where it is held for sampling and analysis; if uranium is not present at a recoverable level, the solution is transferred to the waste tank.

Nuclear Safety. -- Concentration is the principal means for achieving criticality control, since all streams entering N-1 are closely controlled. Administrative procedures F(1), F(2), and F(3) apply.

N-23 Tank.— The N-23 is an 18-in.-diam tank which serves as a pulse regulating surge vessel inserted in the organic input line to the column. All of the clean organic stream, which is recycled after stripping, passes through this line and vessel. The stripped organic is collected in tank P-69 and pumped back to N-23; the holdup volume in this tank (P-69) is continuously monitored for density and is sampled and analyzed for uranium every 8 hr when the extraction system is in operation. Any increase in ^{233}U concentration above a normal ≤ 0.01 g/liter results in corrective action.

Nuclear Safety. -- Limiting the uranium concentration is the principal means for achieving criticality control in the organic solution. Administrative procedures F(1), F(2), and F(3) apply.

P-1 Scrub Column. — In this column, the uranium-bearing organic stream is contacted with an aqueous aluminum nitrate solution to scrub thorium and other impurities and thereby further purify the uranium. The column is fabricated from 5-in. sched. 40 pipe and has 8-in.-diam end sections, both of which are packed with glass raschig rings to provide 24% glass in the packed volume. The entering streams are extractant (5% DSBPP in DEB) carrying ≤ 10 g of ^{233}U per liter and basic aluminum nitrate solution. The organic stream leaves the column with the uranium content unchanged, and the aqueous aluminum nitrate picks up a small amount of thorium and other metals. The normal flowsheet poses no criticality problem; however, in the event that the scrub solution is improperly prepared, uranium could be stripped into the aqueous phase. Since the flow rates are different by a factor of 6 to 8, the uranium

concentration in the aqueous phase could increase to 80 g of ^{233}U per liter in aqueous solution during abnormal flowsheet operation.

Nuclear Safety. -- Concentration and geometry are the primary means for ensuring criticality safety in the column proper; the 5-in. column is safe for 400 g of ^{233}U per liter;[3] the maximum concentration is probably 80 g/liter. Secondary control in the 8-in.-diam end sections is achieved by including a fixed neutron absorber.[3] The 24% glass rings used make this section safe for a ^{233}U concentration of >250 g/liter.[4] Administrative procedures F(1), F(2), and F(3) apply.

R-1 Stripping Column. — In this column, uranium is transferred from the organic solvent stream to an aqueous solution, the form most useful for further processing. The flow rate of the stripping stream is set at one-fourth that of the organic stream flow in order to produce an aqueous stream having a useful concentration. For the normal flowsheet, this will result in ^{233}U concentrations of about 31 g/liter and <1.0 g/liter in the aqueous product stream and the solvent, respectively. Loss of stripping solution flow would result in higher concentrations of uranium within the column. To protect against this contingency, the enlarged end sections are packed with glass raschig rings; 24% of the total volume is occupied by borosilicate glass. Special instruments are installed to sense a decrease in strip-solution flow and sound an alarm if the flow drops below 75% of the set point; the density of the aqueous solution in the column is measured continuously, and an alarm will sound in the event that it increases above the desired level. The same information is also derived from examination of the liquid level and density records of the strip collection vessel, R-3.

Nuclear Safety. -- Concentration is the principal means of ensuring criticality safety in the column proper; the column is safe for 400 g of ^{233}U per liter.[3] Administrative procedures F(1) and F(3) are employed to achieve control at ≤31 g/liter. Operation for 36 hr with no product withdrawal would be required to reach the unsafe accumulation level; however, attainment of such a level is not possible because of the mass

transfer characteristics of the system. Secondary control in the 8-in.-diam end section is accomplished by including a fixed neutron absorber. The 24% glass rings used in these sections provide safety to a ^{233}U concentration of at least 250 g/liter, by comparison with R-25 (see Sect. A.4, Appendix). The comments above regarding this concentration also apply here.

R-11 Tank. — R-11 is an organic-stream head tank in which the product stream from the top of P-1 column is accumulated and fed to the bottom of R-1 column. The tank is a 8.25-ft-long section of 6-in. sched. 40 pipe having a volume of 46 liters. The tank normally contains only organic extractant, which cannot have a ^{233}U concentration greater than 21 g of ^{233}U per liter when the DSBPP content is 5%. In the event of power failure, it is possible that aqueous product solution from the bottom of R-1 column might flow into R-11; the normal concentration of the aqueous stream is 31 g of ^{233}U per liter.

Nuclear Safety. -- The principal means of control are geometry and concentration. The 6-in. sched. 40 pipe is safe for a ^{233}U concentration[3] of 100 g/liter, and the entering aqueous solution would contain about 31 g of ^{233}U per liter. Administrative procedures F(1) and F(3) apply.

3.2.3 Solvent Extraction - (B)

The first section devoted to solvent extraction (Sect. 3.2.2) described the contactors and their immediate auxiliaries; this section describes the functions of the product handling system and recovery and recycle of the organic extractant.

The product system collects aqueous solution from the R-1 (strip) column in R-3 tank, from which it is pumped to the evaporator, R-22, for concentration (see Fig. 5). A companion tank (R-2) formerly used for product now serves as a collection tank for recycle solutions.

Organic extractant solution leaving R-1 column has been stripped of uranium; that is, it contains less than 1.0 g of ^{233}U per liter. This solution collects in P-69 vessel and is recycled to column N-1 by a pump. A portion of the organic is removed periodically and sent to wash tank T-20, where it is contacted with an alkaline aqueous solution to neutralize the nitric acid and to wash out certain chemical substances resulting from the reaction of nitric acid with the solvent. The clean organic is returned to tank P-69. The aqueous wash is collected in P-70 for analysis. When the uranium content is low (<0.01 g/liter), the solution is jetted to waste; if recoverable uranium is present, the solution is jetted to rework vessel N-8.

R-3 Product Recovery Tank. — This tank is a 36-in.-diam vessel packed with borosilicate glass rings to a uniform density of 24% glass. The working volume is 645 liters at 70% of capacity. Liquid level and specific gravity instrumentation are provided for good inventory data, and an air sparge is available for mixing solutions. The major input to the vessel is stripped product solution originating in R-1 column and having a ^{233}U concentration of 31 g/liter. The other input is from vessel R-25 in cell 5; the solution to be sent to R-3 from R-25 will usually be the start-up product from the evaporator which is returned for further concentration. It is also possible to transfer product solution from the storage tanks to R-3 via R-25; when this is done, water for dilution is added to R-25 in order to produce the desired concentration for transfer to R-3.

The tank contents of R-3 can be pumped to the evaporator or jetted to R-2 for recycle to solvent extraction; both vessels are safe for the concentration in R-3 (180 g of ^{233}U per liter); calculational verification is given in Sect. A-5 of the Appendix.

Nuclear Safety. -- Principal control is achieved by including a fixed neutron absorber.[4] Glass-ring packing is used to maintain a subcritical condition at a ^{233}U concentration of ≤180 g/liter (see Sect. A-5 of the Appendix). Administrative procedures F(1), F(2), and F(3) apply; under this heading, concentration and transfer procedures are closely controlled.

R-2 Vessel. — Physically, this vessel is the same as R-3 described in the preceding section; however, its functions are different. Inputs include ion exchange bed elutriant, oxide line recycle solution, and off-specification ion exchange product, which have ^{233}U concentrations of 15-20 g/liter, <10 g/liter, and 120 g/liter, respectively. These inputs are combined in R-2 and diluted to a final ^{233}U concentration of 10 g/liter before transfer to feed adjustment tank S-2.

Nuclear Safety. -- Control is achieved primarily by including a fixed neutron absorber.[4] Glass-ring packing is used to maintain a subcritical condition at a ^{233}U concentration of <180 g/liter (see Sect. A-5 in the Appendix). Administrative procedures F(1), F(2), and F(3) apply.

R-22 Product Evaporator. — This product evaporator is used to concentrate solvent extraction product from 31 g of ^{233}U per liter to ≤250 g of ^{233}U per liter. The evaporator body is fabricated from 5-in. sched. 40 pipe and has a 6-in. sched. 40 pipe jacket and an internal pipe coil for heating. The product concentration is controlled by density instrumentation; the feed rate, the boil-up rate, and the reflux rate are controlled by level and temperature instrumentation in the evaporator.

The 5-in. vessel is safe at a ^{233}U concentration of ≤400 g/liter, a value well above the maximum desired product concentration of ≤250 g/liter. There is need for close operating surveillance when the 250-g/liter concentration is approached since the ^{233}U concentration in the receiving tank, R-25, is limited to this value.

Nuclear Safety. -- Measures used for control are geometry and concentration; the 5-in. nominally reflected evaporator is safe for a ^{233}U concentration of 400 g/liter[3] (see also calculated data in Sect. A-12 of the Appendix). Administrative procedures F(1) and F(3) implement concentration control.

Organic Collection Vessel P-69. — This vessel is positioned at the discharge of the strip column. It is 42 in. in diameter and packed with

glass rings to 24% of its total volume. It has a capacity of 450 liters when filled to 70% of packed volume. A companion vessel, P-69FC, is 12 in. in diameter and packed with glass rings; it is used as a flow-rate-checking vessel for setting the organic pumping rate. Both tanks contain DEB dissolved in 5% DSBPP. Under normal operating conditions, the uranium concentration in this stream is very low; however, in the event of total failure of the strip system, the concentration would rise to 21 g/liter. The glass rings will maintain a subcritical condition up to a concentration of 125 g/liter.

Nuclear Safety. -- Measures used for control include concentration and addition of a fixed neutron absorber.[4] Administrative procedures F(1) and F(3) apply.

Solvent Wash Tank T-20. — Organic extractant is "cleaned-up" in this tank. The tank is 18 in. in diameter and has a working capacity of 100 liters. An external pump loop is used to provide a means for contacting the organic and aqueous phases. Batches of 100 liters of organic and 25 liters of aqueous phase are used in each operation. After the contacting period is complete, the phases are allowed to separate and the aqueous phase is withdrawn from the bottom into aqueous holding tank P-70. The normal concentration of the organic stream is <0.01 g of ^{233}U per liter, with a probable maximum of 1.0 g of ^{233}U per liter; thus the maximum uranium content of one batch is 100 g, which is much less than a critical mass. This will transfer to the aqueous phase during washing.

Nuclear Safety. -- Concentration control is the principal method used to ensure nuclear safety of the organic stream. The concentration in P-69 is always known to be safe before a batch is removed to T-20. Administrative procedures F(1) and F(2) apply.

Aqueous Receiving Tank P-70. — Wash solution from T-20 is collected in this tank after use. The description of washing given above indicates that no more than 100 g at a maximum concentration of 4 g of ^{233}U per liter is transferred into this tank at any given time.

Nuclear Safety. -- Concentration control is the principal means for ensuring nuclear safety. Administrative procedures F(1) and F(2) apply.

3.2.4 Thorium Removal

Product specifications require a lower thorium content in the uranium feed solution than can be readily produced by solvent extraction. Consequently, an ion exchange system has been installed in cell 6-7 to provide additional purification. The equipment includes feed tank X-11, ion exchange beds X-13 and X-23, product tanks X-14 and X-15, and an elutriant tank X-17 for accumulation of resin conditioning solutions.

Feed solution is prepared in tank X-11 by pumping solution containing ≤250 g of ^{233}U per liter from storage tanks U-5 — U-10 into X-11, which contains sufficient water to dilute the uranium to 120 g/liter; airlift pumping is used to mix the solution and water. The feed is then pumped through the resin beds X-13 and X-23 and is collected in X-14. From X-14, it is jetted to X-15. After a cycle has been completed, the beds are reconditioned by a series of acid and water rinses (Fig. 6), which are collected in X-17 for analysis and then jetted to R-2 tank (Sect. 2.4.2).

Tanks X-11 and X-15 are actually batteries, each made up from four lengths of 4-in. sched. 40 pipe, to reduce criticality problems. They are manifolded at the top and bottom and use a pipe-to-pipe airlift mixing scheme, as shown in Fig. 7. Each four-pipe battery contains 140 liters when filled to 70% of capacity. When both batteries are filled with solution containing 120 g of ^{233}U per liter, the total uranium content is 33.6 kg.

X-11 and X-15 Tanks. — These tanks are discussed together because they form a linear array, as installed, with 12-in. center-to-center spacing and 12-in. center-to-wall standoff. The interaction and reflection effects were calculated because the mass of uranium is large, but no hazard potential was found. Criticality calculations were made for a ^{233}U

concentration of 250 g/liter since this is the concentration in the storage tanks from which a batch is pumped. The calculational array is described, and results are noted in Sect. A-11 of the Appendix.

Nuclear Safety. -- Geometry is the principal means of control; each 4-in. pipe, as an isolated vessel, is safe for all solution concentrations.[3] As a reflected array, the system is subcritical with a solution containing 400 g of ^{233}U per liter in all eight pipes (see Sect. A-11 of the Appendix). Administrative procedures F(1) and F(2) apply. Secondary control by fixed neutron adsorbers[4] protects against leaks to the floor.

X-14 and X-17 are the auxiliary tanks described. Although their functions differ, they are identical physically. Both are 18-in.-diam vessels packed with glass rings to a uniform density of 34% glass. Each has a working volume of 115 liters at 70% of capacity. Each vessel has a glass pipe section at the top, into which the glass ring packing extends so that any subsidence can be noted visually.

Nuclear Safety. -- The principal means of ensuring nuclear safety control is addition of a fixed neutron absorber.[4] The tanks are safe either separately or arranged in an array at a ^{233}U concentration of >250 g/liter (see Sect. A-11 in the Appendix for details). Administrative procedures F(1) and F(2) apply. Secondary control is achieved by addition of a fixed neutron absorber[4] on the cell floor to protect against leakage.

3.2.5 Waste System

This system (Fig. 8) collects miscellaneous waste solutions for preliminary treatment and analysis. When recoverable uranium is present, the waste solutions are returned to the solvent extraction system for recycle. The units included are condensate receiver P-6, receivers N-7 and N-8 for miscellaneous process solutions, and N-16, which collects drainage from the vessel off-gas header and miscellaneous sources. All vessels are operated in a batch processing method. Adequate time is available for analysis prior to transfer either to waste or to solvent extraction.

P-6 Condensate Receiver. — This condensate receiver is 60 in. in diameter and has a working capacity of 2240 liters. A 6-in.-deep layer of glass rings is used in the bottom of the vessel for criticality control in small volumes. The input streams consist of condensate from the evaporator R-22 and from steam jackets on uranium processing equipment. The most probable source of uranium in P-6 is evaporator start-up, where carry-over may occur. Uranium from this source would not be expected to exceed about 300 g, which is a large part of the evaporator inventory at start-up. As the solution becomes more concentrated, control is more precise. The other possible source of uranium is leakage from the S-15 dissolver into the steam jacket; very little material from this source would be expected because the operating pressure gradient causes leakage of steam into the dissolver when a hole develops.

The glass-ring-treated layer at the bottom of P-6 holds about 200 liters when full. The anticipated maximum ^{233}U concentration in P-6 from the malfunctions described above is <100 g/liter in the first 20 liters, decreasing to <<10 g/liter at volumes of 200 liters and greater. The liquid in the vessel is sampled and analyzed every 8 hr when operations are in progress.

Nuclear Safety. -- Control is effected by the use of a fixed neutron absorber[4] at low volumes and by concentration at volumes greater than 200 liters. Administrative controls F(1), F(2), and F(3) apply.

Tanks N-7 and N-8. — Tanks N-7 and N-8 form a system for handling waste streams from many points in the solvent extraction system. These streams might include mixed organic and aqueous phases. For example, drainage from the pulsed columns at the end of a campaign would be routed to this system.

The input solutions enter tank N-8, which is mounted at a higher elevation than N-7, and the phases are allowed to separate; the tank can be heated to facilitate this separation. When separation is judged to be complete, the aqueous phase is withdrawn from the bottom of N-8, through a small-diameter section equipped to measure the solution density, so that flow can be stopped when organic solution enters the lower section. The

organic phase can be pumped back into the N-1 column head tank for circulation through the system when desired. The aqueous fraction drawn off into N-7 is analyzed and sent either to solvent extraction or waste.

Because of the large number of equipment items that feed streams to this system from sources that may be high in uranium, each tank is packed with glass rings to a density of 24% glass. Each tank is 24 in. in diameter and has a working volume of about 300 liters.

Nuclear Safety. -- Nuclear safety is ensured, principally by addition of a fixed neutron absorber;[4] the 24% borosilicate-glass system is safe for a ^{233}U concentration of 200 g/liter. Section A-6 in the Appendix reports the estimation of 200 g/liter as a safe concentration in vessel N-16, which is identical to tanks N-7 and N-8. Administrative procedures F(1) and F(2) apply.

Vessel N-16. — This vessel is the drainage point for any condensed or entrained liquids that coalesce in the vessel off-gas system serving cells 5, 6, and 7. It also accepts sampler drains, pulser bleeds, and cell floor sump jet discharges. The latter are of particular significance since tanks containing uranium solutions at concentrations up to 250 g/liter are located in the cells, and leakage from them would be collected in the cell floor sumps.

Vessel N-16 is 24 in. in diameter and is filled with glass raschig rings to a density of 24% glass. It has a working volume of 300 liters and discharges to the hot chemical waste system or to rework vessel N-8.

Nuclear Safety. -- Control is effected mainly by addition of a fixed neutron absorber.[4] The vessel is subcritical if filled with solution at a ^{233}U concentration of 200 g/liter (see Sect. A-6 in the Appendix). Administrative procedures F(1) and F(2) are used. Floor sumps that might contain ^{233}U at >200 g/liter are drained to N-16 by manually activated jets; this ensures that the transfer is within the realm of procedural control.

3.3 Conversion System

The purified $^{233}UO_2(NO_3)_2$ solution is converted into $^{233}UO_2$ powder in the conversion system (Fig. 9). The powder is weighed into cans that are packed in shipping containers for weekly transport to BAPL.

All of the required equipment is designed for this specific purpose. Nine enclosures contain the entire process for a batch continuous operation. Enclosures are fitted with transparent windows, gloves, and bag ports to facilitate maintenance and manual process steps (which are relatively few). Mechanical equipment has been used for solids transfer where possible, and the liquid systems are controlled by powered valves operated from the outer face of the enclosures. A high-capacity alpha ventilation system is a part of the containment system.

The major process steps are: (1) precipitation with ammonia to form ammonium diuranate, (2) centrifugation to remove water from the diuranate slurry, (3) microwave drying to dry the centrifuge cake, (4) furnace calcining of the dry diuranate cake, (5) furnace reduction with hydrogen to attain the specified oxygen/metal ratio, (6) controlled cooling and stabilizing, (7) blending to homogenize several batches into one lot, and (8) weighing and packaging for shipment.

Discussions of the system have been divided into two parts, one dealing with liquid-phase processing and the other dealing with operations with the solid-phase material.

3.3.1 Oxide Production Line: Liquid System

The flowsheet operations for this system start with ion-exchange-processed uranyl nitrate solution. The uranyl nitrate is reacted with ammonia to precipitate ammonium diuranate, which is separated from the liquid phase by centrifugation. The supernate obtained on centrifugation is the major liquid phase stream for waste or recycle. This stream and others of similar origin return to the liquid waste handling enclosure.

For ease of consideration, the operations have been separated into those concerned with solid and liquid systems. The liquid system is assumed

to include the starting feed material and all of the aqueous-phase streams produced during processing. The major stream is the centrifuge supernate, which has been separated from ammonium diuranate; a minor amount of uranium will be present. The stream will be acidified to dissolve any suspended uranium precipitate; and, if analysis shows a recoverable amount, the solution will be collected in the waste recovery tank for transfer via pump to vessel R-2 in cell 7. Other streams containing recoverable uranium from enclosure decontamination operations will also follow this course. Those streams that do not contain recoverable quantities of uranium will be accumulated for isotopic dilution with ^{238}U before being released to the waste tank.

The solution processing operations are conducted in three alpha enclosures, which are fitted with windows and gloves for direct handling. These are: (1) the liquid waste handling enclosure (see Figs. 10-12), (2) the precipitation and centrifugation enclosure (Figs. 13 and 14), and (3) the isotopic dilution enclosure (Fig. 15). Each will be discussed separately to describe the equipment and identify any potential hazard.

Liquid Waste Handling Enclosure. — A plan of vessel installation in this enclosure is shown in Fig. 12. The size and the capacity of each vessel, as well as the spatial distribution of vessels, are shown.

Nuclear Safety. -- Criticality control for the enclosure is based on the safety of all the individual vessels contained within the enclosure. Control is maintained by limiting the mass of ^{233}U to 2.0 kg for the total of all solutions inside the enclosure. Secondary control is effected by safe geometry; the slab formed on the enclosure floor by total loss of solution containment is safe. Administrative procedures F(1), F(2), and F(3) apply.

(A) Uranyl Nitrate Metering Vessel. — The uranyl nitrate metering vessel is the feed measurement point for each process batch. The tank is evacuated to bring in solution from vessel X-15 (in cell 7) through transfer pot X-16 (also in cell 7). Nine to ten liters of uranyl nitrate at 120 $\pm$ 10 g of ^{233}U per liter will be received for each batch. The amount transferred is

observed directly in the 4-in.-diam glass tank, and is checked by liquid level instruments in the solvent extraction control room. There are two locked line valves controlling this flow; one key is held by the supervisor in the control room, and the other is held in the basement by the supervisor for the oxide line. The flow can be stopped at any time by either valve.

Nuclear Safety. -- Geometry is used as the means for ensuring nuclear safety of the UNH metering vessel; the 4-in.-diam glass pipe is safe for any solution concentration.[3] Administrative procedures F(1), F(2), and F(3) apply.

(B) Demineralized Water Metering and Nitric Acid Metering Vessels. — Both the demineralized water metering vessel and the nitric acid metering vessel are 4 in. in diameter by 5 ft long; they do not receive uranium solution during process operations.

Nuclear Safety. -- Geometry is the principal means of control;[3] any accidental infiltration of uranium solution would not cause a critical condition. Administrative procedures apply.

(C) Centrifuge Supernate Vessel. — The centrifuge supernate vessel is a 6-ft-long section of 5-in. sched. 40 pipe. The liquid phase that is removed from the ammonium diuranate precipitate is transferred by vacuum into the supernate vessel for acidification and sampling; the nitric acid required is placed in the tank before a batch of supernate is transferred in order to eliminate the possibility of any alkaline precipitate settling to the bottom of the tank. When visual comparison of the acidified solution in the sight glass with a set of color standards indicates that uranium is present, the solution is transferred to the waste recovery vessel. When this comparison does not show uranium, transfer is made to the waste neutralization vessel.

Nuclear Safety. -- Geometry and concentration are the two bases of control; the 5-in.-diam vessel is safe for any solution or suspension up to

a ^{233}U concentration of 400 g/liter.[3] Administrative procedures F(1), F(2), and F(3) apply.

(D) Waste Recovery Vessel. — The waste recovery vessel is a 5-ft-long section of 4-in. glass pipe. It accumulates solution containing recoverable uranium for sampling and analysis. When the analytical data are complete, the solution is pumped to vessel R-2 in cell 7. The primary input to this vessel is acidified supernate from the centrifuge supernate vessel, which is expected to contain about 0.01 to 0.1 g of ^{233}U per liter in normal operation. Any increased uranium concentration would not be hazardous in this vessel.

Nuclear Safety. -- Geometry is the principal means for ensuring nuclear safety; the 4-in. glass vessel is safe for any solution. Administrative procedures F(1), F(2), and F(3) apply.

(E) Waste Neutralization Vessel. — The waste neutralization vessel is a 6-ft-long section of 8-in. sched. 40 pipe. It has an input from the centrifuge supernate vessel and from external acid (or caustic) supplied to facilitate adjustment of the pH of the accumulated solution. The contents can be transferred out to the waste recovery vessel or to the isotopic dilution vessel. The choice of destination is dependent on the uranium content determined from samples removed for analysis.

Only those supernate solutions which contain very little uranium as measured by color comparison will be input to the neutralization vessel. After three or four of these batches have been mixed (and the pH adjusted if necessary), the visual comparison analysis is made on the combined batches. If uranium color is visible, the solution is transferred to the waste recovery vessel for recycle; when no color is evident, the solution is transferred to the isotopic dilution vessel for disposal as waste. In either case, an analysis based on a sample of solution is the basis of transfer inventory; however, in the case of no color, the transfer may be made before analytical results are available.

Nuclear Safety. -- Concentration control is the means by which nuclear safety is ensured. All solutions entering the vessel have been determined to contain <1.0 g of ^{233}U per liter. Administrative procedures F(1), F(2), and F(3) apply.

(F) Enclosure Rinse Vessel No. 1. — The enclosure rinse vessel No. 1 is a 6-ft-long section of 5-in. sched. 40 pipe. Its function is to collect rinse solutions generated during decontamination of the process enclosures with dilute nitric acid. Decontamination is carried out when a process cycle (production of one lot of material) has been completed and the equipment is empty. The purpose of this operation is to maintain a low radiation level in the equipment. Solutions from all areas are received by vacuum transfer; each enclosure is a separate transfer operation. The collected solution is examined for color; and, if any uranium is found, the solution is sent to waste recovery vessel for recycle. When the presence of uranium is not indicated, the solution is transferred to enclosure rinse vessel No. 2. In either case, an analytical sample is taken for inventory purposes.

Nuclear Safety. -- Geometry and concentration are the bases of control; the 5-in.-diam tank is safe at a ^{233}U concentration of 400 g/liter.[3] Administrative procedures F(1) and F(2) apply.

(G) Enclosure Rinse Vessel No. 2. — Enclosure rinse vessel No. 2 is a 6-ft-long section of 8-in. sched. 40 pipe. It is used to hold wash solutions from enclosure rinse vessel No. 1 for an inventory sample and analysis. The value determined is the basis for isotopic dilution of the solutions with ^{238}U before they are transferred to the waste tank farm.

Nuclear Safety. -- Concentration is the principal means for ensuring nuclear safety. Only solutions that are found to be very low in uranium are transferred into this vessel. Administrative procedures F(1) and F(2) apply.

Precipitation and Centrifugation Enclosure. — The ammonium diuranate precipitate is formed in this enclosure (see Fig. 13), and the resulting slurry is pumped to a centrifuge for dewatering. The aqueous phase that is removed is vacuum transferred to the liquid waste handling enclosure, and the centrifuge bowl containing the wet slurry cake is transferred to the microwave oven for drying.

The specific liquid-containing vessels in the enclosure are: (1) precipitation vessel, (2) washdown vessel, (3) centrifuge system, and (4) microwave oven condensate receiver. As in the waste handling enclosure, these will be discussed separately.

Nuclear Safety. -- The control of safety in the enclosure is based on safety of all contained units. Administrative procedures apply; under this heading, a mass limit of 1.5 kg of ^{233}U has been established for the entire enclosure.

(A) Precipitation Vessel. — The precipitation vessel is a 5-ft-long section of 4-in. glass pipe into which the uranyl nitrate feed solution from the UNH metering vessel is received. A pump loop circulates the solution in the tank from bottom to top. Ammonia gas injected into the circulating stream reacts with the nitrate solution to form ammonium diuranate, and a continuous pH monitor in the circulating stream indicates the presence of excess ammonia when precipitation is complete. When it is determined that precipitation specifications have been met, the slurry is pumped to the centrifuge. The glass precipitation vessel makes visual verification of slurry removal possible, and provides an opportunity to inspect for the amount of residual solids. A washdown nozzle in the top of the vessel introduces a demineralized water stream to flush remaining precipitate to the centrifuge. The same nozzle can be used to introduce a nitric acid rinse to remove adherent deposits; this stream is collected separately in the washdown vessel (described below).

Nuclear Safety. -- Geometry is the principal means of control; the 4-in.-diam vessel is safe for any solution or slurry possible in this system.[3] Administrative procedures F(1), F(2), and F(3) apply.

(B) Washdown Vessel. — The washdown vessel is a 4-ft-long section of 4-in. glass pipe used to receive nitric acid wash solutions from the precipitation vessel. This solution is used to dissolve any diuranate spills resulting from removal of the centrifuge bowl liner. The used wash solution is vacuum-transferred to enclosure rinse vessel No. 1, described previously.

Nuclear Safety. -- Geometry is the principal means of ensuring nuclear safety; the 4-in.-diam vessel is safe for all solutions.[3]

(C) Centrifuge System. — The centrifuge system as used here includes the centrifuge, in which the bulk of the water is separated from the slurry, and the supernate collection system, which gathers the mother liquor for return to the centrifuge supernate vessel in the liquid waste handling enclosure. The centrifuge bowl is 12 in. ID and 5 in. inside depth; a Teflon liner basket is inserted inside the steel bowl. This liner contains the slurry cake and is lifted out for transfer to the microwave drying oven. The internal shape of the liner (see Fig. 14) reduces the volume of the bowl and also provides a handhold for lifting. The total volume of the liner is 7.1 liters, which is safe at ^{233}U concentrations much higher than the flowsheet condition and is also safe for containing a double batch of precipitate, provided the system is unreflected. Several precautions are taken to ensure that no reflector is present. One of these is certification of the plastic used in the liner to verify that it is Teflon rather than any hydrogenous material. A second precaution is installation of a system of interlocks which will shut off slurry feed in the event that supernate begins to accumulate in the catch basin surrounding the spinning bowl; these are shown on the operating flowsheet (Fig. 13).

The final element is the centrifuge supernate vessel, a 4-in.-diam catch pot in which the liquid collects and from which a vacuum transfer to the liquid waste handling enclosure is made. The volume in this pot is measured by a pneumatic liquid level instrument, which is part of the interlock system referred to above.

Nuclear Safety. -- Geometry and concentration are used as the bases for control; calculations made for this specific system are shown in Sect. A-13.1 of the Appendix. Administrative procedures F(1), F(2), and F(3) apply.

Isotopic Dilution Enclosure. — This enclosure (see Fig. 15) contains only one tank, the isotopic dilution vessel, which consists of a 6-ft-long section of 8-in. sched. 40 pipe. There are two inputs to this tank from the liquid waste handling enclosure, one from the waste neutralization vessel and one from enclosure rinse vessel No. 2. There are also inputs of $^{238}UO_2(NO_3)_2$ solution and acid or caustic solution as required.

The function of this enclosure is to dilute the ^{233}U with sufficient ^{238}U to achieve a $^{238}U/^{233}U$ ratio of 100/1, followed by pH adjustment, before releasing any waste solutions to the waste tank farm. The weight of ^{233}U discharged to waste is taken from sample analysis and waste volume.

Nuclear Safety. -- Concentration control is used to ensure nuclear safety; any solution transferred to this tank has been previously analyzed and found to contain very low concentrations of uranium. Administrative procedures F(1), F(2), and F(3) apply.

3.3.2 Oxide Production Line: Solids System

Precipitated ammonium diuranate slurry in the form of a wet cake remains in the centrifuge bowl liner after the supernate has been removed. The bowl containing the wet cake is placed in a microwave oven for removal of the remaining water. Each batch contains 1 kg of ^{233}U and is processed in a safe geometrical arrangement. On completion of the microwave drying, the cake (which is now solid) is dumped from the bowl, crushed to destroy lumps, and weighed into numbered furnace boats 4 x 12 x 1.125 in. deep. The charge in each boat consists of approximately 250 g of ^{233}U, forming a layer about 0.375 in. thick. The loaded boats pass through an airlock into an argon atmosphere enclosure in which they are loaded on the furnace conveyor belt. The belt travels through an Inconel muffle, which is heated with externally mounted electrical resistance heaters. Within the muffle there are three zones throughout which both temperature and atmosphere

are individually regulated in order to control the physical and chemical properties of the uranium oxide product. Boats are discharged into an argon atmosphere box at the end of the belt conveyor, and the powder is then dumped into a stabilizer unit. A controlled atmosphere is used during the period in which the product is cooled; finally, the product is equilibrated in air. The powder moves on a vibrator conveyor to a weighing station, which is used for inventory control, and is then bottled in 1-kg stainless steel batch containers. These containers are retained until analytical results for the sample representing the bottled material have been received and found to be satisfactory. The procedure calls for analyzing one sample from each of the five batches; therefore, five cans containing 5 kg of material will accumulate before being dumped into the blender. The final mixed product obtained from a week of production is weighed into shipping cans containing 500 g of ^{233}U per can. These cans are bagged out and transferred to the shipping area (Sect. 3.4).

The entire conversion system is operated on the basis of 1 kg of ^{233}U per process batch, and equipment is sized accordingly. The use of geometrically safe vessels and of mass limits in the wet processing has been discussed in Sect. 3.3.1.

The microwave oven enclosure houses the drying and delumping equipment; it processes one batch at a time. The Teflon liner for the centrifuge bowl, containing the wet cake, is transferred to the microwave oven. The cake is dried while still in the bowl liner; the dried cake is then dumped into the granulator. The bowl liner is returned to the centrifuge for use with the next batch of material, and the solids are loaded into furnace boats in weighed quantities (250 g of ^{233}U per boat). This results in a layer about 0.375 in. thick in the boat.

Nuclear Safety. -- The principal controls in the centrifuge bowl liner are geometry and mass limit (see calculations in Sect. A-13.1 in the Appendix). Secondary control is achieved by administrative procedures which ensure that only a Teflon bowl is used. These include complete material analysis and purchase documentation on the stock used for bowl manufacture. The principal control in the granulator is the mass limit

of 1 kg of ^{233}U as dry oxide per batch. The principal control in furnace boats is geometry; the 3/8-in.-thick slab of oxide is geometrically safe.[3] Administrative procedures F(1), F(2), and F(3) apply; under this heading, a posted mass limit of 1.5 kg of ^{233}U for the oven and loading enclosure is applied.

(A) Furnace Loading and Furnace Enclosures. — The dry powder is processed in these enclosures by heat and gas-phase reaction. The boats previously loaded with powder are air-locked, four at a time, into the argon-filled loading area and are set on the conveyor belt which moves them through the furnace. The boats are positioned sequentially in line on the belt, and the furnace muffle is of proper size and construction to prevent the accumulation of boats at one point. (An obstacle would cause the trays to stand still while the smooth conveyor slipped underneath them.)

Nuclear Safety. -- The control principles are safe slab geometry[3] and exclusion of moderator; no water can enter the furnace muffle. Administrative procedures F(1), F(2), and F(3) apply; under this heading, a ^{233}U mass limit of 5 kg is applied to the furnace (see Sect. A-13.2 for safe mass of an unmoderated oxide powder).

(B) Furnace Unloading and Stabilizer Enclosure. — The furnace unloading and the stabilizer enclosures receive the dry powder in trays from the furnace. These trays are unloaded in batch lots (1 kg of ^{233}U) and fed through the stabilizer, which discharges the product into a stainless steel container in 1-kg batches.

Nuclear Safety. -- Control is primarily ensured by limiting the ^{233}U mass to 1 kg as oxide[3] and by excluding moderators. No water is present in these enclosures. Administrative procedures F(1), F(2), and F(3) apply.

(C) Blending and Weighing Enclosure. — The blending and weighing enclosure houses a V-blender, in which the amount of oxide that is accumulated during one week of production (i.e., 20 kg of ^{233}U) is collected

and homogenized. The uniform material is then weighed into shipping cans (containing 500 g of ^{233}U each) that are bagged out for transfer to the shipping area. Quality control requires that one batch out of every five be sampled and analyzed before any of them is dumped into the blender. This results in a holdup of one to five stainless steel containers with 1 kg of ^{233}U in each, and a final accumulation of all batches in the blender.

Nuclear Safety. -- Nuclear safety in the blender is ensured by limiting the ^{233}U mass limit and by excluding moderators; 22 kg of ^{233}U as oxide is well below the mass limits defined in Sect. A-13.2 of the Appendix. No water is present in this enclosure. Administrative procedures F(1), F(2), and F(3) apply.

3.4 Methods Used for Shipment

The finished oxide product is weighed on an electronic balance and sealed in a screw-top steel can with a rubber gasket. The scale operates a printing device which furnishes a label bearing the can number, the lot number, and weight data; five copies are printed. After the label has been attached, the can is sealed in a plastic bag and bagged out of the weighing enclosure. Two of these cans are placed in an isotope can, which is marked with labels from each shipping can; and two isotope cans are placed in approved shipping containers (DOT* permit No. 5795). Each individual can contains 500 g of ^{233}U; the filling is done automatically and is controlled by the electronic balance. The four cans make the total contents of each shipping container 2.0 kg of ^{233}U, or 10 or 11 shipping containers per lot.

The containers are kept in an air lock separate from the process line for loading. There is a small pass-through door to permit cans to be transferred to the container loading area. The air lock opens to the street to facilitate loading the transport vehicle when the packaging is complete. Captive transportation units under "exclusive use" contract will be used for movement to BAPL at Pittsburgh.

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*DOT — Department of Transportation.

3.5 Loss of Solution Containment

Several tanks used to contain uranyl nitrate solution in various phases of the process have been evaluated and found to be subject to adequate criticality control in normal service. These tanks contain relatively large quantities of uranium, and it is desirable to consider the result of leakage or rupture of these vessels. Since the areas in which leakage from such tanks would collect are of prime interest, the nuclear safety of these areas is discussed below.

3.5.1 Floor of the Center Pipe Tunnel

The floor under the ten uranyl nitrate solution storage tanks is completely enclosed by concrete curbing, and the entire floor and wall of the volume below the tanks is covered with a welded stainless steel liner. This volume is filled with borosilicate-glass packing rings, which occupy 34% of the total volume. The total solution capacity of the packed volume is 6000 liters, whereas the nine tanks, when full, contain only 3600 liters. Thus there is adequate protected space to contain the total contents of the ten tanks.

Two sumps in the steel floor pan are low points for collecting solution. Each sump is equipped with liquid level instrumentation to provide notification (alarm) of leakage. There is a suction tube that terminates in each sump leading to Room 303A, which is adjacent to the vacuum transfer system serving the storage tanks. These tubes are capped but can be connected directly to a vacuum line leading into transfer tank U-20 so that spills to the floor can be recovered (removed to the empty U-tank kept on hand at all times).

Nuclear Safety. -- Control is effected principally by the use of fixed neutron absorbers.[4] Administrative procedures F(1) and F(2) apply. The important features are the leak alarm and a preplanned procedure for recovering leakage in this area.

3.5.2 Floors of Cells 5, 6, and 7

The floors of cells 5, 6, and 7 are covered with welded stainless steel liners, which extend up the side walls for several feet. Each cubicle of each cell has a floor drain, and the floor in the cubicle is pitched to that drain. All of the drains in each cell are connected (below the floor level) to a common sump that is 10 in. in diameter. These common sumps do not drain but, instead, are equipped with level-indicating instrumentation and a steam-operated jet for transfer of any liquid accumulation to vessel N-16. The locations of drains and sumps and the pitch of the floor in each area are shown in Figs. 16-18. The volume of liquid contained in each area was calculated for several pool depths, using cell dimensions shown in these figures. These volumes are recorded in Tables 2, 3, and 4.

Table 2. Floor Volume vs Maximum Pool Depth for Cell 5

Cell Area[a]	Volume (in liters) at Maximum Depth			
	1 in.	2 in.	3 in.	4 in.
E	2.4	19	65	155
D	4.9	39	131	311
C	1.2	9	30	71
A and B	7.5	58	197	467
Total	16.0	125	423	1004

[a]Refer to Figure 16.

Table 3. Floor Volume vs Maximum Pool Depth for Cell 6

Cell Area[a]	Volume (in liters) at Maximum Depth			
	1 in.	2 in.	3 in.	4 in.
A	0.3	8	38	103
B	0.3	8	38	103
C	0.3	7.5	35	96
D	3.5	28	94	222
E	2.6	21	70	165
F	0	0.3	9	40
G	0	0.4	10	47
Total	7.0	73.2	294	776

[a]Refer to Fig. 17.

Table 4. Floor Volume vs Maximum Pool Depth for Cell 7

Cell Area[a]	Volume (in liters) at Maximum Depth			
	1 in.	2 in.	3 in.	4 in.
R-1 A	2.2	18	60	142
R-2 C	2	16	53	126
R-3 D	2	16	53	126
D-60 E	2.8	22	74	176
P-70 B	2.5	20	66	157
Open F	5	39	131	311
Total	16.5	131	437	1038

[a]Refer to Fig. 18.

From examination of the volumes shown, it is clear that appreciable volume releases in the cell would result in maximum pool depths greater than a safe slab thickness. Borosilicate-glass raschig rings placed in the low points were chosen as the means for providing criticality control. Some preliminary calculations were made to determine how much protection "a reasonable amount of glass" would afford. (A reasonable amount of glass is a quantity that does not create a significant problem in maintaining use of the floor surface in the cell.) It was determined that a sufficient number of rings to treat the 3-in.-deep (maximum) pool would cover about three-fourths of the cell floor; this makes it possible to use a grating bridge across the glass rings and also to use the floor surface at the edges. A floor coverage of 14 boxes of glass rings (134 liters of glass) was decided on for each cell, distributed as indicated in Figs. 16-18.

The resulting volume relationships, shown in Table 5, were used in plotting the curves shown in Fig. 19. These curves illustrate the thickness of the solution layer that would be formed above the top of the glass-ring-packed layer as 60 kg of uranium was progressively diluted. The curves assume uniform dilution; in the most probable real case, dilution would be the result of sprinkler system operation. In such an instance, the top layers would be more dilute and the solution in the glass-packed section would be the most dense. This situation is not less safe than the idealized case. The volumes of solution (∿25 liters in each case) in the ring-packed sumps are not included in the tables or in the curves representing thickness.

Nuclear Safety. -- Nuclear safety is ensured principally by the addition of fixed neutron absorbers.[4] Administrative procedures F(1) and F(2) apply. Transfer from the cell floor to N-16 tank for recovery is made via manually operated jets; this allows supervisory control of the time required and quantity transferred.

3.5.3 Floors of Enclosures in the Conversion System

There are three enclosures in the conversion system that contain uranium solutions which might possibly be released to the floor. Each enclosure has a sump at the low point which is drained by vacuum to the

Table 5. Solution Capacities of Cells 5, 6, and 7

Pool Depth	Volume Contained (liters)		
	Cell 5	Cell 6	Cell 7
Void volume at 3 in. level	399	290	437
Volume of rings	134	134	134
Solution volume at 3 in. level	265	156	303
Solution volume at 3.5 in. level	499	326	560
Solution volume at 4 in. level	832	553	904
Volume increase per inch above 4 in. level	713	651	779

enclosure rinse vessel No. 1, a 5-in. sched. 40 pipe vessel in the liquid waste enclosure.

The three enclosures are: (1) precipitation and centrifugation, (2) liquid waste handling, and (3) isotopic dilution. In each instance, the capacity of a safe slab thickness is much greater than the total uranium that could be contained in the enclosure. The volumes, maximum capacities, and maximum mass limit for each enclosure are shown in Table 6. The maximum capacity is calculated on the basis of a 4 in.-thick-slab, which is safe for 30 g of ^{233}U per liter when fully reflected. The concentration of 30 g of ^{233}U per liter is used because this is the most restrictive concentration on the slab thickness curve shown in ref. 3; a higher or lower concentration would, for the same mass of uranium, result in a more safe condition.

Nuclear Safety. -- The primary control is effected via slab geometry. Secondary control is accomplished through administrative procedures F(1) and F(3).

Table 6. Volumes, Maximum Capacities, and ^{233}U Limits of Enclosures

Enclosure	Floor Area (ft^2)	Volume of 4-in.-thick Slab (liters)	Total Uranium Content[a] (kg)	^{233}U Mass Limit (kg)
Precipitation and centrifugation	8.25	77.9	2.3	1.5
Liquid waste handling	35	330	9.9	2.0
Isotopic dilution	12	113	3.4	0.2

[a]At 30 g/liter.

3.6 Equipment Interactions

Cells 5, 6, and 7 contain all of the vessels and columns used in the purification systems, and also several that were used for other operations but are now in standby. The possibility of interaction between vessels containing major amounts of uranium was considered, and a calculation was made for the grouping containing the largest quantity in close array. The array was the full set of ion exchange tanks X-11, X-15, X-14, and X-17; no problems were found. The calculational data are reported in Sect. A-11 of the Appendix.

A check of all cell vessels was made to verify that those items not in use are completely separated from the working system, through both solution and off-gas piping connections. The report of findings is shown in Figs. 20-22. Figures 20-25 show the spatial distribution of all equipment in the cells.

4. PROCESS PROCEDURES

This section describes the procedures to be followed in conducting the operation of the process. Each unit operation proceeds according to tested run sheets which have been drafted with consideration of the safety of property and personnel in mind. In addition, materials accountability procedures that ensure control of the movement of fissile materials from the time of receipt until transfer as product and/or nonrecoverable waste are described. A discussion of radiation surveillance and an explanation of a procedural requirement (i.e., that both the equipment design and the plans for administering the operating facility be reviewed, prior to beginning operations, by an independent group of scientists and engineers who are competent in matters of nuclear safety) are also included.

4.1 Run Sheets

Each unit operation (see Appendix C for listing) will be conducted according to tested run sheets, which include check sheets listing the steps that are to be carried out sequentially in the operation. Written into these run sheets are the precautions to be taken, the observations and data to record, and acknowledgments (by signature) of completion or authorization (by supervisors) to proceed further. The run sheets, which comprise a section of the Operating Manual for the LWBR Program at ORNL, in which detailed descriptions of equipment, instrumentation, process chemistry, safety procedures, operational procedures, etc., are included, are issued to operating personnel prior to each run as one item in a "Run Condition Folder." A typical run sheet is shown in Appendix C. Other items contained within the folder are: (1) Run Number, (2) Objectives of the run, (3) Amount and type of material to be processed, (4) Source of material to be processed, (5) Run conditions, (6) Deviations from standard operating procedures or flowsheets, (7) Disposal of product wastes, (8) Copies of (or references to) auxiliary run sheets used in preparing reagents, testing equipment, cleanup, sampling and sample handling, record keeping, etc., (9) Abnormal building or equipment conditions, and (10) Data sheets or tables.

The Run Folder is assembled with concurrence of the Project Leader, the Building Supervisor, and the Technical Group.

4.2 Accountability

4.2.1 Responsibility

Overall responsibility for maintaining a strict accountability program for the LWBR Support Program rests with the Technical Group. It is their responsibility to maintain the records, evaluate the analytical data, initiate the accountability reports, be cognizant at all times of the distribution of all accountable material, and keep the Project Leader informed.

4.2.2 Verification of Material Received in Shipments

The quantity of ^{233}U in all material received or shipped will be determined by applying the data obtained from the analyses of samples of homogeneous measured lots. In the case of solutions, these data include weighing lots of each solution and applying the analytical data, that is, uranium concentration and specific gravity (Sp. G.). In the case of $^{233}UO_2$ powder product, the analytical data (i.e. the uranium content in wt %) is applied to the weight of the lot.

4.2.3 Measurement of Plant Input

The material input to the plant consists of the following:

(1) Uranyl nitrate ($^{233}UO_2(NO_3)_2$) which is stored in the pipe tunnel storage tanks.

(2) Uranium oxide ($^{233}UO_3$) powder, which is stored in the wells in the penthouse.

(3) Thoria-urania scrap (ThO_2-$^{233}UO_2$) in the form of pellets and powder.

Additionally, two other internal sources of material will be processed. These will be the recycle material that comprises off-specification product, and waste solutions containing recoverable quantities of ^{233}U.

Uranyl Nitrate Stock Solution. — The $^{233}UO_2(NO_3)_3$ stock solution is of adequate purity to be suitable for direct processing in the ion exchange (I.E.) system. The input measurements of this material will be conducted in the I.E. feed tank (X-11). After the material has been homogenized by circulation and sparging, samples will be removed for analyses. The resulting analytical data will be applied to the volume of solution in the feed tank (determined from liquid level (LL) and Sp. G. measurements referred to tank calibration charts). The results may be compared with the depletion volume in the storage tank from which the stock solutions were obtained. The uranium contents of these stock solutions are always known from previous measurements, since they are determined in the same manner as in the I.E. feed tank.

Uranyl Oxide Stock. — The $^{233}UO_3$ stock is also of such purity that it may be processed directly in the I.E. system; however, it must be converted into nitrate solution first. In conversion to nitrate solution, the contents of the required number of powder containers (the identities of the cans of powder are known from our shipping receipt records) will be transferred to the Oxide Dissolver (S-15) and dissolved completely in nitric acid. Care will be taken to effect a complete transfer of the powder contents by brushing and by visual inspection of the cans. After dissolution, the dissolver solution will be transferred (via rinses, etc.) to a calibrated collection vessel (R-35), where it will be thoroughly mixed by sparging and samples will be taken for analyses. The total amount of dissolved ^{233}U will then be determined based on the analytical results and tank volume measurements (made as described above), and taking into account any "heel" that was initially present in the collection vessel. The quantity of ^{233}U present in the dissolver can then be compared with the quantity appearing on the shipping-receipt records for confirmation.

Thoria-Urania Scrap. — The thoria-urania scrap material must be processed through the solvent extraction system to separate the ^{233}U from gross thorium that is present. It is expected that other metallic impurities will also be separated. The scrap from identified cans in which

it was shipped will be crushed and then dissolved in nitric acid catalyzed with fluoride. Dissolution will take place in a dissolver designed for this purpose (S-20). During dissolution, which will progress at a relatively slow rate, it will be the practice to maintain an undissolved inventory of crushed ThO_2-$^{233}UO_2$ pellets in order to obtain a reasonable production rate. When the dissolver solution reaches the desired composition (<10 g U/liter, Th/U $\geq$ 17), as indicated by the instrument for measuring Sp. G. and by analysis of samples, it will be drained to a calibrated collection vessel (S-2), where, after mixing thoroughly, samples will be withdrawn and volume measurements will be made from LL and Sp. G. data referred to the tank calibration chart (as discussed above). By applying the results of analyses of these variables to the volume of solution in the collection vessel, the quantity of ^{233}U in each dissolver solution will be determined. As a general rule, the quantity of ^{233}U originating in specific cans or containers will not be verified; instead, only the ^{233}U content of a particular shipment will be verified by starting with the empty scrap dissolver and ending with cleanout-dissolvings (one or more), measuring the ^{233}U content of each dissolver solution (including cleanout), and comparing the total ^{233}U in these solutions to the amount indicated in the shipping documents for the scrap charged.

4.2.4 Measurement of Plant Output

Plant output will consist of: (1) finished $^{233}UO_2$ powder meeting all specifications, and (2) waste discharges to permanent waste disposal facilities.

Finished $^{233}UO_2$ Powder Product. — The $^{233}UO_2$ powder produced over a period of a week will be accumulated and blended to obtain a homogeneous lot. Representative samples will be removed from each lot of powder and submitted for analysis. The powder will then be packaged in cans, each containing about 500 g, and weighed. By applying the results of chemical analyses to the weight of the dry powder, the total amount of ^{233}U present in the lot will be determined.

Waste Discharges. — Waste effluent from the facility will be converted into homogeneous aqueous solution and then collected and sampled in calibrated

tanks. The volumes of these tanks are determined from LL and Sp. G. measurements referred to the calibration chart. The total ^{233}U content of the solutions will then be determined by applying the analytical data to the measured volumes. These data will also form the basis for poisoning the waste solution with 100 parts of ^{238}U per part of ^{233}U prior to discharge to the hot chemical waste system.

The solid wastes (containing ^{233}U) that consist of contaminated swabs, plastic, etc., will first be estimated from material balance measurements. In the event that recoverable quantities of ^{233}U appear to be present, an effort will be made to recover as much as is practical within the limits of safety and economics.

In any event, the enclosures in the $^{233}UO_2$ powder facility will be swept clean, first by vacuuming and then by swabbing at weekly intervals (i.e., after each lot of blended $^{233}UO_2$ powder has been produced). This is necessary not only to maintain the radiation level (due to ^{232}U decay daughters) at the minimum but also to minimize and control unaccounted for accumulations of ^{233}U in the enclosures.

4.2.5 In-Process Inventory Measurements

The quantity of ^{233}U present in the process feed to the solvent extraction and ion exchange systems can be determined from current volume measurements and previously obtained analytical data. The ^{233}U content of liquid product collection tanks can be estimated with a fair degree of accuracy (within about ± 5 to 10%) from the LL and Sp. G. indicated on the instruments by means of equations (or curves) relating Sp. G. to uranium concentration and the tank calibration charts. For more precise measurements, the results of chemical analyses of the solution for uranium concentration would be used instead of the empirical relationship described above.

Liquid waste collection tank inventories will be determined by sampling the contents and applying the analytical results to measured volumes.

Estimates of the inventory of ^{233}U within each enclosure of the $^{233}UO_2$ powder facility during the processing interval will be based on the technique of batching measured volumes in a calibrated tank of restricted volume;

the ^{233}U content of each batch of solution will be determined by applying the results of chemical analyses of the lot of nitrate feed solution (which will be of sufficient size to supply all of the $^{233}UO_2$ produced for a "product lot") to the batch volume. As each batch of nitrate feed solution is initiated, a batch inventory card will also be initiated. The batch inventory card, which includes a check-off space (checked by the technicians as each step is completed) for each step in the powder conversion process, will accompany the batch (only one will be in an enclosure at any given time) as it moves as a discrete entity through the system until it reaches the furnace loading enclosure. Here, each batch is subdivided into four identifiable subbatches of equal size and loaded onto the conveyor belt that moves through the furnace. The inventory of the furnace is approximately equivalent to one day's production; therefore, the batch inventory cards will accumulate at the furnace enclosure. As the identifiable subbatches emerge from the furnace and the powder is screened and stabilized, each day's production will be accumulated into a sublot which will be held, pending evaluation of its quality based on analysis of samples. As each sublot is evaluated, it is collected in the blender. As each batch of $^{233}UO_2$ powder is accumulated into a sublot, the corresponding batch inventory cards are held at the blending and weighing enclosure. When the sublots are transferred to the blender in accumulating a lot of powder, the batch inventory cards will be collected; in addition, a second inventory card, a "Lot Inventory Card," will be initiated. The latter card will identify each batch of $^{233}UO_2$ powder and the maximum ^{233}U content can be estimated from these data. This card also contains space for recording the identity and the weighing data for each product can by the technician; in addition, it provides space for showing the identity of the isotope can (and its contents), the position of each isotope can in the shipping container (DOT permit No. 5795), and the serial number of the shipping container. The "Lot Inventory Card" will accompany the lot into the storage area.

The ^{233}U content of each primary can of finished $^{233}UO_2$ powder will be determined from chemical analyses of representative samples taken from each blender lot (Lot Sample) and the net weight of the oxide present.

Material balances will be determined for each lot of feed material processed. Since one lot of nitrate feed produces one lot of blended $^{233}UO_2$ powder, this will be a straightforward accounting for the feed material in the lot of product and the wastes discharged to the tank farm plus any recycle material accumulated from cleanout operations.

Internal material balances around the dissolvers will be calculated to detect, recognize, and account for undissolved materials. In this practice the ^{233}U content of the input material will be that determined from shipping receipts and from records listing each container whose contents were charged to the dissolver.

Internal material balances around the feed preparation system (tanks S-2 and S-4) will be calculated for each batch of feed material to determine plant input (corrected for recycle material, determined separately) and to ensure nuclear safety control.

Similarly, internal balances around the solvent extraction system will be calculated, based on feed input and output measurements determined as described above. During the progress of a run, a rough balance will be made once each 8-hr shift by the shift supervisor from volume changes in the feed, product, and waste tanks as determined from data displayed on the control panel instruments, coupled with the known composition (from analyses made at the beginning of the run) of the feed, past experience in evaluating conditions of the system as indicated on the control panel in terms of equilibrium, probable uranium content of waste and product, possible leakage, possible diversion of process streams, and the previously known (by measurement) uranium holdup of the system. This practice allows early detection of abnormal conditions or malfunctions and, in addition, provides a basis for immediate corrective action.

A programmatic material balance for the ^{233}U will be calculated by comparing the material accounted for with that indicated on shipping receipts. Included in the balance will be a determination of the unaccounted for material with some evaluation of the associated uncertainty.

4.2.6 Surveillance

The LWBR support program facilities are located in Building 3019 within the confines of the security boundary of ORNL. The ^{233}U feed stock materials are stored in tanks or wells. Admission to these areas is limited to authorized personnel who possess keys to the security locks on the access doors. The dissolution, solvent extraction, and ion exchange equipment is located in the cells; access to and operation of these areas require possession of keys to security locks on the cell doors and to the locked valves in the processing equipment. The $^{233}UO_2$ powder facility is located in the basement of Building 3019; access to this area, limited to authorized personnel, requires possession of the key to the security locks on the access doors.

All the keys to the locks in the process valves and to the access doors are kept in a locked cabinet located in the control room; access to keys to this cabinet is limited to (1) the Project Leader, (2) the Chief of Operations, and (3) the Duty Shift Supervisor.

The facility will be occupied and under the direct surveillance of the duty technicians on weekdays on a round-the-clock basis; on weekends and holidays, the access doors to the facility will be kept locked and the building will be patrolled by members of the ORNL Guard Department.

4.3 Radiation Survey

In addition to the continuous facility monitoring system described in Sect. 6, Health Physics personnel assigned to the area will routinely monitor the unlimited access areas (offices, conference room, lunchroom, analytical laboratories, change house, etc.) for transferrable contamination and radiation. In addition, they will survey all shipments, samples, and containers of waste for surface contamination and radiation to ensure compliance with applicable ORNL and other regulations. Personnel entering and leaving contaminated work areas are routinely monitored individually for contaminated clothing, hands, and feet by Health Physics personnel using portable survey instruments.

4.4 Nuclear Safety Reviews

While this report documents an analysis of the entire array of the facilities involved in the LWBR Support Program, individual units of equipment (and in some instances, subarrays of equipment) have been reviewed by the Criticality Safety Review Committee. Table 1 gives an indication of the extent to which this has been done.

5. OPERATING SAFEGUARDS

5.1 Operating Organization

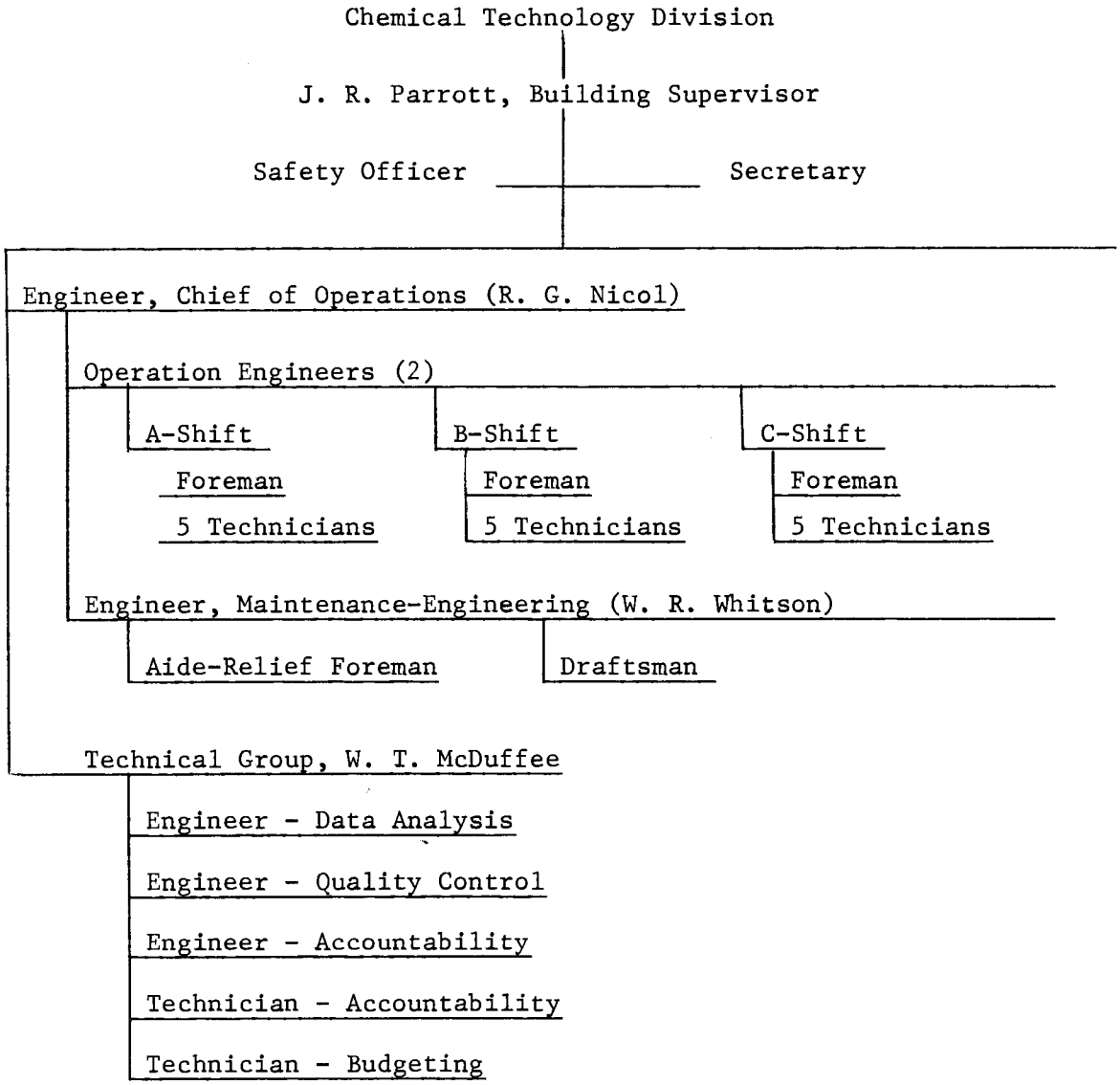
The responsibility for the safe operation of the entire facility rests with the Pilot Plant group, which also operates the cell bank and all related equipment. Building service equipment, including all ventilation systems, is operated and routinely checked by this group. It is their responsibility to see that the required services are operating normally for the supporting groups in the laboratories and other areas of the building.

The operating staff consists of 9 engineers, 4 foremen, and 18 technicians, as shown in Table 7. The operation continues on a 5-day week, 24-hr/day schedule with a shift organization, composed of a foreman-technician and five technicians on each shift, that is supported by technical and engineering groups (including a Building Safety Officer). Craft support in maintenance and repair activities is supplied by the engineers from the Instrumentation and Controls and Plant and Equipment Divisions; health and safety support is received from the Health Physics Division.

5.1.1 Responsibilities

Chief of Operations. — The shift operating group reports to the Chief of Operations. In addition to supervising the shift groups, he has the primary responsibility for administrative control and supervision, with the assistance of a Maintenance Engineer, of all maintenance activities in the building.

Table 7. LWBR Organization



Shift Supervisor. — The Shift Supervisor on duty is responsible for all current operations in the process area, including the determination that any maintenance to be started can be done safely and will not interfere with any operation in progress. He must be aware of all cell bank and building service equipment operations that are under way, and he has the authority and responsibility to stop or change any of these operations.

Building Safety Officer. — The Building Safety Officer is responsible for all safety matters in the building, including zoning regulations, emergency manuals and procedures, safety training, and other similar functions. In addition, he is responsible for reviewing and approving all nonroutine maintenance requests and procedures, as well as radioactive material transfer procedures.

Technical and Engineering Groups. — These groups review and analyze all data generated in the process operations and determine process conditions for all runs.

Building Supervisor. — The above groups report to the Building Supervisor, who has overall responsibility for the operation of the entire facility.

5.1.2 Training Programs

A full-time training session will be held at the beginning of the operations to fully acquaint all personnel with the facility, standard procedures, processes, and safety matters (see Appendix B for a list of topics). This will be followed by a six-month period of check-out operations in which all personnel are trained in their responsibilities. Future training is the responsibility of each shift foreman; process matters are coordinated by the Chief of Operations, and safety training is organized by the Building Safety Officer.

5.2 Standard Operating Procedures

Although all personnel have been trained so that they are completely familiar with the operations they will be performing, all operations will be done by following a detailed procedure and checklist.

5.2.1 Checklists and Run Sheets

All operations will be conducted according to tested, written operating procedures that provide all information required for the safe operation of the building, such as process chemical information, radiation safety information, and general building safety rules. These will be subject to continued review by the Chief of Operations and the technical and engineering groups. Supplementing this is an Emergency Manual, which contains the procedures for handling abnormal conditions and emergencies. One of the emergency features described in the manual is a "scram" switch, which is a device for shutting down all operations connected with solvent extraction from a single point. This switch interrupts electric power circuits and also operates solenoid valves in the air supply lines to shut down all equipment in a proper state. The protected items include the column, the feed make-up, and the product recovery systems. The specific supplies of importance to criticality are the steam to the two dissolvers (S-15 and S-20), steam to the product evaporator R-22, and steam to the heating jacket of the feed adjustment tank S-2.

Detailed run sheets, which include detailed step-by-step checklists listing every operation (such as each valve to be operated) in sequence and specifying the exact conditions, flows, chemical compositions, etc., to be followed in a specific run are also provided. These are reviewed prior to issuance by the Chief of Operations; concurrence is obtained from the engineering and technical groups.

5.2.2 Building Log

All building service equipment, including ventilation systems, air compressors, water systems, etc., are routinely inspected at the beginning of each shift. Operating conditions are logged and compared with the

normal. This log is checked by the Shift Foreman, and any abnormal condition is immediately investigated.

5.2.3 Transfer of Information Between Shifts

The Shift Supervisor is responsible for maintaining an accurate and concise record of operation in the shift log. This continuing log is the primary means of transmitting information between shifts. An overlap period of 12 min by the nontechnical people provides an opportunity for smooth switchover even for a continuous operation. Operations are not normally shut down at the end of a shift. The supervisors normally discuss the operation for about 30 min at the shift switchovers.

5.2.4 Procedure Changes

As the processes change, procedures will be constantly revised to bring them up to date, to refine the operations, and to correct errors as they are noted. The Chief of Operations is responsible, with the concurrence of the Building Supervisor, for all changes in operations and the necessary revisions to the run sheets and checklists. Certain changes are appropriately reviewed by the Safety Officer and the area Health Physics representative.

5.3 Maintenance Procedures

All maintenance procedures throughout the building are done under a system designed to ensure complete safety at all times. Appropriate individuals will approve each step and closely supervise the actual maintenance operations.

5.3.1 Work Request System

Maintenance work may be requested by many people in the building. These requests are submitted to the Maintenance Engineer, who reviews them and prepares the preliminary information relative to the work requested for the Plant and Equipment Division craft forces, who do the work. At this stage the job is reviewed by the Chief of Operations, who determines

whether the job should be done, and, if so, what priority it should have, what safety reviews (if any) are required, and when the job should appropriately be started. The approved request is then transmitted to the Plant and Equipment Division field engineering forces, who complete preparations to do the job. When the preparations are finished, the Plant and Equipment Division foremen review the job with the operating shift supervisor on duty, who ascertains that the job can be done safely at that time. He makes sure that valves and electrical breakers are appropriately tagged out and that any other preparations, including completion of a radiation work permit before giving approval for the job to proceed, are completed. Similar approval is made at the beginning of each new shift in the event that a job carries over to the next shift.

When the work has been completed, the Plant and Equipment Division foreman informs the operating shift supervisor of anything incidental to the work request which might affect the operations.

While some minor variation of the above described sequence may occur, the key steps--review and approval by the Chief of Operations and final endorsement by the Shift Supervisor--are mandatory.

5.3.2 Detailed Procedures

Most maintenance jobs are done from instructions given by the Maintenance Engineer, plus auxiliary sketches and drawings. However, more detailed procedures are required for in-cell or otherwise hazardous maintenance work. Each specific job must be reviewed and approved either by the Chief of Operations or by the Building Supervisor prior to starting the job.

Maintenance involving removal of equipment from the cell bank, and other in-cell maintenance where the cell is to be opened, will be done according to detailed step-by-step procedures.

6. ALARM SYSTEMS

The alarms incorporated into the Building 3019 monitoring system are described in detail in ref. 1. Pertinent sections relating to the nuclear safety of the building are repeated here for the sake of completeness.

6.1 Coincidence Alarm Systems

The coincidence alarm system receives signals from radiation, air, and alpha air monitors, and from neutron detectors placed throughout the building, including the cells, the secondary containment area, and the operating and office areas. Locations of the detectors are given in Table 8. The signals from the detectors are displayed as individual audio-visual alarms (lights, buzzers) on the coincidence panel of the Emergency Panelboard; most of these are arranged in groups of three of more. If two or more signals are received from any specific group of detectors (indicating the possibility of a break of containment or a nuclear excursion), an audio-visual "coincidence alarm" is displayed in addition to the specific alarms of the individual detectors. A coincidence alarm automatically causes the building to be placed in the containment condition. If the coincidence alarm has resulted from the detection of two or more "low level" radiation or contamination conditions, the building evacuation alarm will not sound. However, if the coincidence alarm stems from detection of "high levels" of radiation and/or contamination, the building evacuation alarm will sound. The indicator lights on the coincidence panel are duplicated on the secondary emergency panel located outside the north entrance of the building.

6.2 Fire Alarms

Heat detectors described in the Safety Report[1] generate a signal which activates the fire fighting system strategically located in the cells (fog nozzles). [It should be noted here that, upon arrival at the building, Fire Department personnel may manually restrict or shut off the flow of water to any sprinkler network, depending on their assessment of the situation. Normally, the flow of water would be turned off after about 10 min and the conditions reevaluated.]

Table 8. Facility Contamination and Radiation System

Monitor	Location	Coincidence Group
	Facility Contamination System	
CAM[a]	East control room	E
CAM	West control room	E
CAM	East penthouse	E
CAM	West penthouse	E
CAM	HRLAF east bay	F
CAM	HRLAF west bay	F
CAM	Hall, outside room 209	F
CAAM[b]	HRLAF	G
CAAM	East control room	G
CAAM	Penthouse, west	G
CAAM	Penthouse, east	G
	Facility Radiation System	
Gamma Monitor		
Monitron[c]	Pipe tunnel, west	A
	Basement	A
	Plenum area, west	A
	Sample gallery, west	A
	Pipe tunnel, east	B
	Cell 2	B
	VPP makeup area	B
	Penthouse, east	B
	Pipe tunnel, center	C
	PRFR makeup area	C
	Penthouse, west	C
	Penthouse, center	C
	Cell 4, second level, N.	None
	Cell 4, second level, S.	None
	Cell 4, first level	None
	Room 211	None
	Room 303A	None
Neutron Detectors[d]	Penthouse	D
	HRLAF	D
	East control room	D

[a]Constant air monitor, beta-gamma.

[b]Constant alpha air monitor.

[c]Monitors are present at each location shown in this section of table.

[d]Neutron detectors are present at each location shown in this section of table.

When triggered, the sprinkler system sounds an alarm outside the building. This alarm is repeated in the Laboratory Fire Alarm System and, in effect, summons the Fire Department. Detectors in the supply lines to the cell manifold of the fog nozzle system (normally dry) sound an alarm on the Emergency Panelboard and display an indicator light (and sound an alarm) showing the location of the fire.

7. EMERGENCY PLANS

Plans for combating and minimizing the effects of emergencies that might occur in Building 3019 are described in this section. These plans constitute a part of, and are coordinated with, the ORNL Emergency Planning.

In general, the plans for dealing with emergencies are given in Sects. 2 and 3 of the Building 3019 Emergency Manual. The personnel currently assigned to the responsible positions will be posted in Building 3019 and will have received (in training sessions) instructions as to what action to take.

The alarm system in Building 3019 was described in Sect. 6. The alarms, in conjunction with the indicator lights displayed on the Emergency Panelboard, constitute the data needed for determining the nature of the emergency (i.e., the occurrence of a nuclear excursion, a fire, a mechanical failure of a ventilation system, a breach of containment, etc.). The Building Supervisor and the Shift Supervisors will be trained in evaluating the information displayed on the Emergency Panelboard and in determining what emergency actions to undertake as a result. Particular emphasis will be placed on the "coincidence alarms" and the significance of the data displayed for each type of credible emergency.

Emergency Procedures. — The emergency procedures developed here stem from the philosophy that maximum effort will be exerted toward eliminating and minimizing injury to personnel and that property damage will be considered next. Thus, then, has led to the following approach:

(1) Sounding the alarm.

(2) Exiting from the building as rapidly as possible.

(3) Obtaining assistance as soon as possible.

Assessment of the emergency and the approach to be taken toward combating it are determined after help arrives.

Implementing the approach above, a brief emergency procedure for Building 3019 has been prepared and distributed to all personnel assigned to Building 3019. A copy is included in Appendix D.

8. REFERENCES

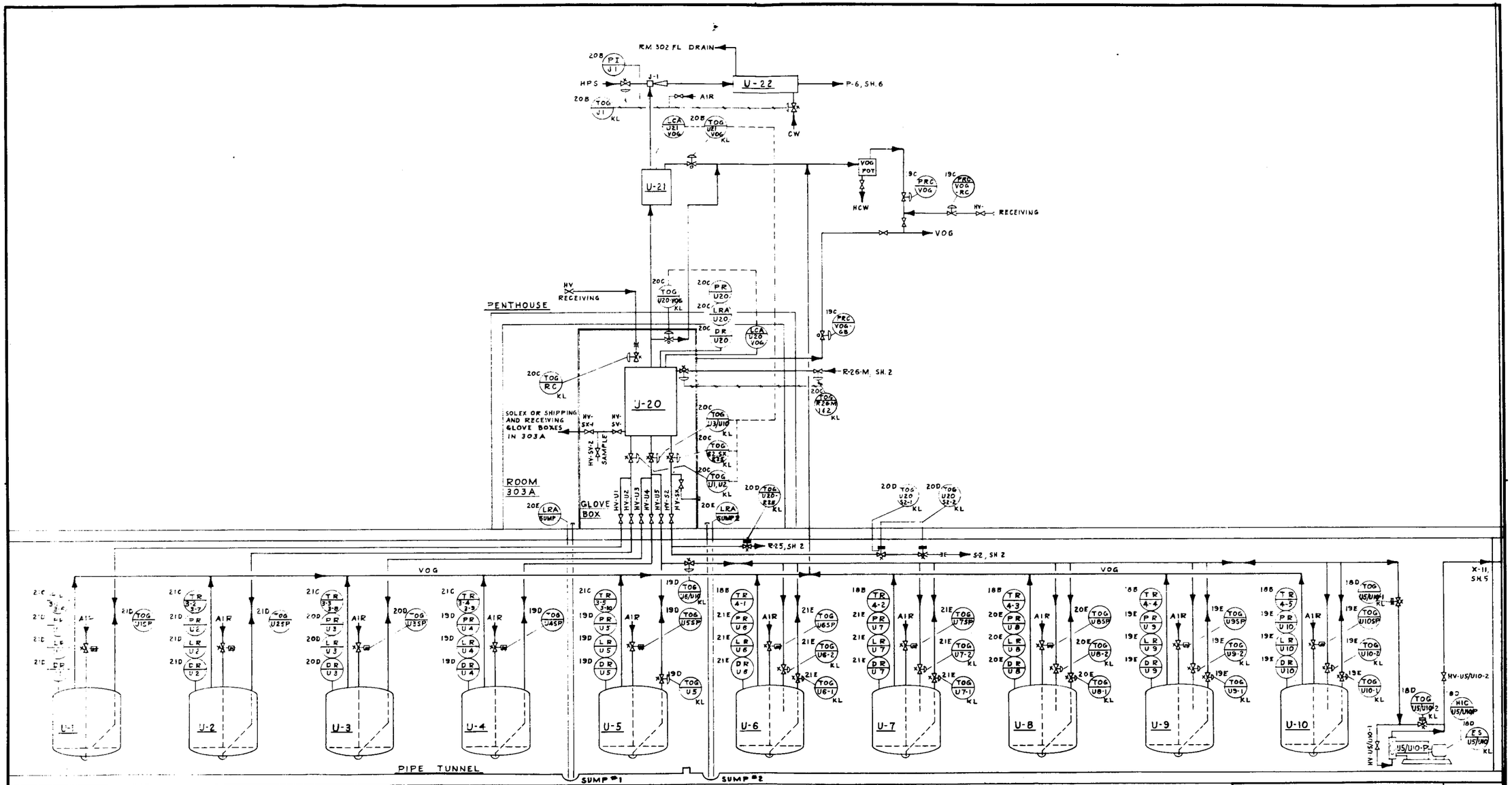
1. R. W. Horton et al., Safety Analysis: LWBR Support Program in Building 3019 Pilot Plant, ORNL-TM-3567 (March 1972).

2. U.S. Atomic Energy Commission, AEC Manual, Chap. 0530, "Nuclear Criticality Safety."

3. Nuclear Safety Guide, TID-7016, Rev. 1 (1961).

4. Subcommittee 8.5 of the Standards Committee of the American Nuclear Society, "Use of Borosilicate Glass Raschig Rings as a Fixed Neutron Absorber in Solutions of Fissile Material," American National Standards Institute (ANSI) Standard N16.4.

9. FIGURES



SYMBOLS

- MANUALLY OPERATED VALVE
- PRESSURE REGULATING VALVE
- AIR OPERATED VALVE
- PISTON OPERATED VALVE
- CHECK VALVE
- SOLENOID OPERATED VALVE
- NORMALLY CLOSED-OPERATING POSITION
- (O) INDICATES VALVE FAILS OPEN
- (X) INDICATES VALVE FAILS CLOSED
- XX TYPE OF INSTRUMENT
- YY EQUIP. PC NO. SERVED BY INST
- ZZ INST LOCATION ON PANELBOARD
- KL KEY LOCK
- PRESSURE REDUCER
- JET
- AGITATOR

VOG- VESSEL OFF GAS
HCW- HOT CHEMICAL WASTE (RADIOACTIVE)
CCW- COLD CHEMICAL WASTE
CD- CONDENSATE DRAIN (STEAM)
SU- SAMPLE UNIT
CENTRIFUGAL PUMP
PULSE PUMP
NOTE-TRANSMITTERS NOT SHOWN
PRIMARY LINES
SECONDARY LINES
INSTRUMENT LINES
ELECTRICAL

REFERENCE DRAWINGS	NUMBER
^{233}U LIQUID STORAGE FACILITY FLOWSHEET II	R20327-EE-026-D
^{233}U LIQUID STORAGE FACILITY FLOWSHEET I	R20327-EE-020-D
ENG. FLOWSHEET THORIUM REMOVAL SYSEM SH. II	CP-021-D
ENG. FLOWSHEET THORIUM REMOVAL SYSTEM SH. I	CP-020-D
S-15E R-35 EQUIPMENT SCHEMATIC FLOWSHEET	CP-018-C
OPERATING FLOWSHEET NO. 6 WASTE SYSTEM	CP-067-D
OPERATING FLOWSHEET NO. 5 IX SYSTEM	CP-066-D
OPERATING FLOWSHEET NO. 4 SX SYSTEM SHEET B	CP-065-D
OPERATING FLOWSHEET NO. 3 SX SYSTEM SHEET A	CP-064-D
OPERATING FLOWSHEET NO. 2 HEAD END SYSTEM	CP-063-D

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TOLERANCES UNLESS OTHERWISE SPECIFIED:
FRACTIONS ± ~
DECIMALS ± ~
ANGLES ± ~
SCALE: NONE

NO.	REVISIONS	DATE	APPD	APPD

DRAWN GREGORY	DATE 12-14-70	SUBMITTED	DATE	APPROVED	DATE
DESIGNED JEN	DATE 11/20/70	APPROVED	DATE	APPROVED	DATE
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE

~~REFERENCE DRAWINGS~~ ~~NUMBER~~

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

LWBR SUPPORT PROGRAM BLDG. NO. 3019

Fig. I. Liquid Storage Operating Flowsheet.

SUBMITTED FAH 4/20/71 | ACCEPTED 4/22/71 | APPROVED

F 12229 CP 062 D REV.

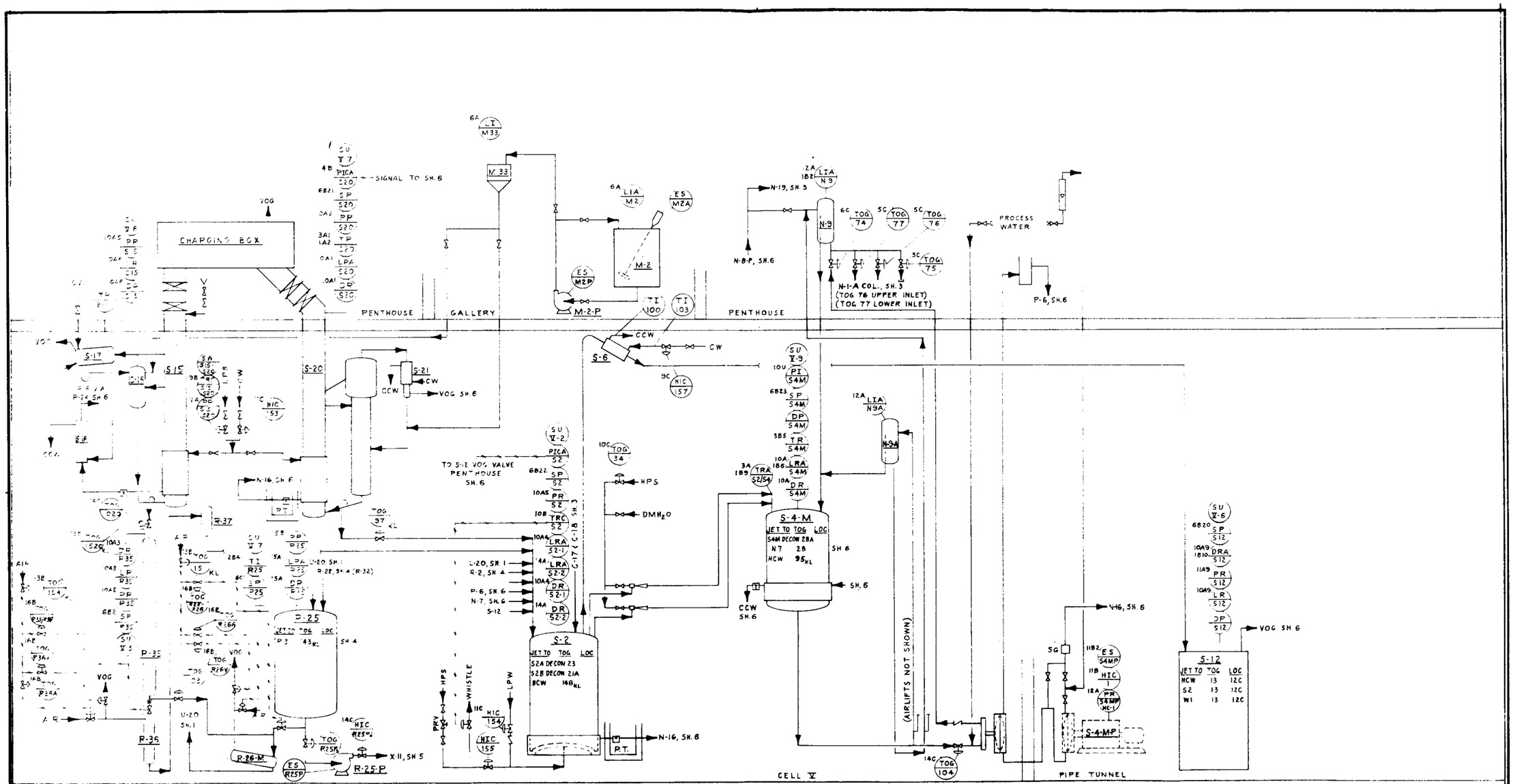

Fig. 2

OPERATING FLOWSHEET NO.3 SX SYSTEM SHEET A	CP-064-D
OPERATING FLOWSHEET NO.1 LIQUID STORAGE FACILITY	CP-062-D
REFERENCE DRAWINGS	NUMBER

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TOLERANCES UNLESS OTHERWISE SPECIFIED:
FRACTIONS ± ____
DECIMALS ± ____
ANGLES ± ____
SCALE: NONE

NO.	REVISIONS	DATE	APPD	APPD

DRAWN GREGORY	DATE 1-6-71	SUBMITTED	DATE	APPROVED	DATE
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE

~~REFERENCE DRAWINGS~~ ~~NUMBER~~

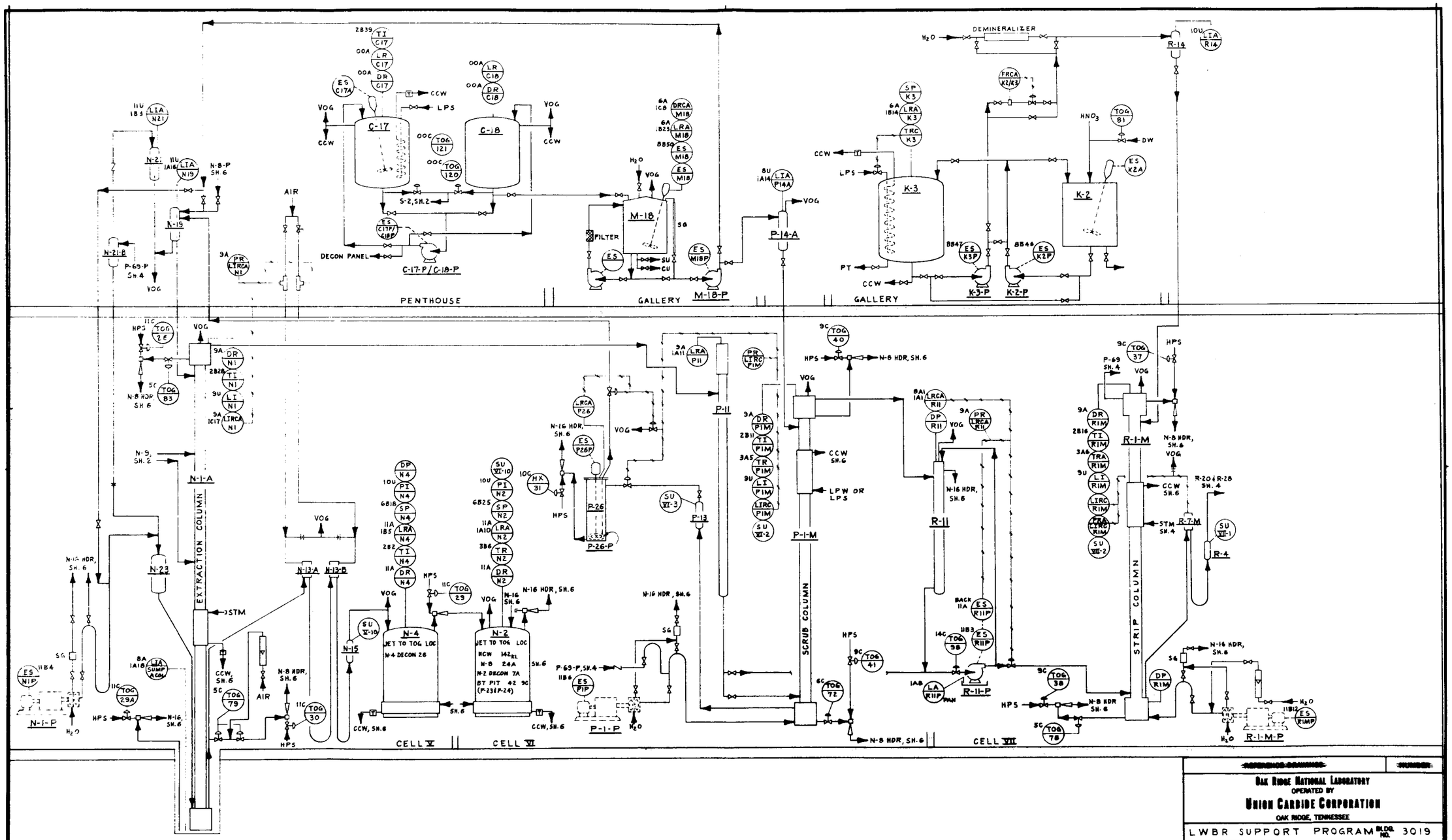
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

LWBR SUPPORT PROGRAM BLDG. NO. 3019

Fig. 2. Head End Operating Flowsheet.

SUBMITTED ACCEPTED APPROVED

12229 CP 063 D REV.



REFERENCE DRAWINGS	NUMBER
OPERATING FLOWSHEET NO. 1-LIQUID STORAGE SYSTEM	CP-062-D
OPERATING FLOWSHEET NO. 2 HEAD END SYSTEM	CP-063-D
OPERATING FLOWSHEET NO. 4-5X SYSTEM SHEET B	CP-065-D

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TOLERANCES UNLESS OTHERWISE SPECIFIED:
FRACTIONS ± ~
DECIMALS ± ~
ANGLES ± ~
SCALE: NONE

NO.	REVISIONS	DATE	APPD	APPD

DRAWN	DATE	SUBMITTED	DATE	APPROVED	DATE
GREGORY	1-15-71				
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE

~~REFERENCE DRAWINGS~~ ~~NUMBER~~

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

LWBR SUPPORT PROGRAM BLDG. NO. 3019

Fig. 3. Solvent Extraction Operating Flowsheet (A).

SUBMITTED FEH 4/14/71 | ACCEPTED 4/14/71 | APPROVED W. J. Ung. (E&A)

F 12229 CP 064 D REV.

ORNL Dwg 71-4783

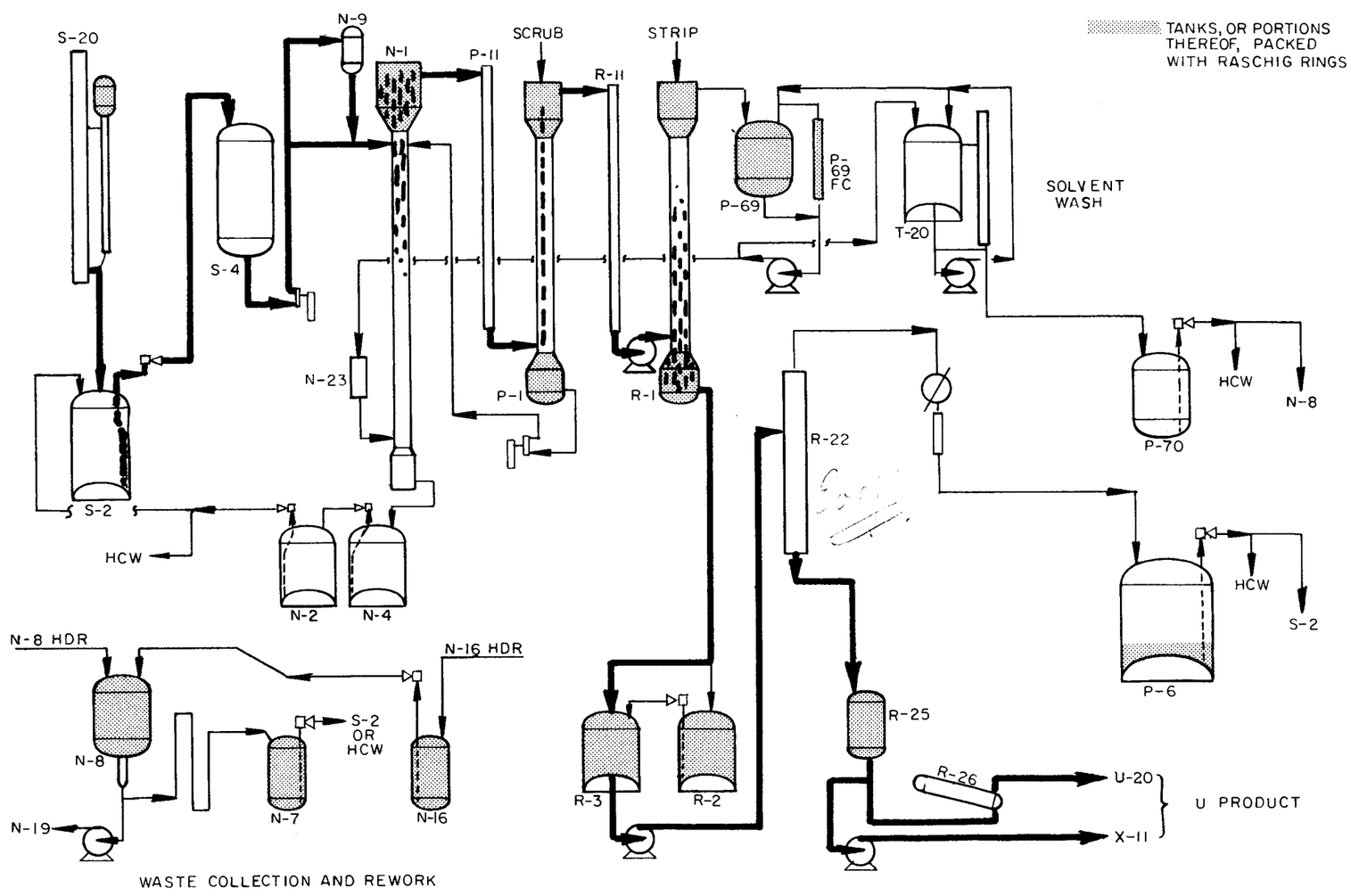
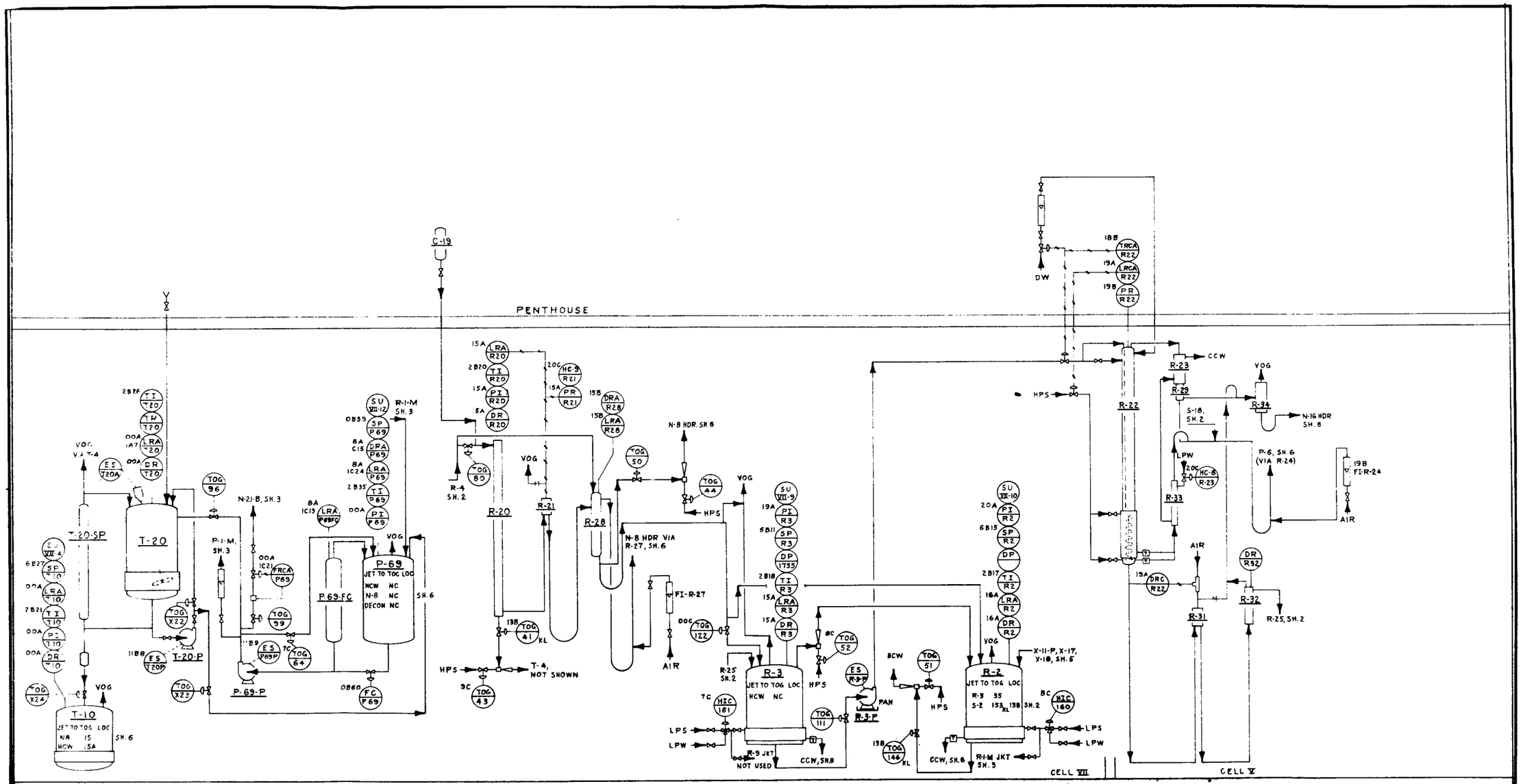


Fig. 4. Schematic Equipment Flowsheet - Solvent Extraction System.



REFERENCE DRAWINGS	NUMBER
OPERATING FLOWSHEET NO. 1 LIQUID STORAGE SYSTEM	CP-062-D
OPERATING FLOWSHEET NO. 3 SX SYSTEM SHEET A	CP-064-D
OPERATING FLOWSHEET NO. 5 I X SYSTEM	CP-066-D

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TOLERANCES UNLESS OTHERWISE SPECIFIED:
FRACTIONS ± ~
DECIMALS ± ~
ANGLES ± ~
SCALE: NONE

NO.	REVISIONS	DATE	APPD	APPD

DRAWN GREGORY	DATE 1-12-71	SUBMITTED	DATE	APPROVED	DATE
DESIGNED FPM	DATE 10/1/70	APPROVED	DATE	APPROVED	DATE
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE

~~REFERENCE DRAWINGS~~ ~~NUMBER~~

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

LWBR SUPPORT PROGRAM BLDG. NO. 3019

Fig. 5. Solvent Extraction Operating Flowsheet.

SUBMITTED FPM 4/6/71 | ACCEPTED 4/20/71 | APPROVED W. E. Lyon

F 12229 CP 065 D REV.

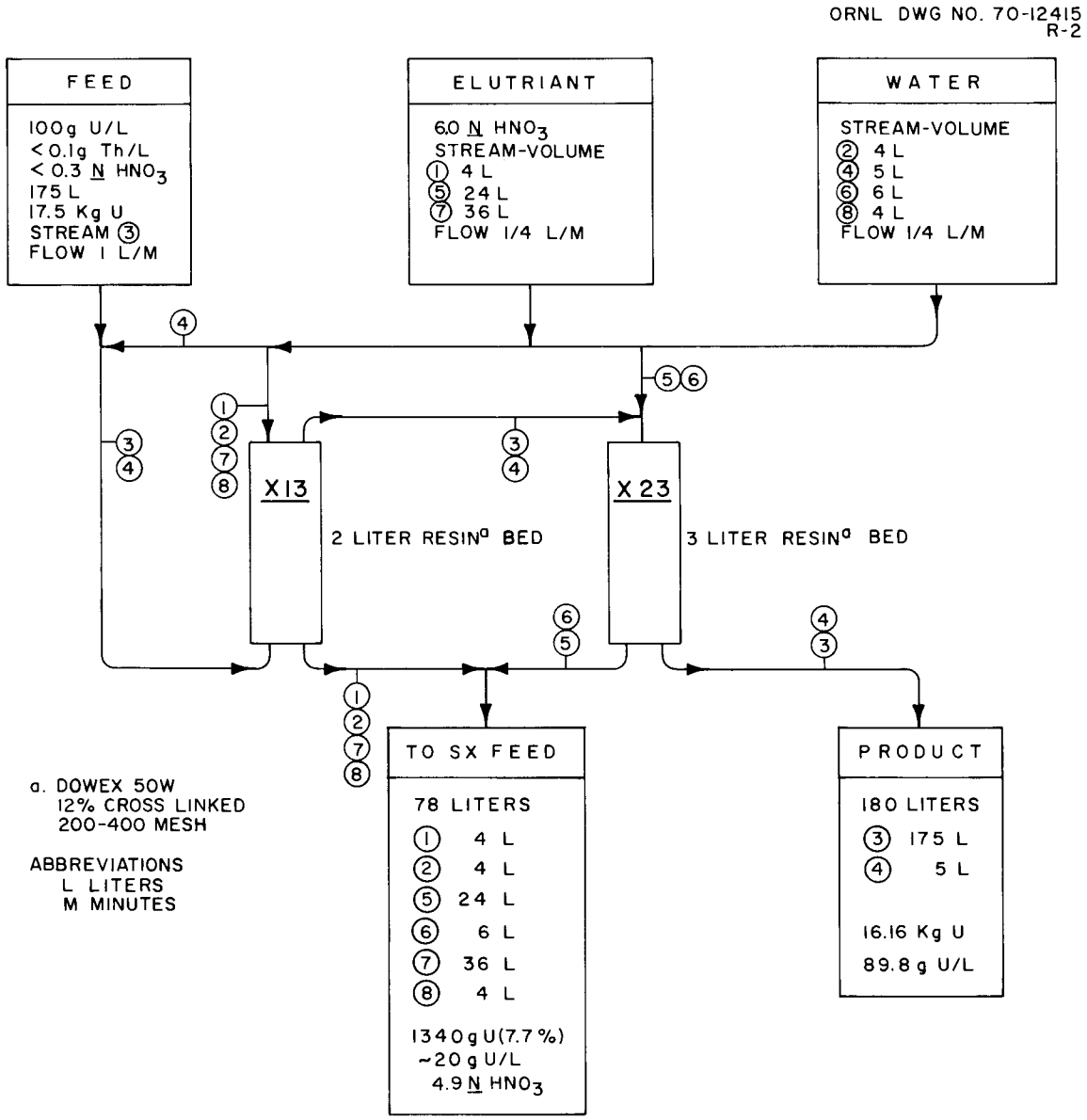
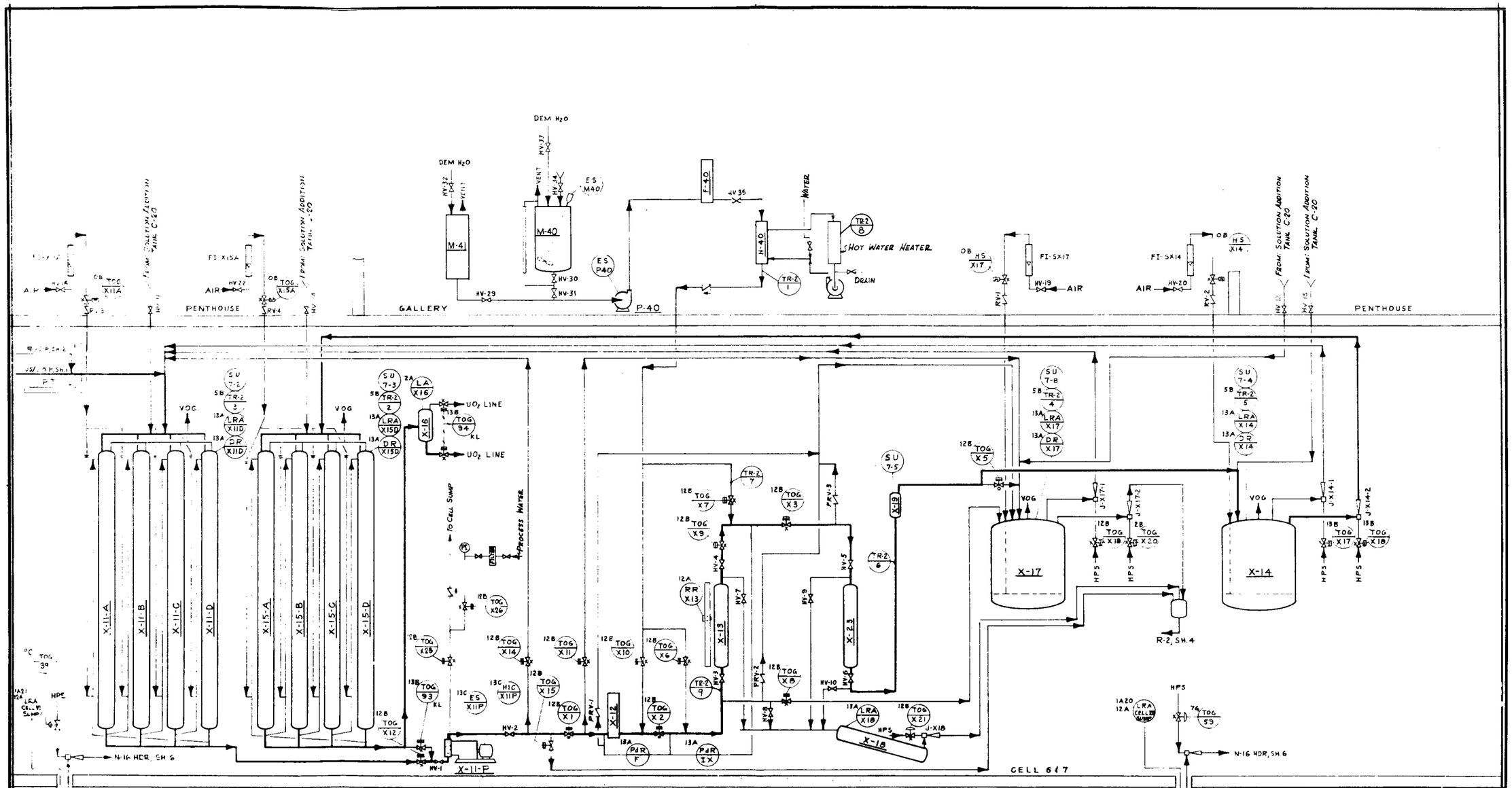


Fig. 6. Process Flowsheet - Ion Exchange System.



REFERENCE DRAWINGS	NUMBER
OPERATING FLOWSHEET NO. 6 WASTE SYSTEM	CP-067-D
OPERATING FLOWSHEET NO. 4 SX SYSTEM	CP-065-E
OPERATING FLOWSHEET NO. 1 LIQUID STORAGE SYSTEM	CP-062-D

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FRACTIONS ±
DECIMALS ±
ANGLES ±
SCALE: NONE

NO.	REVISIONS	DATE	APPD	APPD
DRAWN GREGORY	DATE 12-19-70	SUBMITTED	DATE	APPROVED / DATE
DESIGNED	DATE	APPROVED	DATE	APPROVED / DATE
CHECKED	DATE	APPROVED	DATE	APPROVED / DATE

~~REFERENCE DRAWINGS~~ ~~NUMBER~~

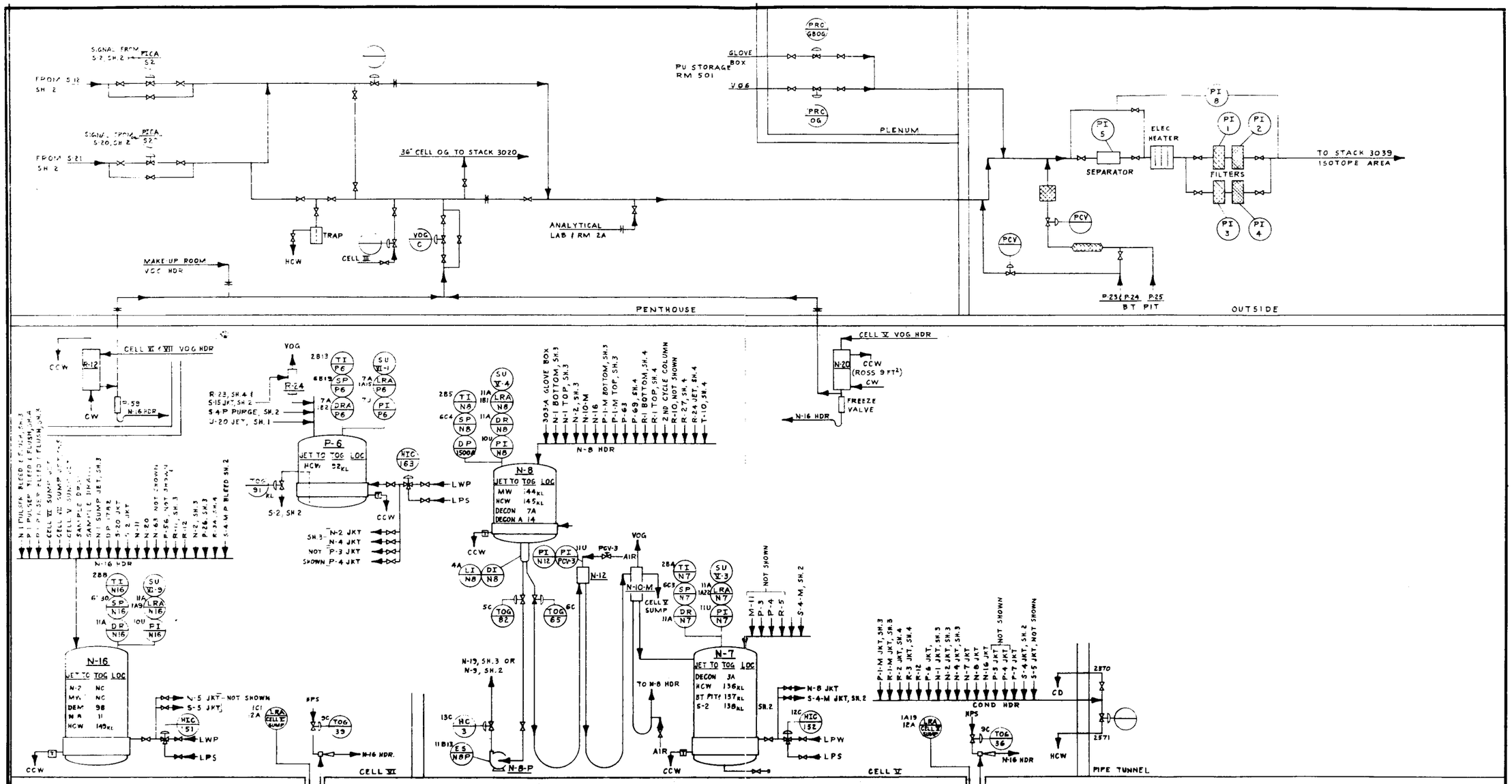
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

LWBR SUPPORT PROGRAM BLDG. NO. 3019

Fig. 7. Ion Exchange System Operating Flowsheet.

SUBMITTED / ACCEPTED / APPROVED

F 12229 CP 066 D REV.



OPERATING FLOWSHEET NO. 5 - I X SYSTEM	CP-066-D
OPERATING FLOWSHEET NO. 1 LIQUID STORAGE FACILITY	CP-062-D
REFERENCE DRAWINGS	NUMBER

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TOLERANCES UNLESS OTHERWISE SPECIFIED:
FRACTIONS ± ~
DECIMALS ± ~
ANGLES ± ~
SCALE: NONE

NO.	REVISIONS			DATE	APPD	APPD
DRAWN GREGORY	DATE 12-29-70	SUBMITTED	DATE	APPROVED	DATE	
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE	
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE	

~~REFERENCE DRAWINGS~~ ~~NUMBER~~

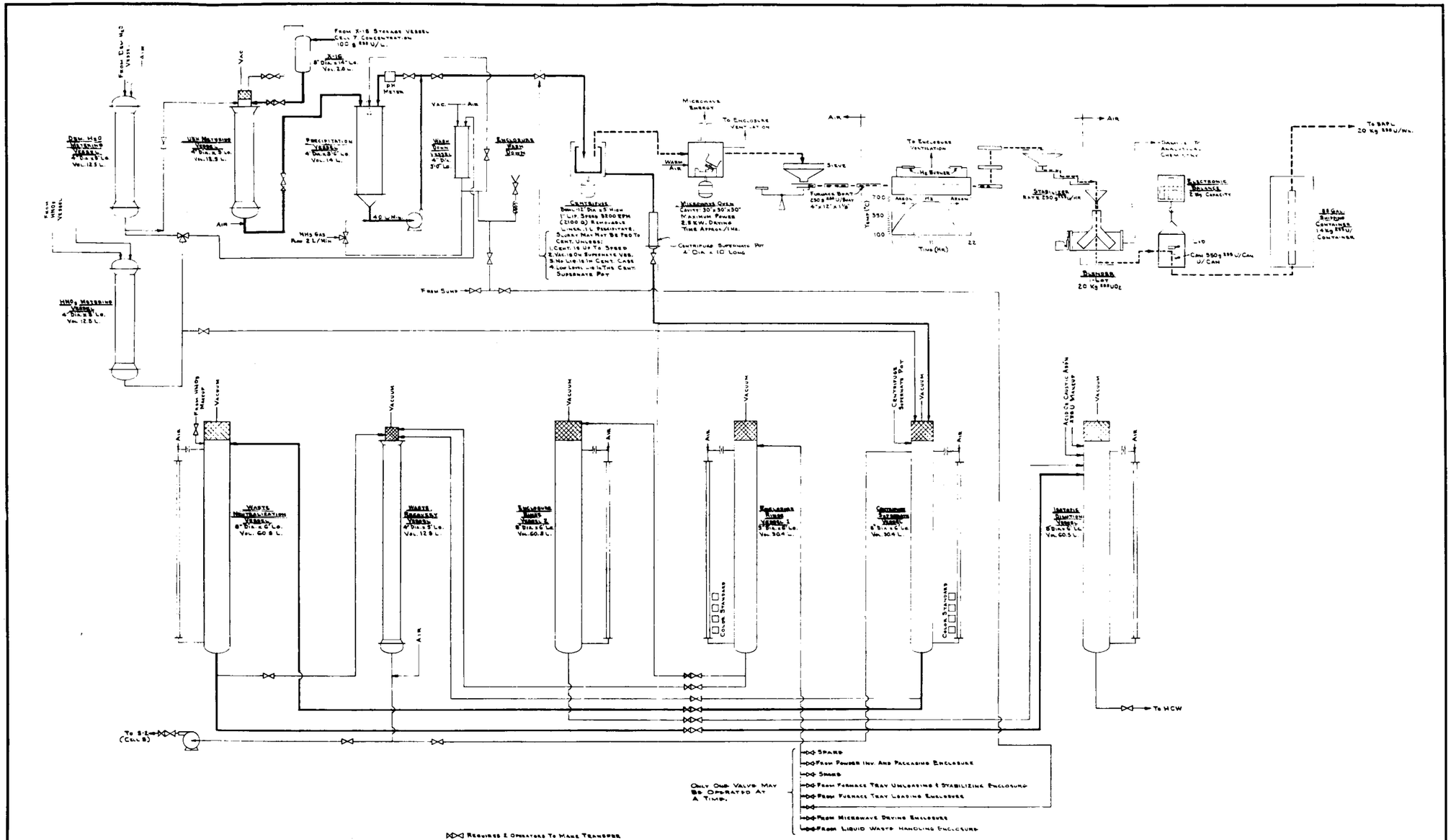
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

LWBR SUPPORT PROGRAM BLDG. NO. 3019

Fig. 8. Waste Recovery System Operating Flowsheet.

SUBMITTED ACCEPTED APPROVED

F 12229 CP 067 D REV.



REFERENCE DRAWINGS					NUMBER
OAK RIDGE NATIONAL LABORATORY OPERATED BY UNION CARBIDE CORPORATION OAK RIDGE, TENNESSEE					
235 UO2 FACILITY					DWG. NO. 5019
Fig. 9. Equipment Flowsheet					
NO.	REVISIONS	DATE	APPD	APPD	
DRAWN AL Frawen 12-15-70	SUBMITTED DATE	APPROVED DATE	APPROVED DATE		
DESIGNED T Davenport	DATE	APPROVED DATE	APPROVED DATE		
DESIGNED	DATE	APPROVED DATE	APPROVED DATE		
SCALE: None					
F	2229	CD	M03	B	SHEET 17

TOLERANCES UNLESS OTHERWISE SPECIFIED: FRACTIONS ± ___ DECIMALS ± ___ ANGLES ± ___

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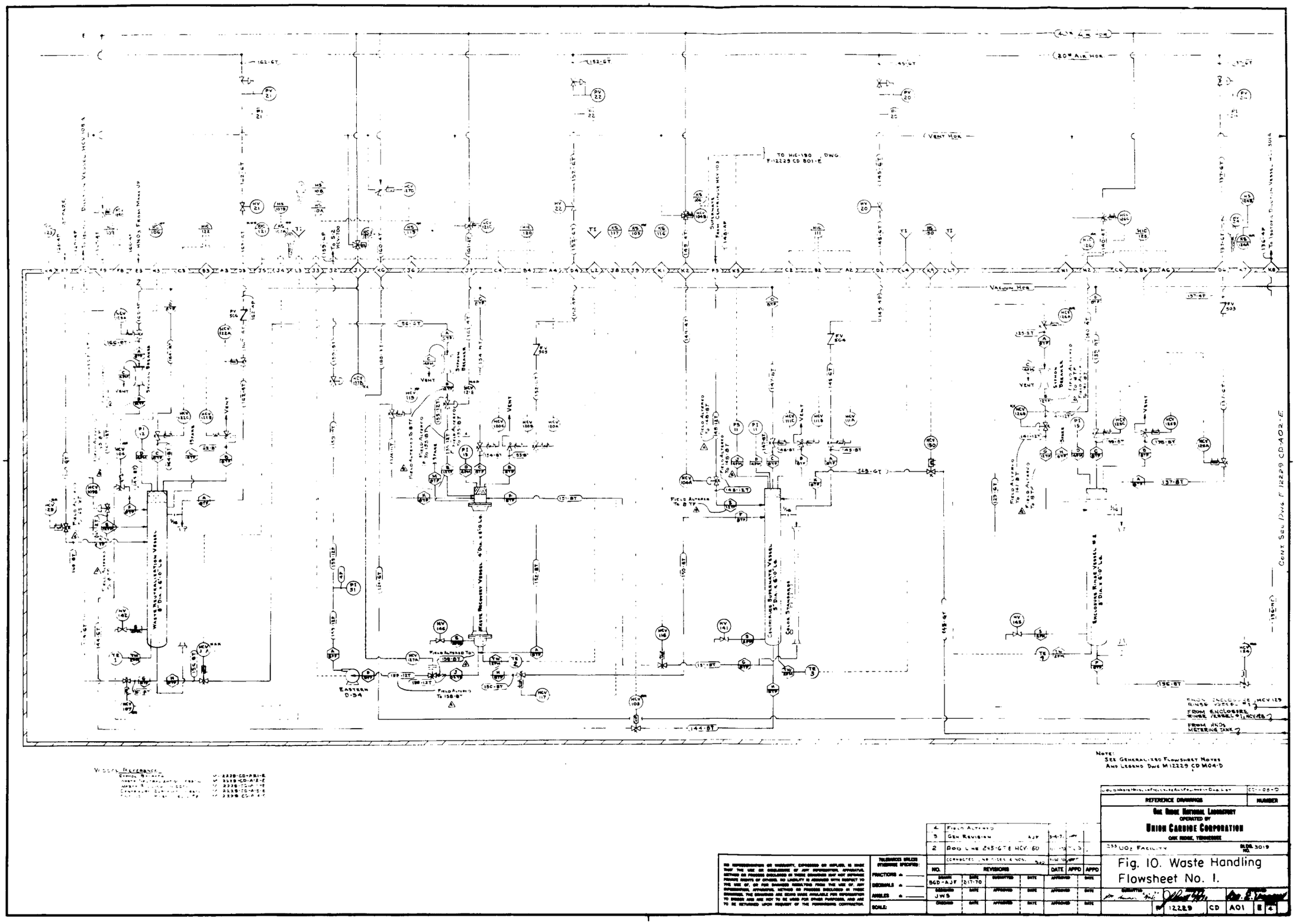
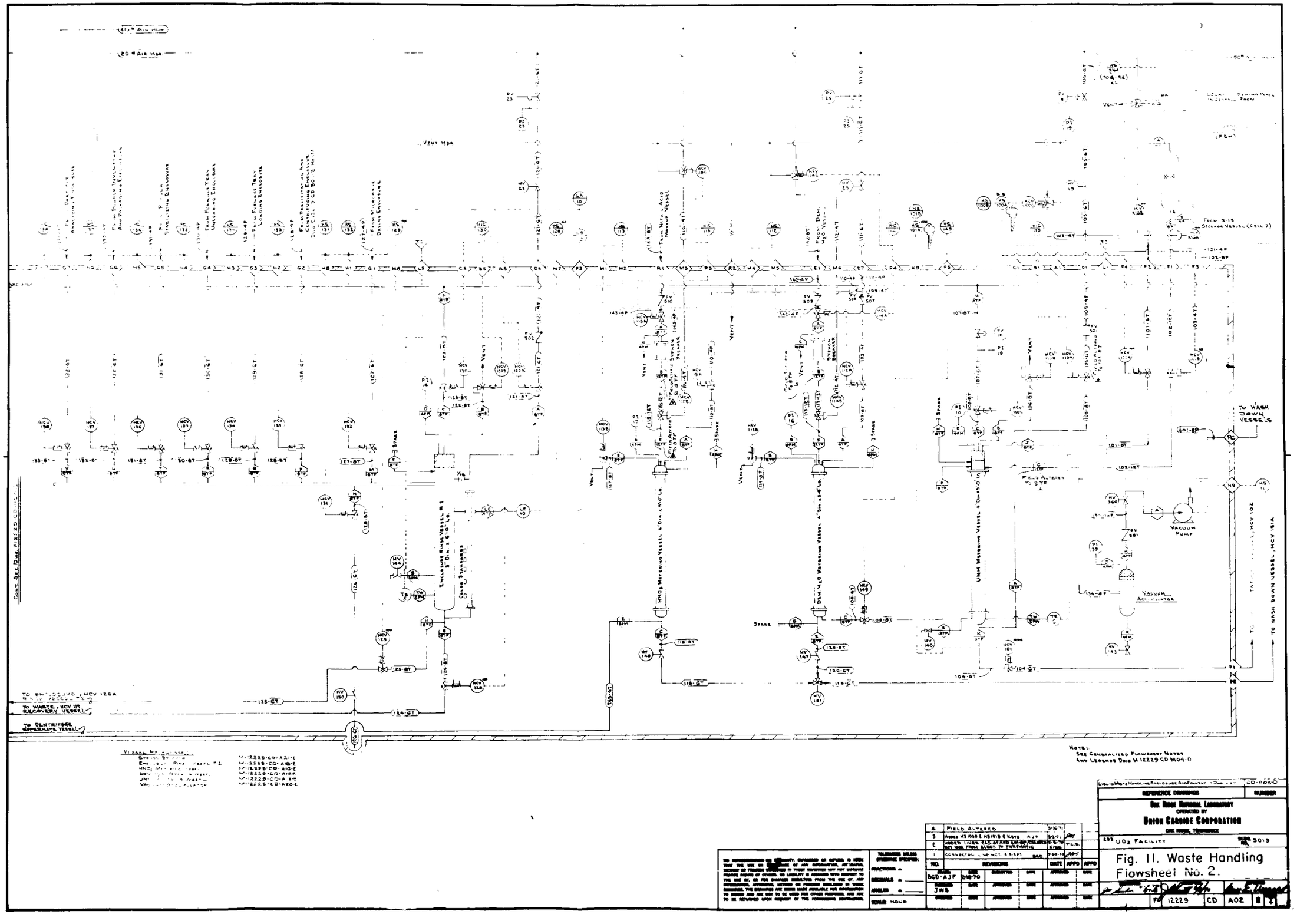

Vacuum Hdr.
Vent Hdr.
Waste Recovery Vessel 4' Dia. x 6'-0" Lg.
Waste Neutralization Vessel 8' Dia. x 6'-0" Lg.
Enclosure Rinse Vessel #2 8' Dia. x 6'-0" Lg.
Cont. See Dwg. F 12229 CD-A02-E
Note:
See General-Use Flowsheet Notes
And Legend Dwg M12229 CD M04-D
Oak Ridge National Laboratory
Operated by
Union Carbide Corporation
Oak Ridge, Tennessee
Fig. 10. Waste Handling
Flowsheet No. 1.


20 # Air Hdr.
Vent Hdr.
From Nitric Acid Manup Vessel
From Deh. H2O Vessel
From X-115 Storage Vessel (Cell 7)
Vacuum Pump
Vacuum Accumulator
To Wash Down Vessels
To Wash Down Vessel, HCV 102
To Wash Down Vessel, HCV 101A
To Encl. Cooler, HCV 12GA
To Waste, HCV 119
To Centrifuge Supernate Vessel
Encl. Cooler Vessel No. 1
Spare
Field Altered
Note:
See Generalized Flowsheet Notes And Legends Dwg M 12229 CD M04-0
Reference Drawings
Number
Oak Ridge National Laboratory
Operated By
Union Carbide Corporation
Oak Ridge, Tennessee
UO2 Facility
Fig. 11. Waste Handling Flowsheet No. 2.
Revisions
Date
Appd
Drawn BGD-AJF 12-18-70
Checked JWB
Scale None
Fd 12229 CD A02 B 2

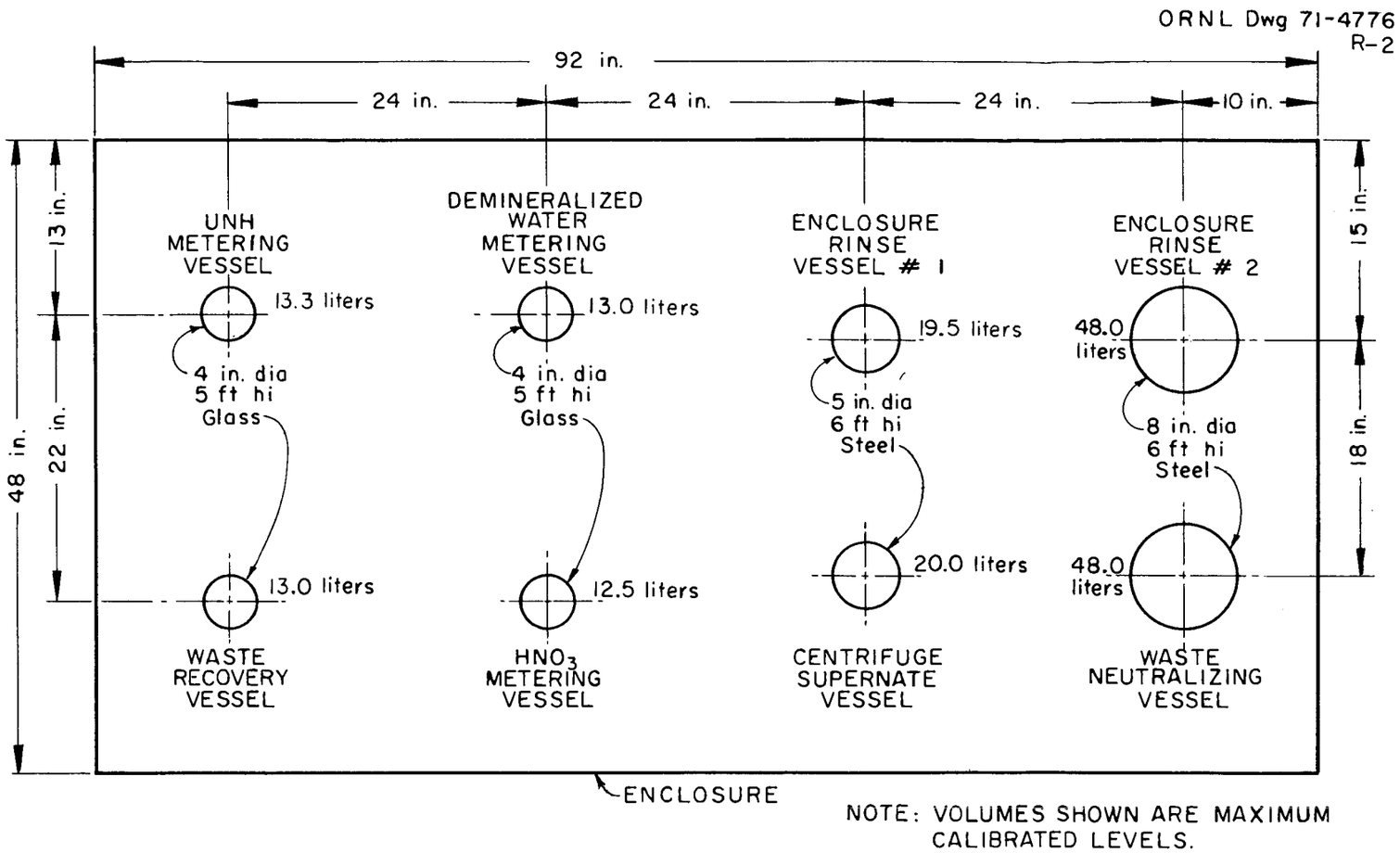
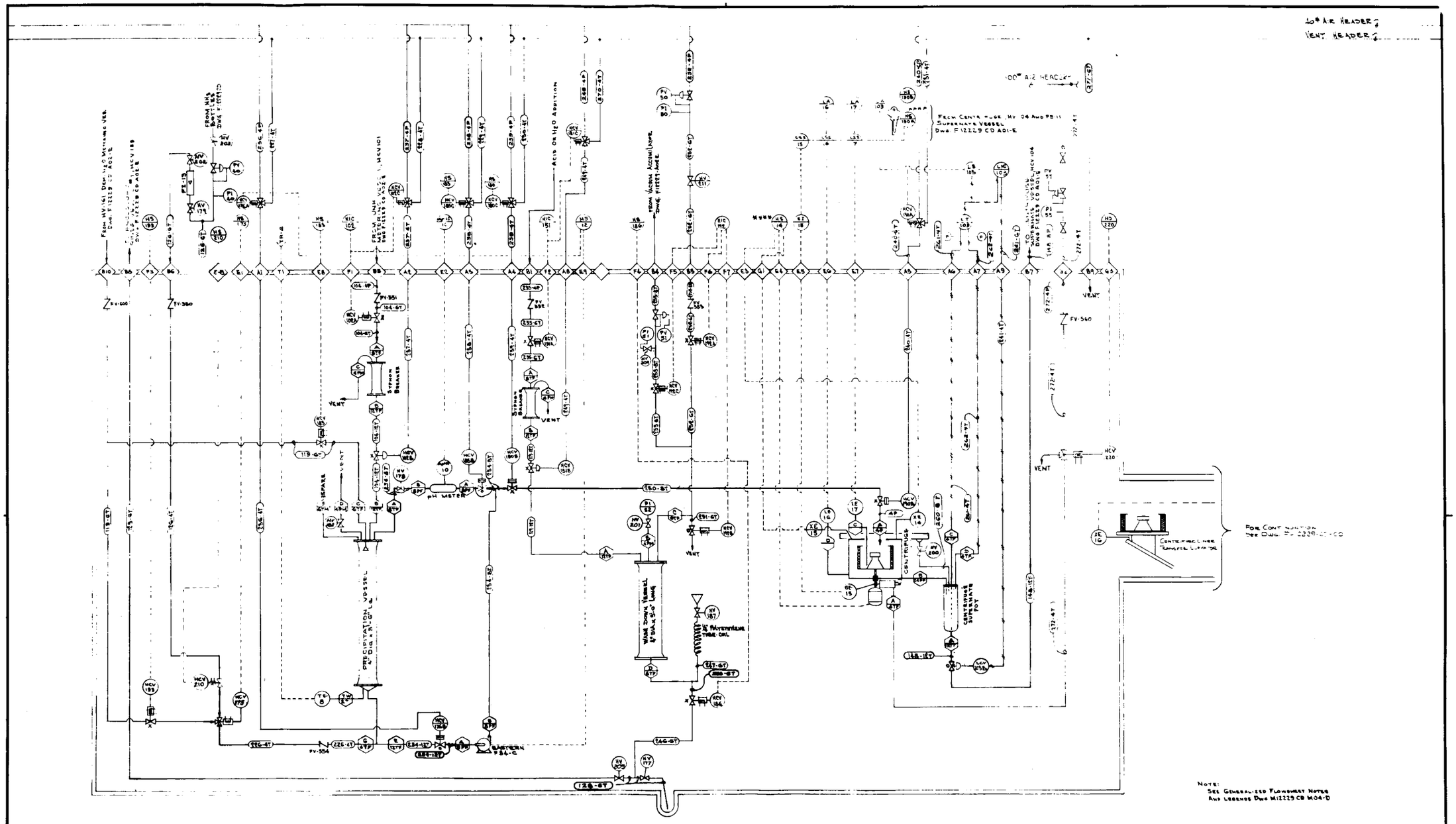


Fig. 12. Plan of Vessels Installed in Liquid Waste Handling Facility.



2	GENERAL REVISIONS						
1	GENERAL REVISION			A.S.P.	3-4-71		
NO.	REVISIONS				DATE	APPD	APPD
DRAWN DSVALL	DATE 3-26-70	DESIGNED	DATE		APPROVED	DATE	
DRAWN EN. DSM	DATE	APPROVED	DATE		APPROVED	DATE	
DRAWN	DATE	APPROVED	DATE		APPROVED	DATE	

TOLERANCES UNLESS OTHERWISE SPECIFIED:
FRACTIONS ± ____
DECIMALS ± ____
ANGLES ± ____
SCALE: NONE

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PROJECT OR PROCESS AND PROPRIETARY PROCESS DRAWING LIST — LD-505-D
REFERENCE DRAWINGS — NUMBER

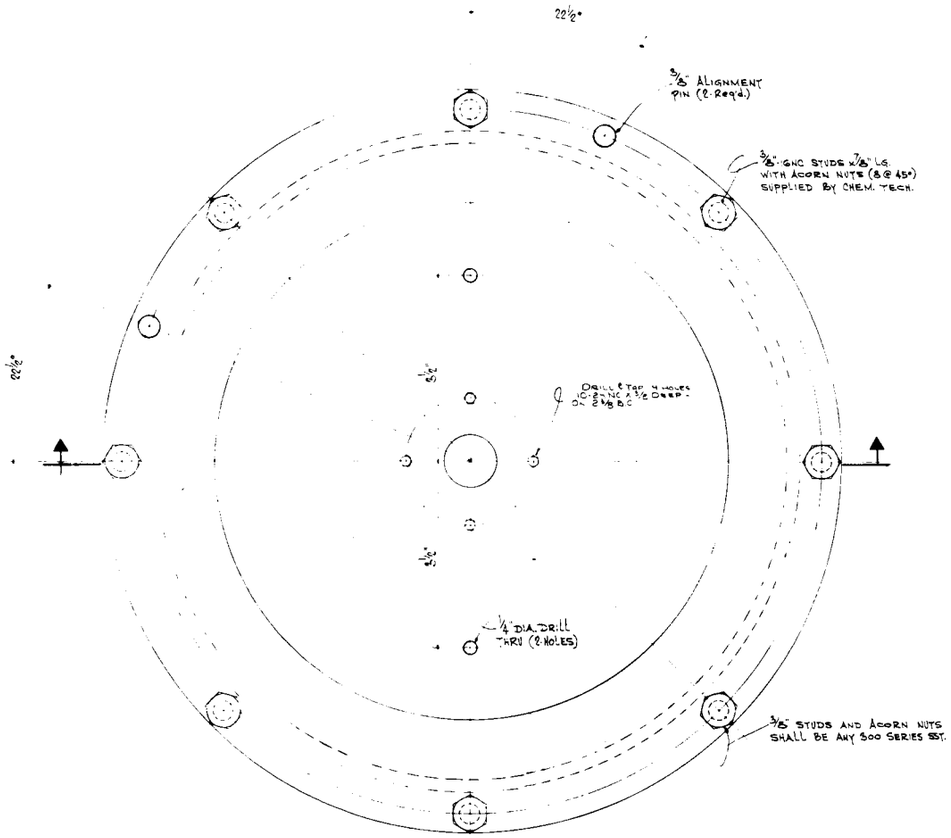
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

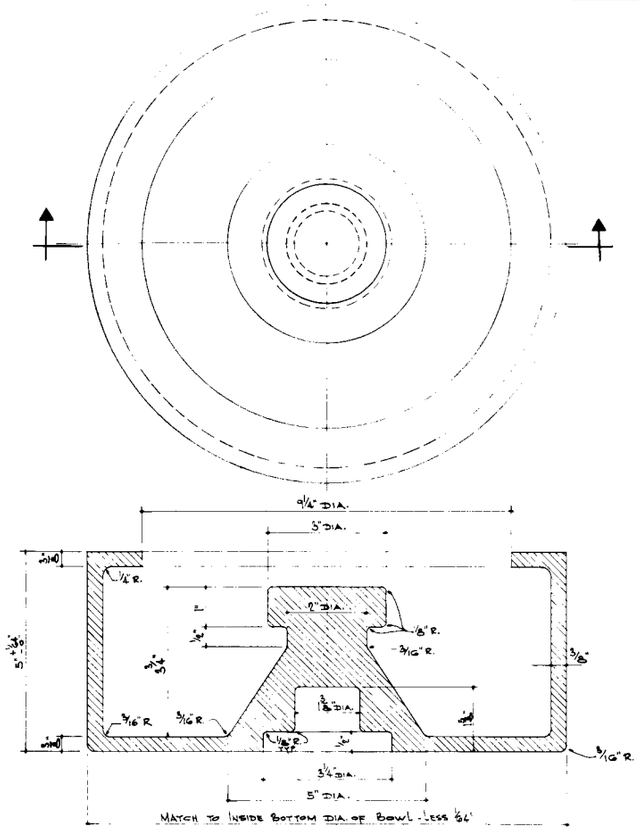
UO2 FACILITY — DWG. NO. 30-9

Fig. 13. Precipitation and Centrifugation Operating Flowsheet.

F | 12229 | CD | 801 | 2

PARTS LIST					
PART	DWG NO.	REQD	DESCRIPTION	STOCK SIZE	MATERIAL

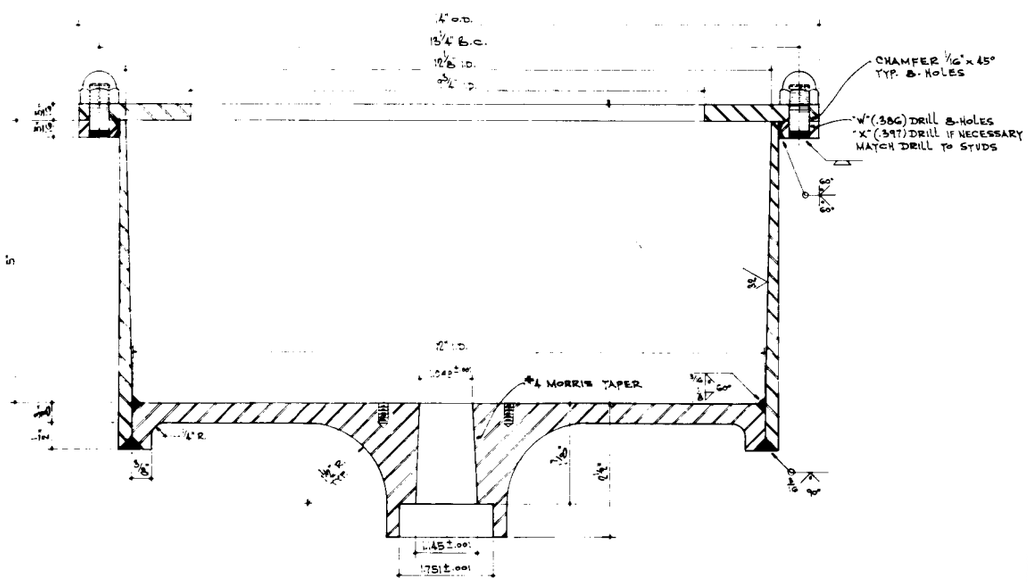




LINER DETAIL

MAT'L: TEFLON

3/4" = 1"



CENTRIFUGE BOWL DETAIL

ALL MAT'L SHALL BE 304L SST UNLESS NOTED

FULL SIZE.

NOTE:

ALL WELDS SHALL BE INSPECTED IN ACCORDANCE WITH MET-WR-001, SECTIONS 14 & 15.

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GENERAL SPECIFICATIONS

UNLESS OTHERWISE SPECIFIED:

1. BREAK ALL SHARP EDGES
2. TYPE, GRADE, OR FINISH OF MATERIAL MAY BE CHOSEN BY FABRICATOR.
3. MACHINED SURFACE FINISH SHALL NOT EXCEED: (ASA B46.1—1962) ✓

TOLERANCES UNLESS OTHERWISE SPECIFIED:

FRACTIONS ± ____

DECIMALS ± ____

ANGLES ± ____

SCALE:

NO.	REVISIONS				DATE	APPD	APPD
DRAWN J.W. KIDD	DATE 5-12-71	SUBMITTED	DATE	APPROVED	DATE		
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE		
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE		

PRECIPITATING & CENTRIFUGING OPERATING FLOWSHEET — 2D-B01-E

REFERENCE DRAWINGS — NUMBER

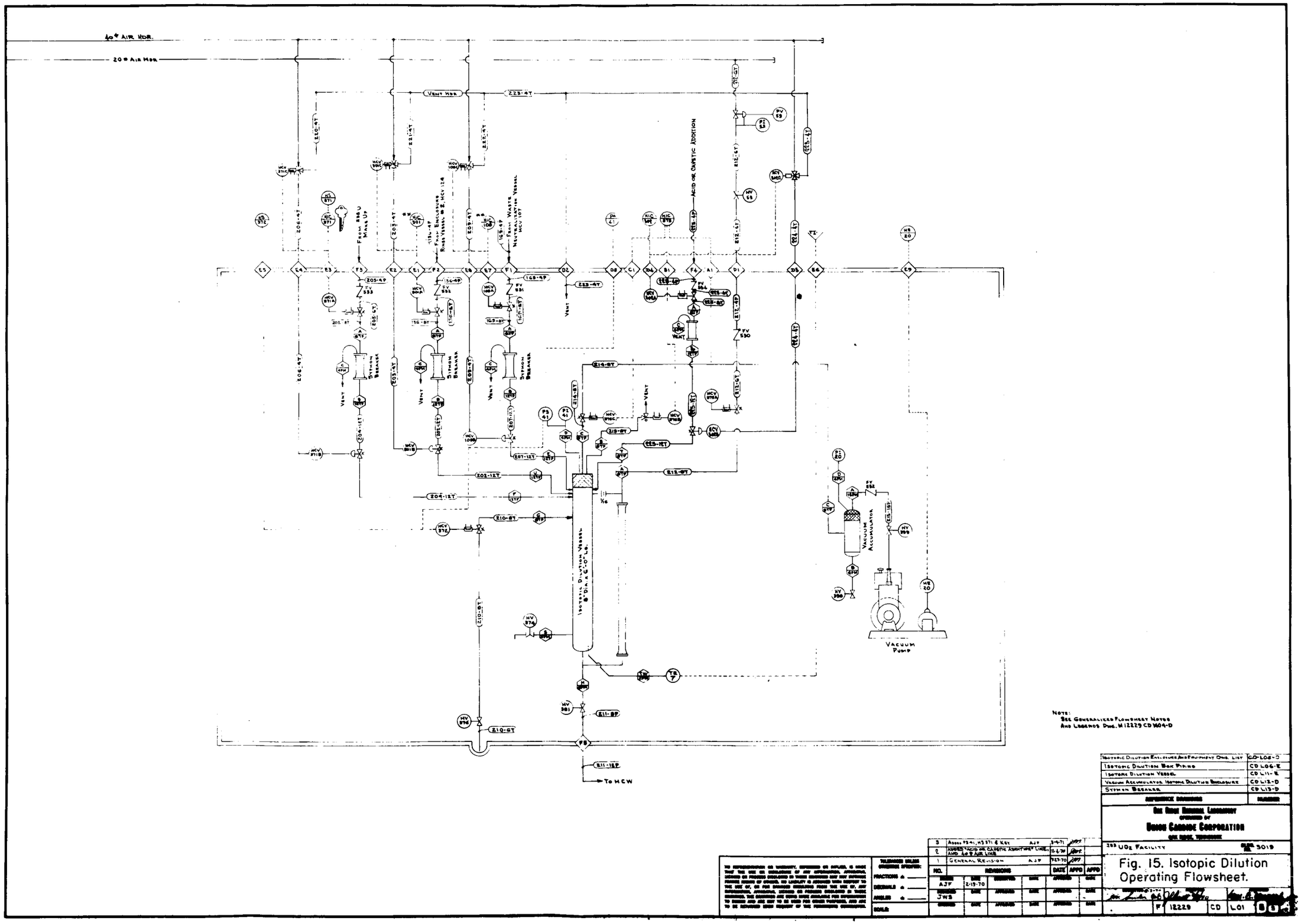
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

BLDG. NO.

Fig. 14. Centrifuge Bowl and Liner.

SUBMITTED — RECOMMENDED — APPROVED

M 10009 | 2D | B15 | 8 | 0



Note:
See Generalized Flowsheet Notes
And Legends Dwg. M 12229 CD W04-D

Isotopic Dilution Enclosure Area Equipment Dwg. List	CD-L06-D
Isotopic Dilution Box Piping	CD L06-E
Isotopic Dilution Vessel	CD L11-E
Vacuum Accumulator Isotopic Dilution Enclosure	CD L12-D
Siphon Breaker	CD L13-D
Reference Drawings	Number

Oak Ridge National Laboratory
Operated by
Union Carbide Corporation
Oak Ridge, Tennessee

No.	Revisions	Date	Appd	Appd
3	Added TS-41, HS 371 & Key	AJF 3-9-71		
2	Added "Acid or Caustic Addition" Line, and 40# Air Line	12-6-70		
1	General Revision	AJF 7-23-70		

Tolerances Unless Otherwise Specified:
Fractions ± ___
Decimals ± ___
Angles ± ___
Scale:

Drawn AJF 2-19-70 | Checked | Approved | Date
Traced JWS | Checked | Approved | Date

233 UO_2 Facility — Dwg. No. 3019

Fig. 15. Isotopic Dilution Operating Flowsheet.

F 12229 CD L01 B

ORNL Dwg. 71-8087

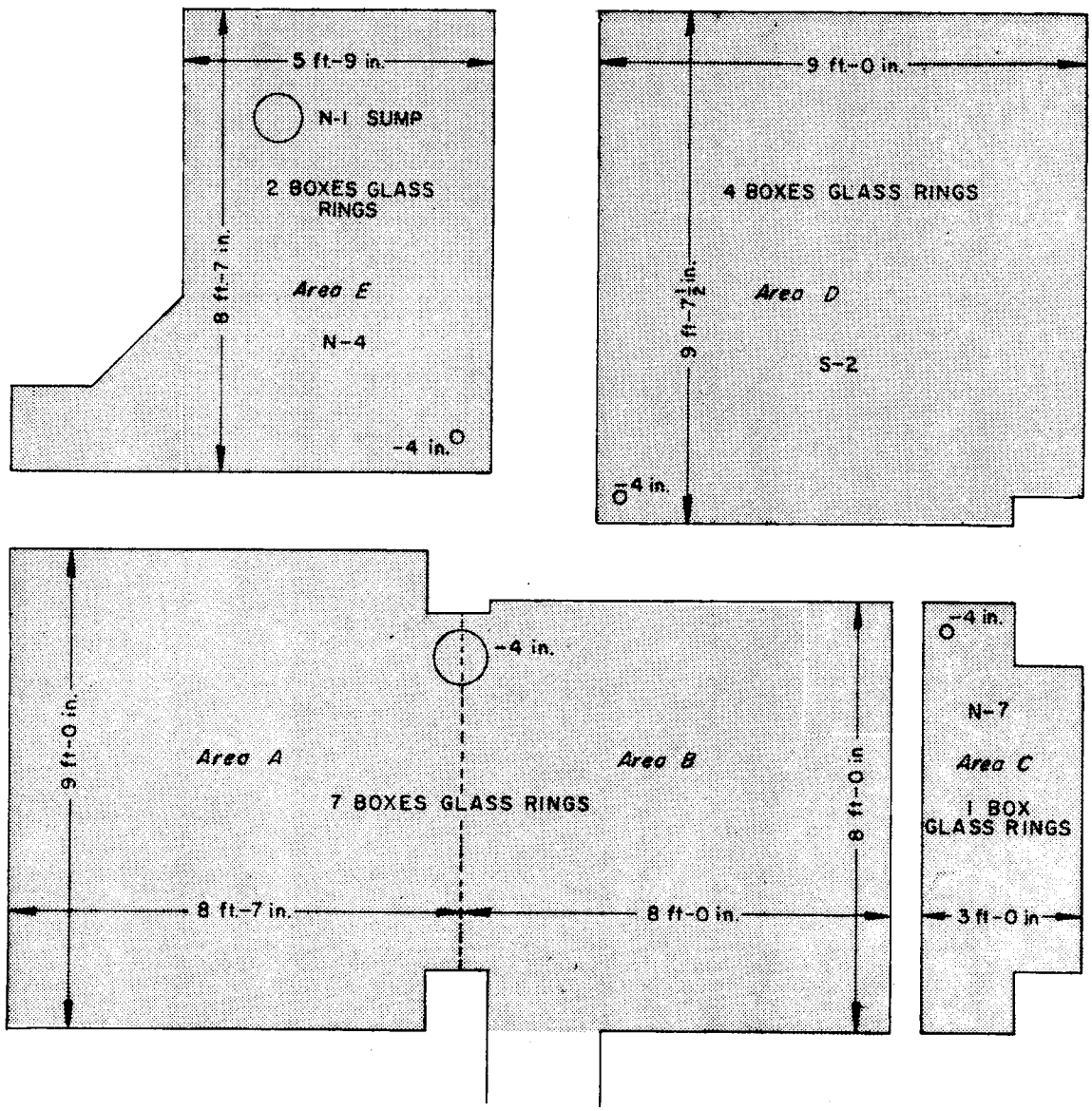
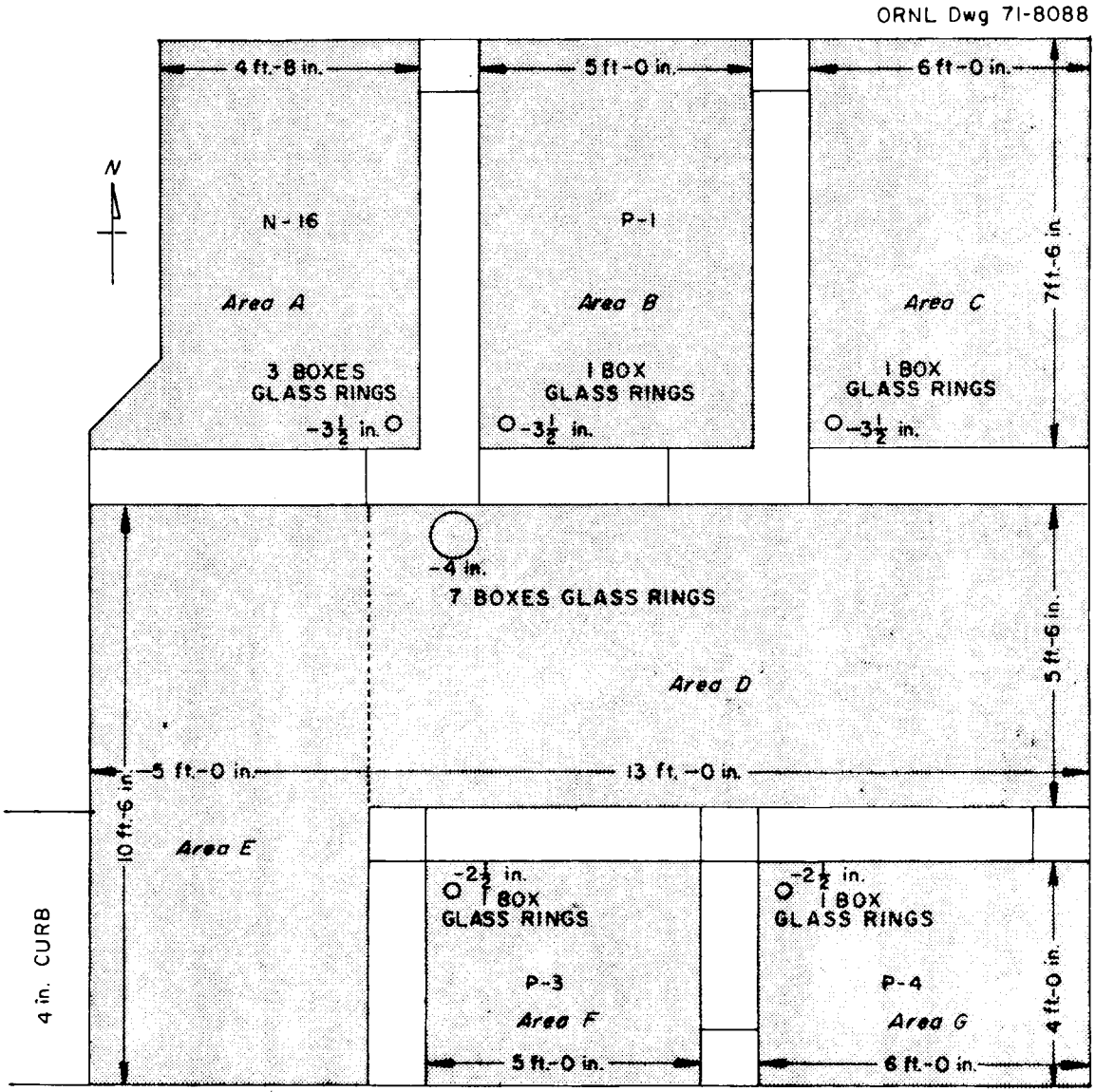


Fig. 16. Floor Plan of Cell 5.



Note: 1 BOX = 345 RINGS
= 9.58 LITERS OF GLASS

Fig. 17. Floor Plan of Cell 6.

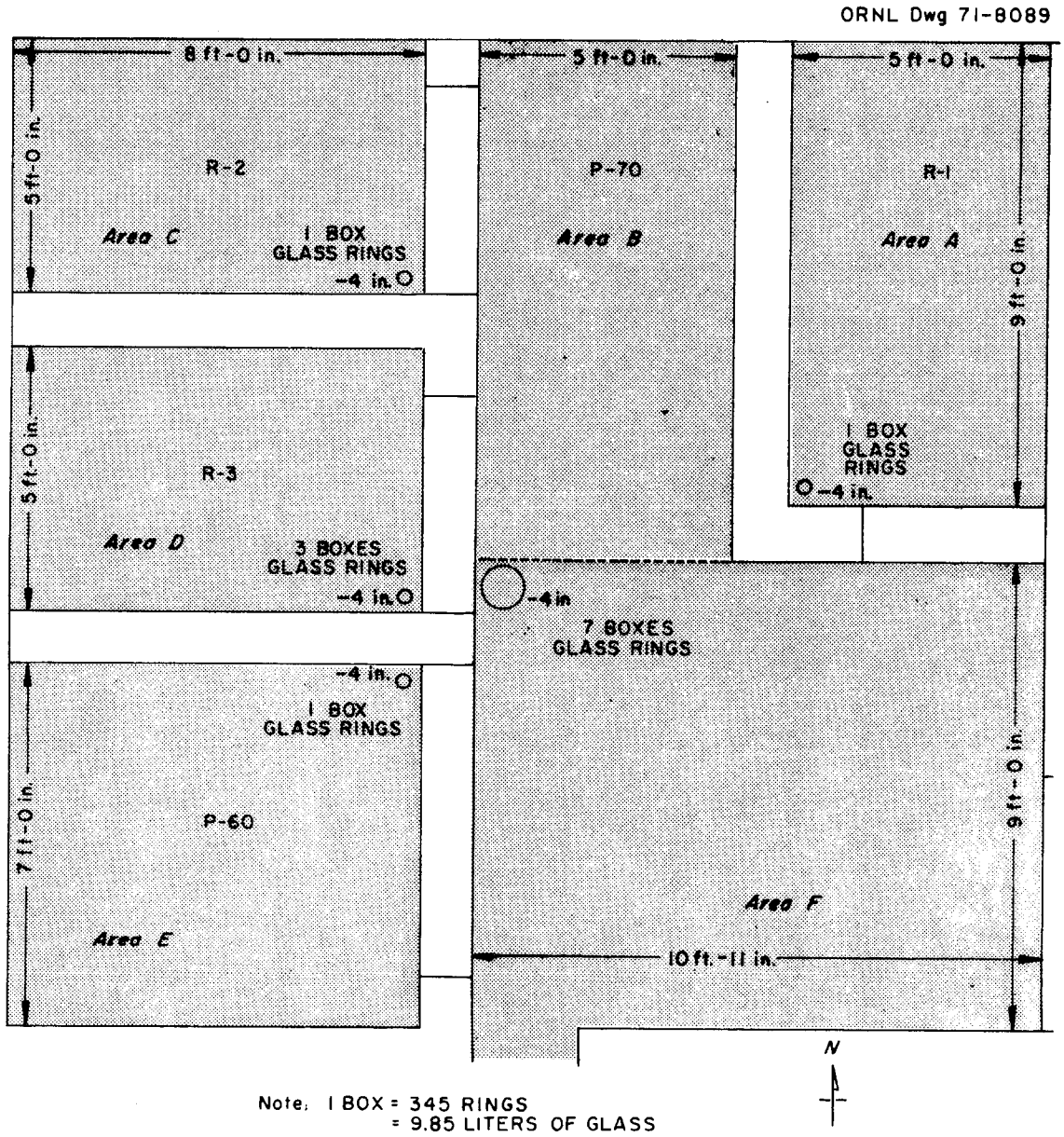


Fig. 18. Floor Plan of Cell 7.

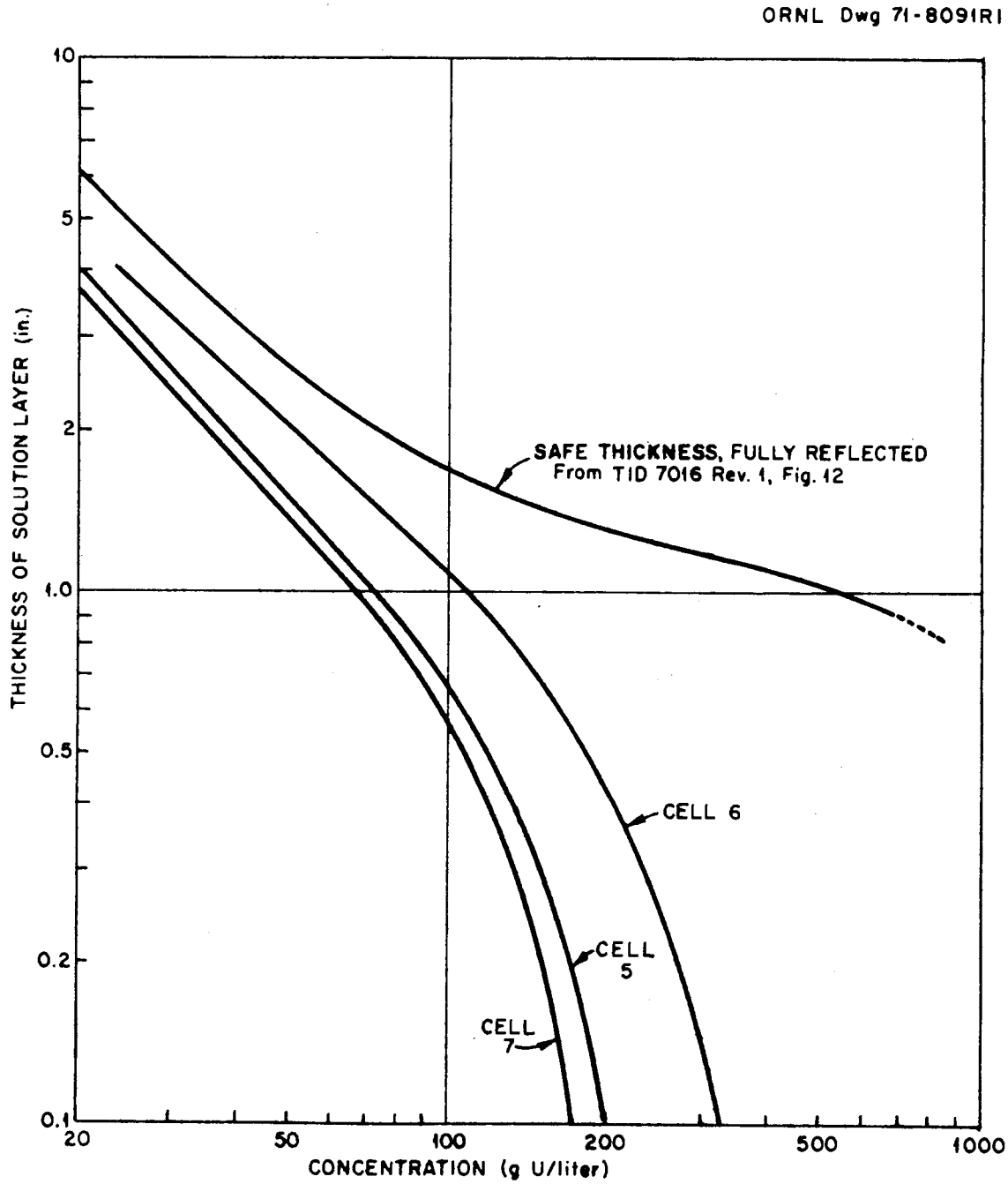
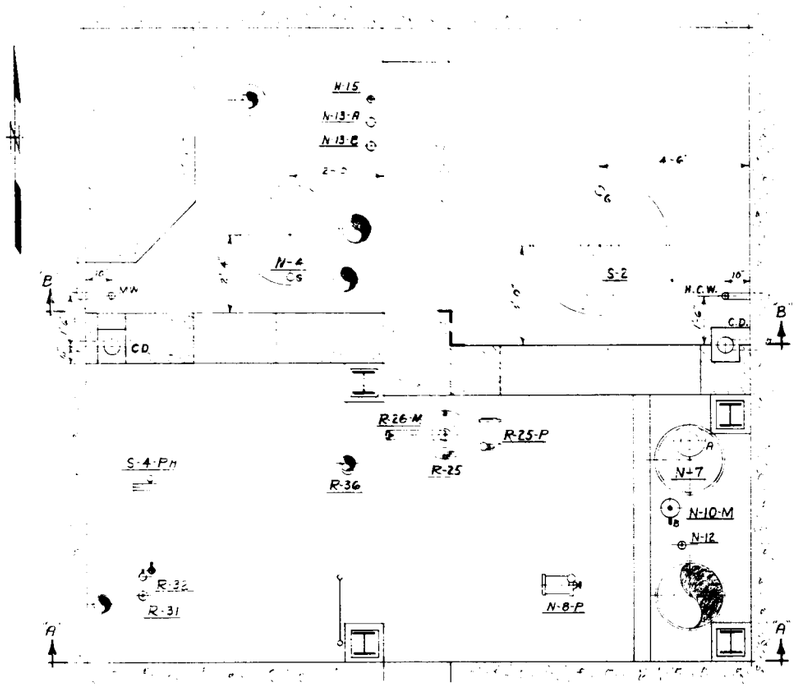
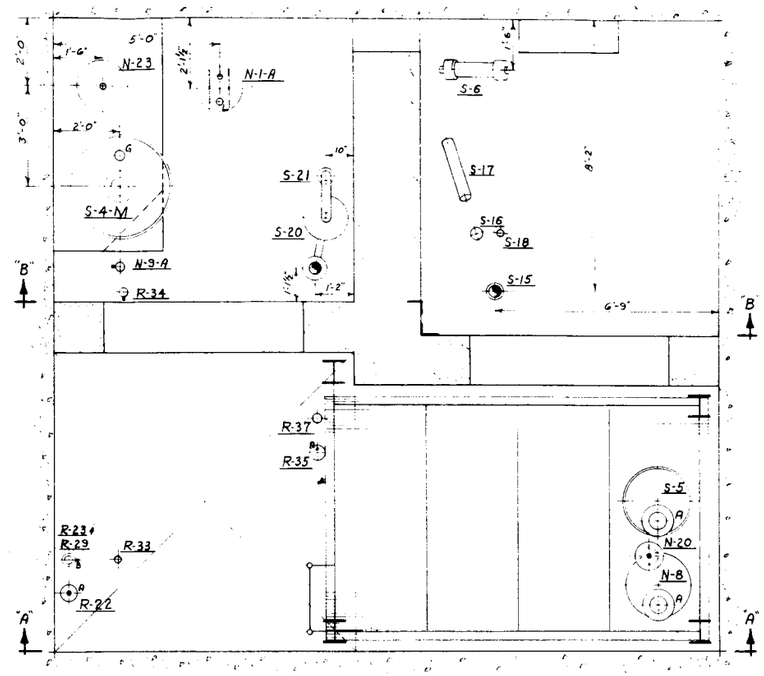


Fig. 19. Plot of Solution Concentration Required to Contain 60 kg of Uranium vs Thickness of the Solution Layer Above the Ring-Packing Depth on the Cell Floor.



LOWER PLAN



UPPER PLAN

NOTES

1- ALL ITEMS SHOWN ON DRAWINGS CP-059-D AND CP-060-D ARE APPROXIMATE LOCATIONS.
2- DRAWINGS LISTED IN NOTE 1 SUPERSEDES DRAWINGS D-11705 AND D-11706.

Out of Service	In Service
R-36, R-37	N-1-A 24% glass in top section
S-5 (DH)	N-4
S-6	N-7, N-8 24% glass packing
	N-9-A, N-10, N-12, N-20, N-23
	R-25 24% glass packing
	R-22, R-23, R-25-P, R-26
	R-29, R-31, R-32, R-33, R-34, R-35
	S-2, S-4, S-4-P
	S-15, S-16, S-17, S-18
	S-20
	S-21 40% glass packing

DH = Drain Hole

REFERENCE DRAWINGS	NUMBER
DRAWING INDEX SHEET No. 2	CP-054-D
DRAWING INDEX SHEET No. 1	CP-053-D
EQUIPMENT LAYOUT SECTION A-A B-B CELL V	M-12229-CP-060-D

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Fig. 20. Equipment Layout in Cell 5.

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DRAWN	DATE	SUBMITTED	DATE	APPROVED	DATE
GREGORY	12-21-70				
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
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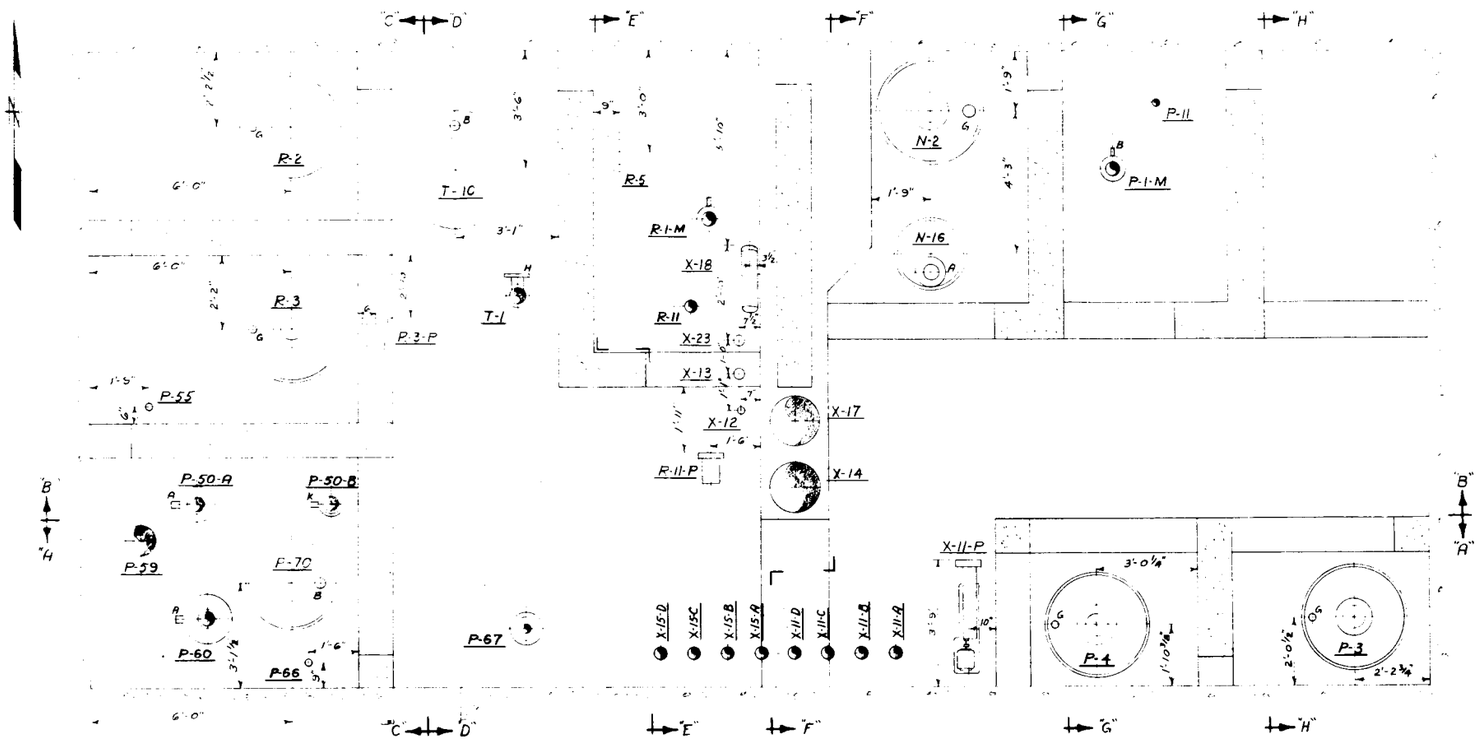

FLOOR PLAN

NOTES

THIS DRAWING AND THOSE LISTED AS REFERENCE DRAWINGS SUPERSEDES THE FOLLOWING DRAWINGS D-11737, D-11738, D-23533, D-23534, D-23535 AND D-23536.

DRAWING D-23533 HAS GENERAL NOTES STATING TYPE OF SUPPORTS TO BE SUPPLIED BY FIELD.

Out of Service	In Service
P-3	N-2
P-4	N-16 24% glass packing
P-50A (DH)	P-1-M 24% glass in end sections
P-50B (DH)	P-11, P-70
P-55	R-1-M 24% glass in end sections
P-59	R-2, R-3 24% glass packing
P-60 (DH)	R-3-P
P-67	R-11, R-11-P
R-5	X-11, X-15 (A, B, C, D)
T-1 (DH)	X-12, X-13, X-23, X-18
T-10	X-14, X-17 34% glass packing

DH = Drain Hole

REFERENCE DRAWINGS	NUMBER
ELEVATIONS "B-B", "D-D"	CP-051-D
ELEVATIONS "E-E", "F-F", "G-G"	CP-050-D
ELEVATIONS "A-A", "C-C"	CP-049-D
PLATFORM LEVEL & ELEVATION "H-H"	CP-048-D

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Fig. 21. Equipment Layout in Cells 6-7 at Floor Level.

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DRAWN	DATE	SUBMITTED	DATE	APPROVED	DATE
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
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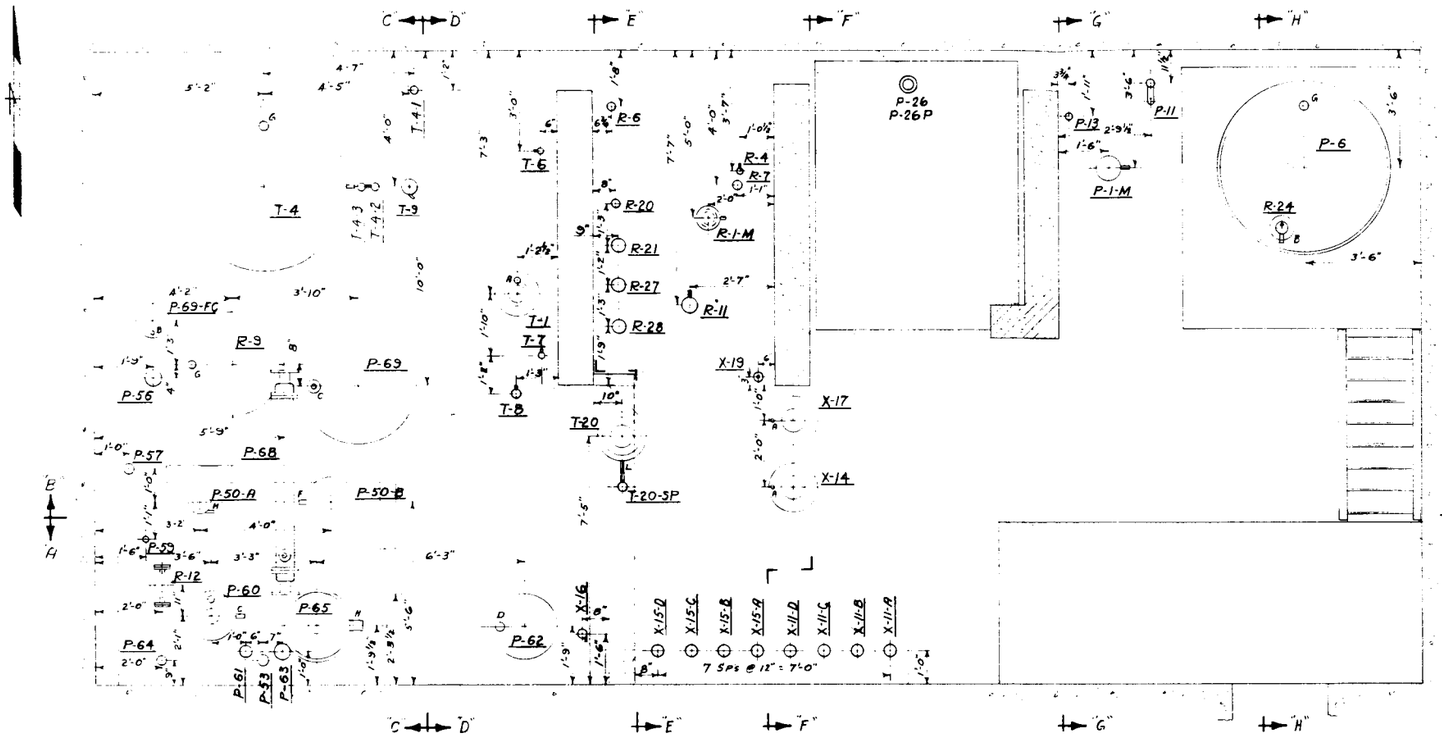
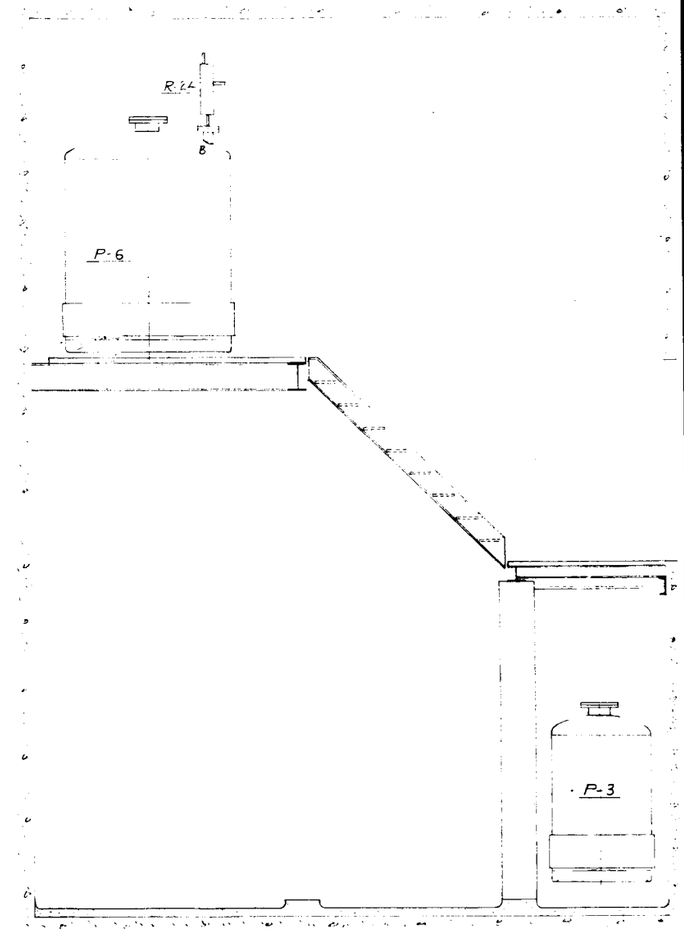
SUBMITTED 4/27/71 ACCEPTED 4/27/71 APPROVED

M 12229 CP 047 D



FLOOR AT PLATFORM LEVELS



ELEVATION "H-H"

Out of Service	In Service
P-50A (DH)	P-1-M 24% glass in end sections
P-50B (DH)	P-6 24% glass 6" deep
P-53, P-56	P-11
P-57, P-59	P-26, 26-P
P-60 (DH)	P-69, 69-FC 24% glass packing
P-61, P-62, P-63	R-1-M 24% glass in end sections
P-64, P-68	R-4, 7, 11, 20, 21, 27, 28
R-6, R-9, R-12	T-20
T-1 (DH)	T-20 SP
T-4, T-6, T-7	X-11, X-15 (A, B, C, & D)
T-8, T-9	X-14, 17 34% glass packing
DH = Drain Hole	

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	DRAWN Gregory	DATE 10-6-70	SUBMITTED FEH	DATE 4/20/71	APPROVED	DATE	
	DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE	
	CHECKED	DATE	APPROVED	DATE	APPROVED	DATE	

FLOOR PLAN CELL VI & VII — CP-047-3
REFERENCE DRAWINGS — NUMBER

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Fig. 22. Equipment Layout in Cells 6-7 at Platform Level.

SUBMITTED F.E.H. 4/20/71 — ACCEPTED 4/21/71 — APPROVED

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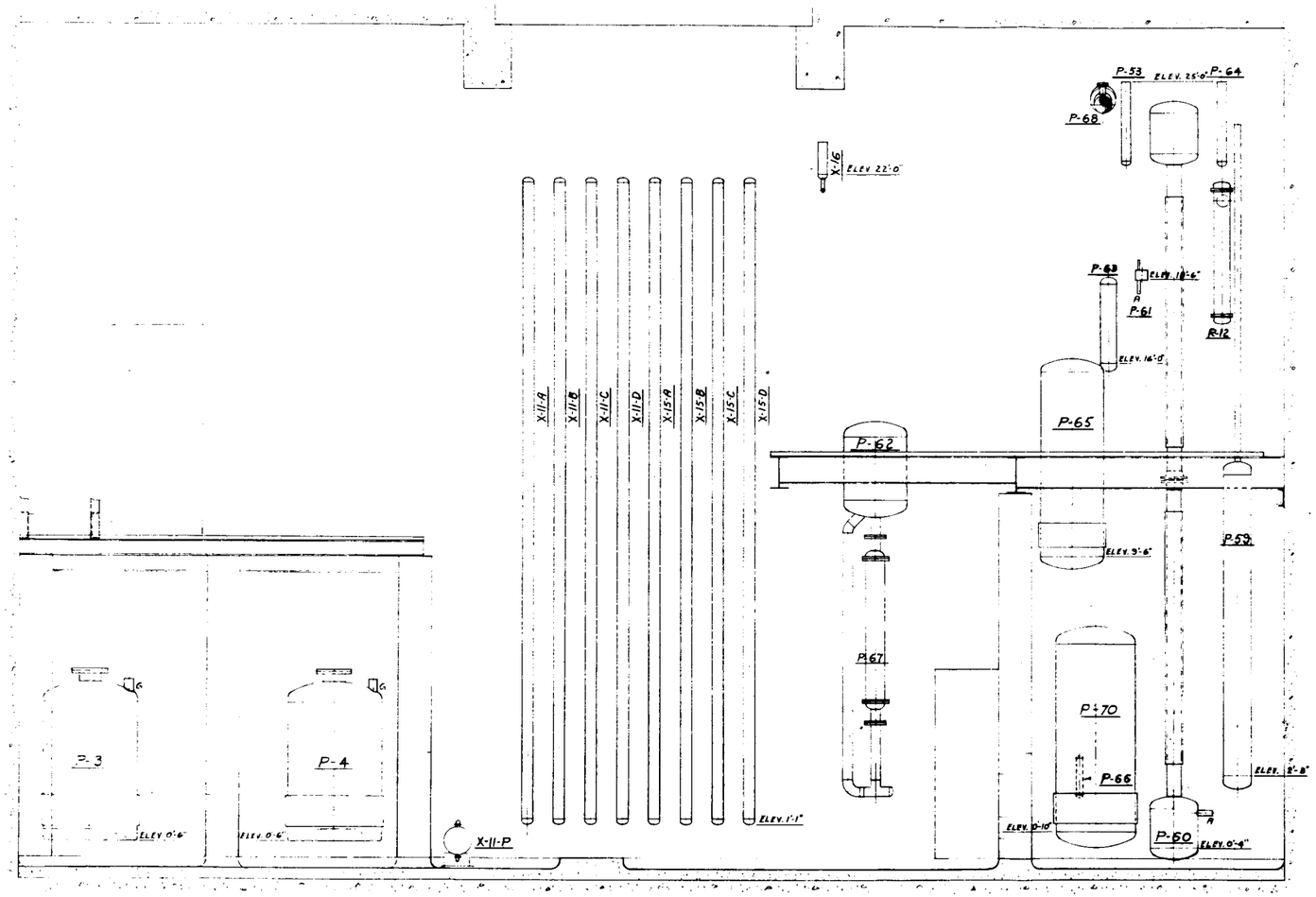
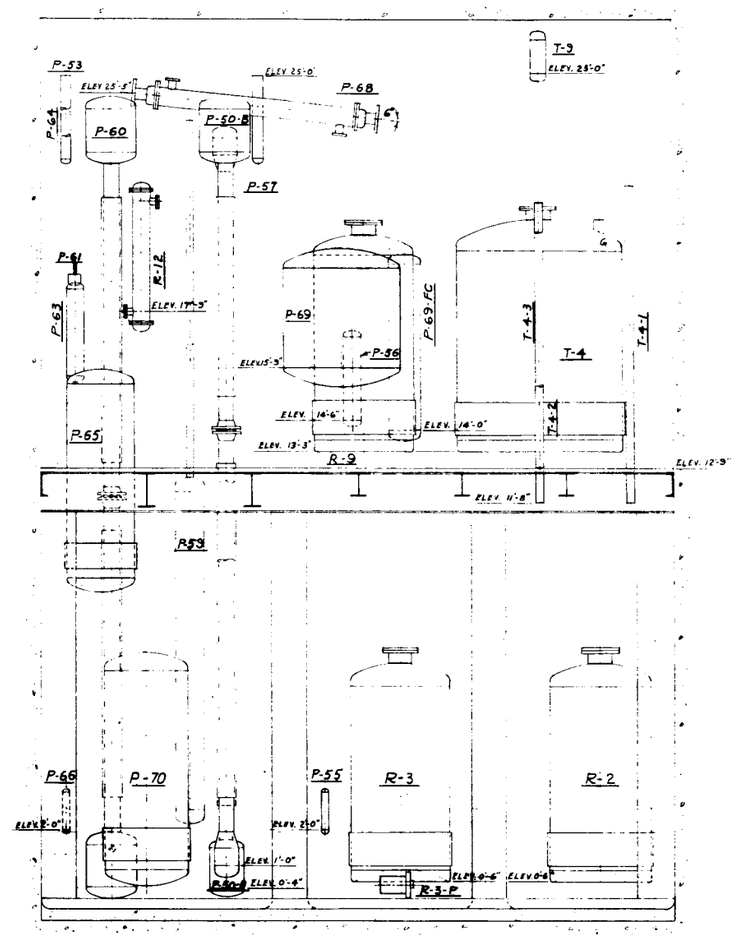


ELEVATION "A-A"



ELEVATION "C-C"

FLOOR PLAN CELL VI & VII	CP-047-D
REFERENCE DRAWINGS	NUMBER

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Fig. 23. Equipment Layout in Cells 6-7 Elevations A-A and C-C.

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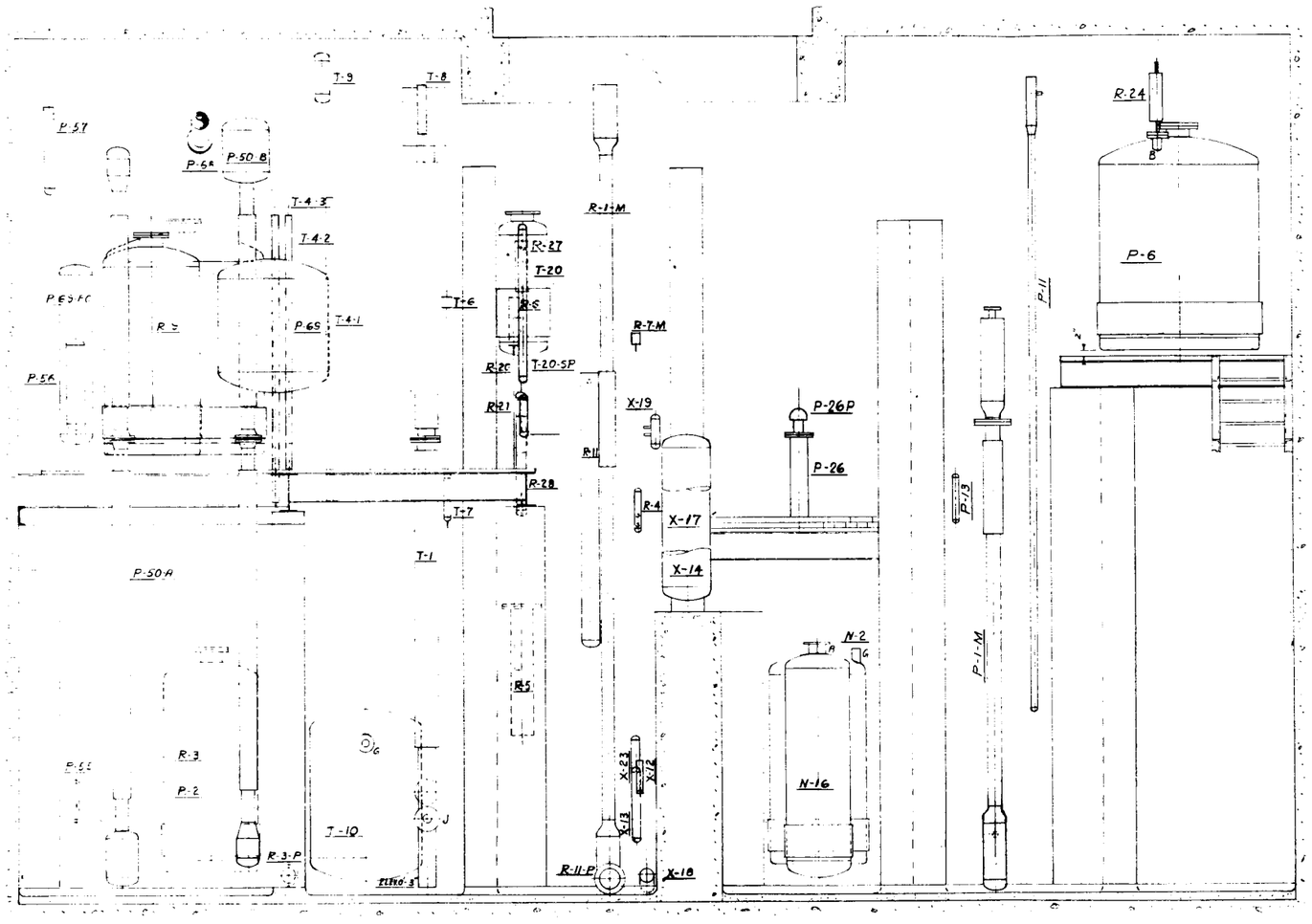
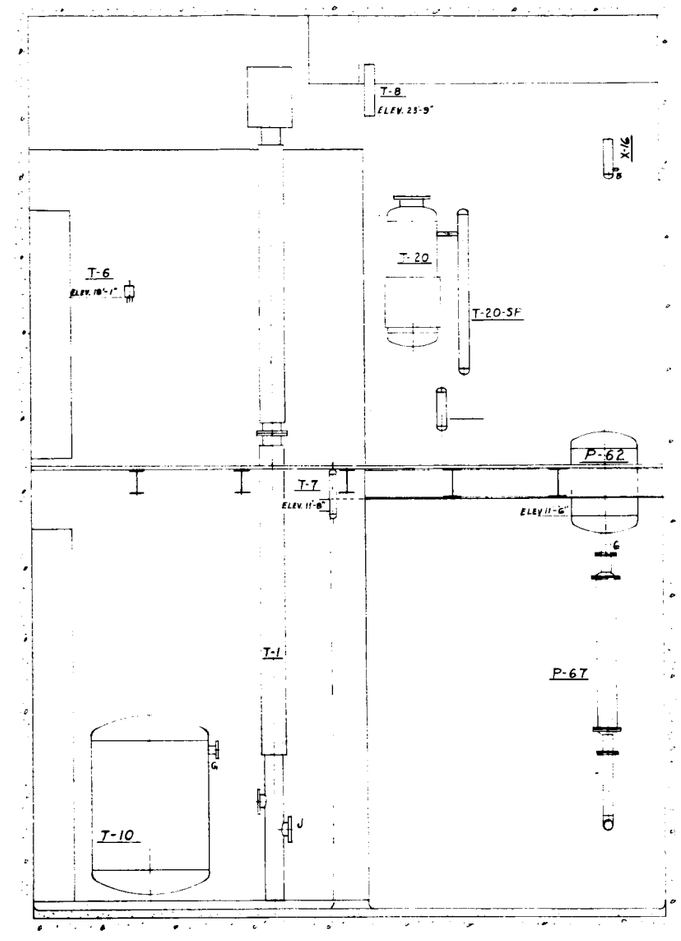
GREGORY	10-6-70	FFH	4/29/71		
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE

SUBMITTED FFH 4/29/71 ACCEPTED 4/29/71 APPROVED

M 12229 CP 049 D REV.



ELEVATION "B-B"



ELEVATION "D-D"

FLOOR PLAN CELL VI & VII	CP-047-D
REFERENCE DRAWINGS	NUMBER

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Fig. 24. Equipment Layout in Cells 6-7 Elevations B-B and D-D.

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DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE

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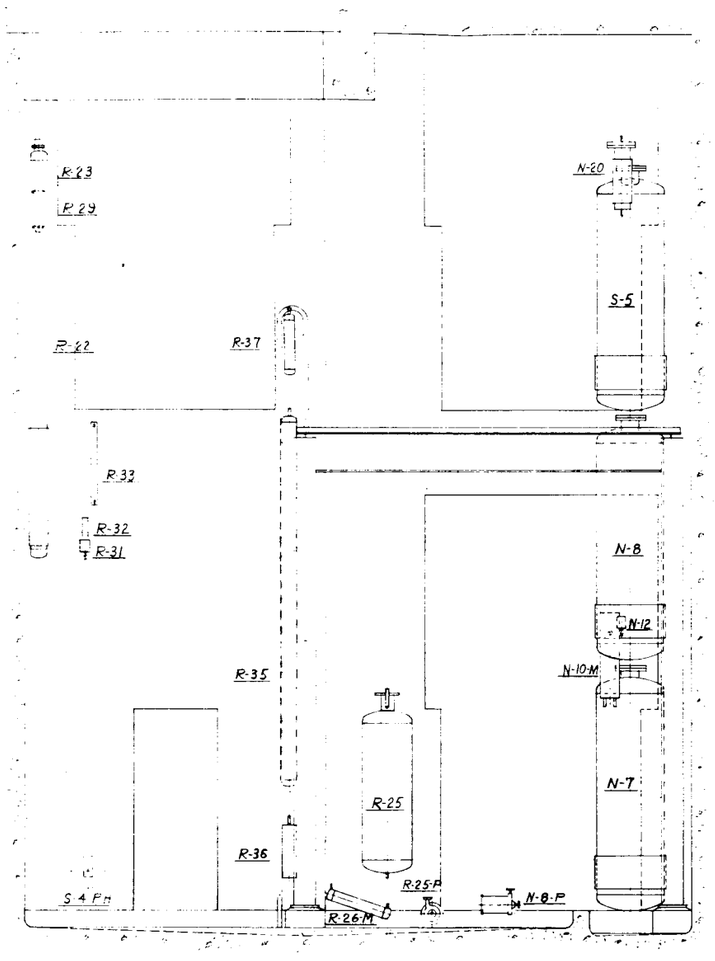
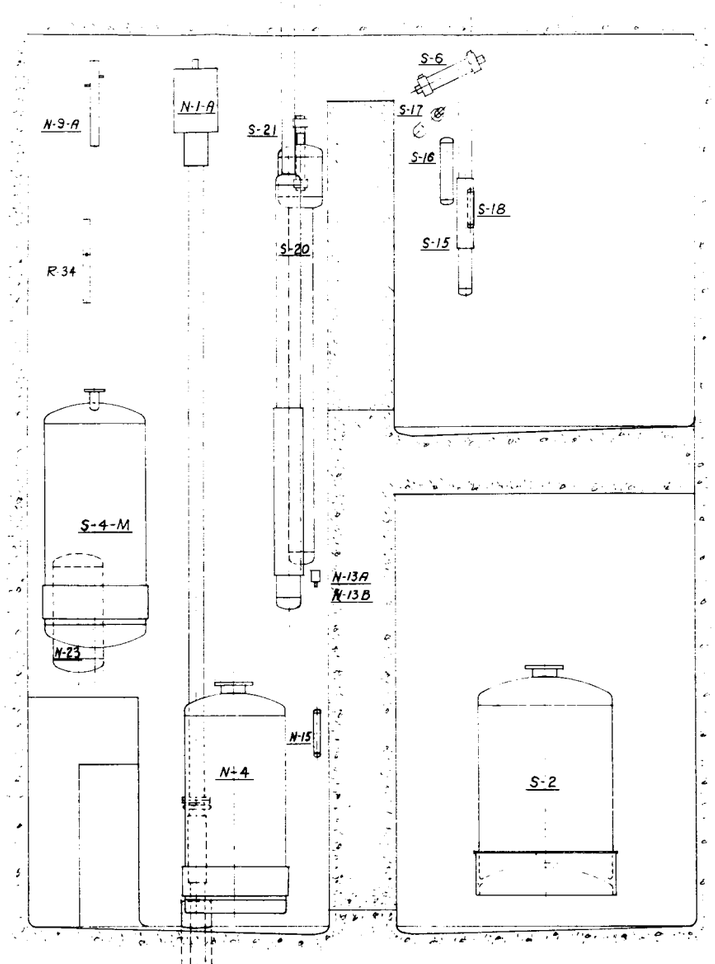
M- 12229 CP 051 D



SECTION "A-A"



SECTION "B-B"

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ANGLES ± —
SCALE: 1/2 : 1'-0"

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EQUIPMENT LAYOUT FLOOR PLANS CELL V — M-12229-CP-059-D
REFERENCE DRAWINGS — NUMBER

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Fig. 25. Equipment Layout in Cell 5 Elevations A-A and B-B.

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10. APPENDIXES

10.1 Appendix A: Criticality Calculations

D. W. Magnuson*

A.1 Calculational Methods

Criticality calculations in support of the nuclear safety evaluation of the Building 3019 Pilot Plant have been performed with either the ANISN Transport Code[1] for one-dimensional problems or the KENO Monte Carlo code[2] for two- or three-dimensional geometry. Hansen-Roach 16-group cross sections[3] were used in both codes. For the reasonance absorbers, the cross-section set was selected to be appropriate for the scattering cross section of the medium per absorber atom. The ANISN code was also used to generate cell-averaged cross sections[4] for input to other calculations and also to calculate the neutron age to thermal energy. The nonleakage probability, $P_{n\ell}$, was estimated for some of the single vessels from the approximate one-group equation. Using $P_{n\ell}$ and the calculated value of k_∞, k_{eff} was determined from the following relationship:

$$k_{eff} = \frac{k_\infty}{(1 + M^2B^2)} ,$$

where

M^2 = the migration area, cm^2; $M^2 = L^2 + \tau$, the sum of the diffusion length squared and the neutron age,

B^2 = the buckling, cm^{-2}.

A.2 Raschig Ring-Solution Systems

A calculational model,[4] the Cylindrical Tube Model, had been developed to evaluate some exponential experiments with raschig rings and fissile solutions. This model consisted of a cylinder of solution, an annulus of glass, and an annulus of solution, all infinite in length. A mirror boundary condition was placed on the outside surface which returned all neutrons; there was no leakage of neutrons. In this model inside and outside radii

*Y-12 Plant.

of the glass tube were made equal to the raschig ring radii, and the radius of the outside solution is calculated from the volume of solution outside the glass measured in the tanks filled with raschig rings. Therefore, the volume fraction of glass and the volume fractions of solution inside and outside the glass correspond to those for vessels filled with raschig rings. Only 90% of the normal boron in Pyrex glass, corresponding to 3.67 wt % boron (instead of 4%), was used in these calculations. The calculated k_∞ values using the ANISN-S_8 transport code as a function of ^{233}U concentration for various glass volume fractions are given in Fig. A-1 for 1.5-in.-OD borosilicate-glass raschig rings in uranyl nitrate solution.[4] For some of these k_∞ calculations, the output of the ANISN code (which provided cell-averaged cross sections) was used so that further calculations could be performed in other geometries, thereby simulating raschig-ring-filled solution storage or process tanks.

A.3 Uranyl Nitrate Storage Tanks

The $^{233}UO_2(NO_3)_2$ solutions were diluted to a ^{233}U concentration of 250 g/liter and transferred to the ten 36-in.-diam tanks, U-1 through U-10, upon arrival from Hanford. Criticality calculations were made for this system. Finite geometry calculations for a cylindrical unreflected tank and for the ten-tank arrangement were made with the KENO Monte Carlo code, using the cell-averaged cross sections for the solution-glass mixture from a previous ANISN calculation with the Cylindrical Tube Model. Certain approximations were made in order to simplify the calculation for the tank arrangements. Each cylindrical tank was 36 in. ID and 63.4 in. high. The thickness of each stainless steel tank wall was assumed to be 3/16 in., although some walls were 1/4 in. thick. All center-to-center distances between the tanks were assumed to be the smallest dimension (38 in.). The lead between five tanks was ignored. Both the 4-in.-lead shield on two sides of the ten-tank arrangement and the concrete reflector on all sides were included. The results of these calculations are summarized in Table A-1.

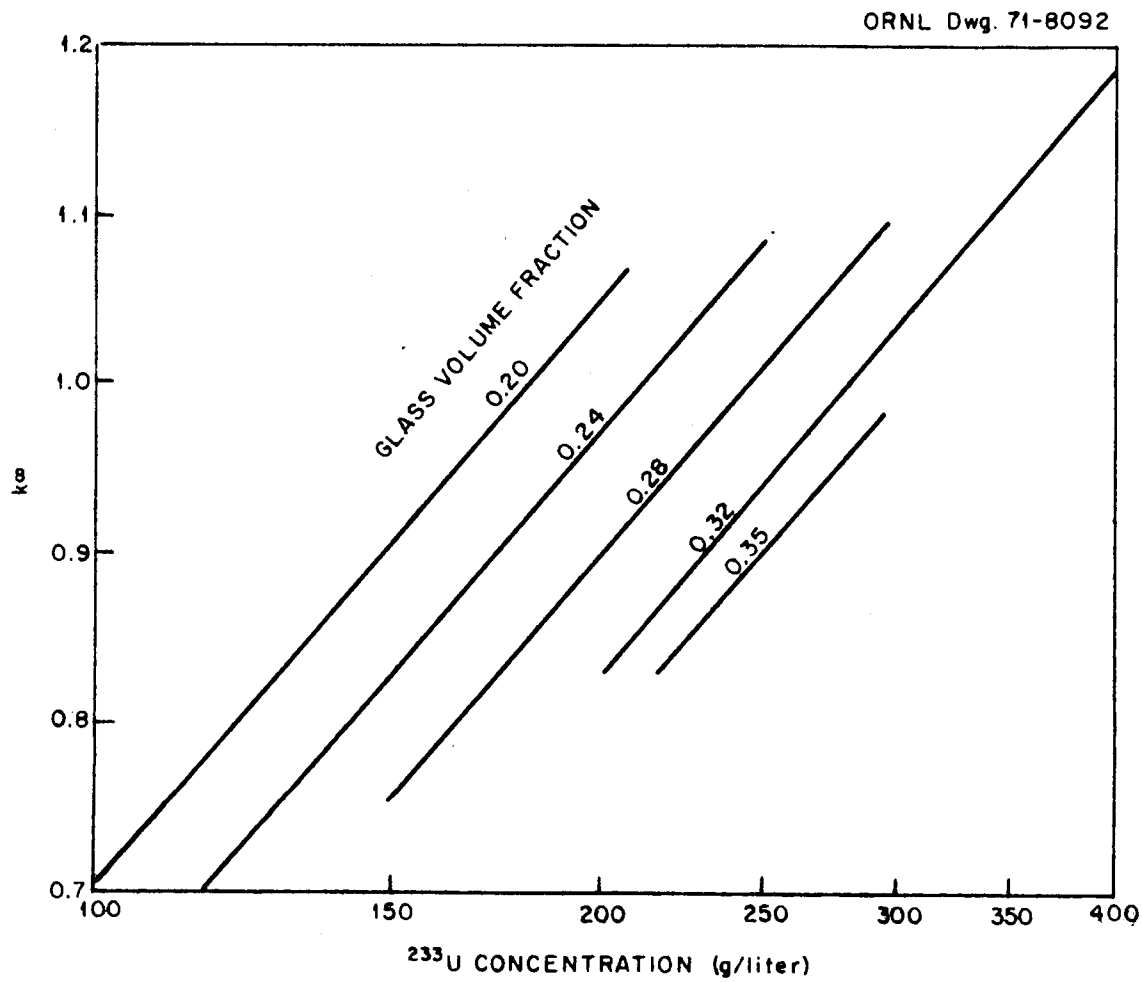

Fig. A-1. Calculated Values of k_∞ as a Function of ^{233}U Concentration in Uranyl Nitrate Solutions Containing 1.5-in.-OD Borosilicate-Glass Raschig Rings.

Table A-1. Calculated Multiplication Factors for the $^{233}UO_2(NO_3)_2$--35 vol % Raschig Ring Storage Facility at 250 g of ^{233}U per liter

Code	System	
ANISN-Cylindrical Tube Model	Infinite	$k_\infty = 0.9050$
KENO	1 tank unreflected[a]	$k_{eff} = 0.8066 \pm 0.0018$
KENO	10 tanks in cell	$k_{eff} = 0.8442 \pm 0.0078$

[a]The k_{eff} value is estimated to be 0.810 from $k_{eff} = k_\infty/(1 + M^2B^2)$, where $M^2 = 49\ cm^2$ and B^2 is estimated using an extrapolation length of 7 cm.

A.4 Transfer Tank R-25

The collection or transfer tank, R-25, was filled with glass raschig rings containing 24 vol % glass. Using the KENO code, the calculated k_∞ for 250 g of ^{233}U solution and the above glass[4] was 1.0863, and the calculated k_{eff} for this tank located 6 in. from one wall and 18 in. from the floor in a large concrete cell was 0.8200 ± 0.0024.* It is seen that the neutron leakage from this tank or vessel assures subcriticality for the concentration range up to 250 g of ^{233}U per liter.

A.5 Storage Tanks R-2 and R-3

The maximum concentration of ^{233}U for storage tanks R-2 and R-3, each of which contains 24 vol % glass, is expected to be 180 g of ^{233}U per liter, corresponding to a calculated k_∞ of 0.92 for glass having 90% of the standard amount of boron. Neutron leakage from these tanks, 36 in. OD and 66 in. high, is estimated to be 0.083. (This is based on an extrapolation distance of 7 cm and a migration area of 38 cm^2.) The k_{eff} for each tank is estimated to be 0.84. The storage tanks are believed to be subcritical by an amount perhaps four or five times the uncertainty in the calculation.

*A value of 0.86 is calculated from $k_\infty/(1 + M^2B^2)$, where $M^2 = 38\ cm^2$, a value calculated for 24% glass--solution mixtures,[4] and B^2 is evaluated by assuming that the extrapolation distance, λ, is 7 cm.

The ^{233}U concentration entering tanks R-2 and R-3 from the evaporator (recycle solutions) or from any other source should not be permitted to be higher than 180 g/liter.

A.6 Miscellaneous Waste Catch Tank N-16

Miscellaneous Waste Catch Tank, N-16, is 24 in. in diameter and 72 in. long and is filled with 24 vol % borosilicate-glass raschig rings. For 200 g of ^{233}U solution, $k_\infty = 0.97$. The k_{eff} is estimated to be 0.83 from the one-group approximation. Leakage of neutrons from this vessel ensures subcriticality for ^{233}U concentrations up to 200 g/liter.

A.7 Oxide Storage

Preliminary calculations were made, using the ANISN-S_8 transport code, of the critical infinite cylinder radii for $^{233}UO_2$-water mixtures reflected by water. These results are given in Table A-2 and shown in Fig. A-2. Comparison of these calculations with Fig. 36 in TID-7028[5] shows differences only in the undermoderated region where metal and oxide density differences are important for determining critical sizes. Additional calculations were performed for the infinite cylinder critical radii for dry oxide at one-third and one-half of the theoretical density as well as k_{eff} calculations for various mixtures having a diameter of 3.75 in. A KENO calculation for an infinite array of 3.75-in.-diameter $^{233}UO_2$ cylinders in concrete (using oxide at one-third of the theoretical density and arranged on a triangular pitch of 18 in.) resulted in a k_∞ of 1.10 ± 0.02. Although the nine holes of one storage arrangement were not considered to be infinite, it was apparent that this thickness of concrete permits considerable interaction and does not isolate the units.

The one-dimensional cylindrical model given in Table A-3 was used for a more accurate mockup of the storage hole. The $^{233}UO_2$ has double containment in aluminum cans. The inside diameter was reduced to 3.25 in., and the maximum amount of water in the oxide and voids was included for moderation. The amount of concrete was reduced from the equivalent hexagonal cell area radius of 24.00 cm to 20.16 cm in order to provide conservatism

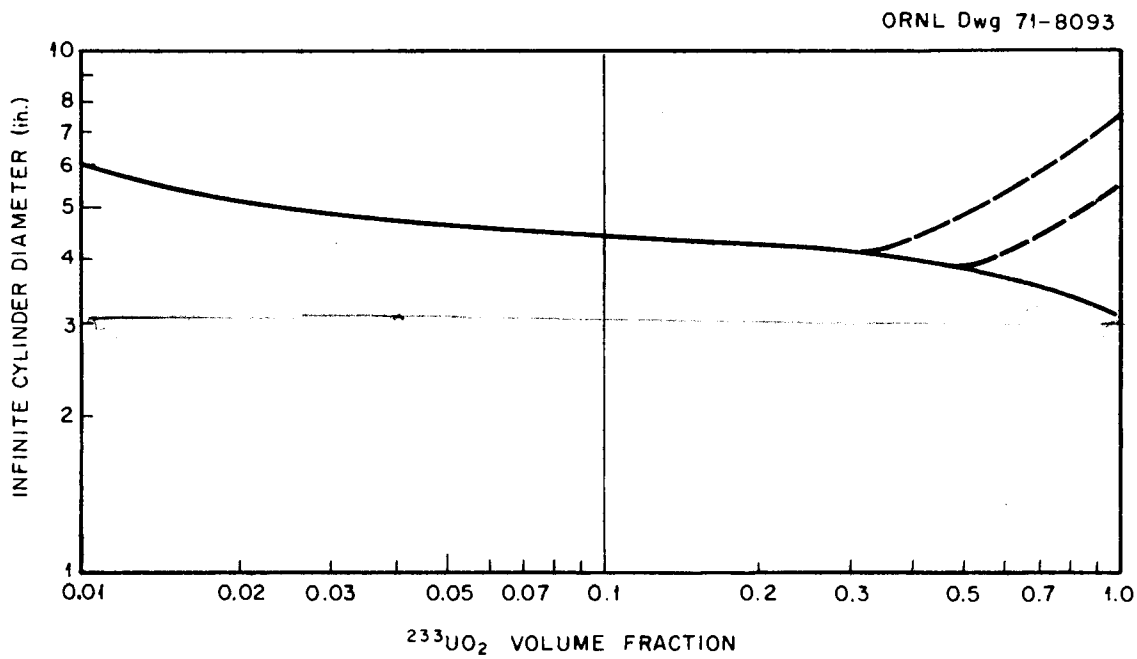

Fig. A-2. Calculated Critical Diameters for Water-Reflected Infinite Cylinders of $^{233}UO_2$-H_2O.

Table A-2. Calculated Critical Radii and k_{eff} Values for Water-Reflected Infinite Cylinders of $^{233}UO_2$-H_2O Mixtures

$^{233}UO_2$			H/U	Infinite Cylinder		
Density (g/cm^3)	Volume Fraction	Water Fraction	Atom Ratio	Radius (cm)	Diameter (in.)	k_{eff}
3.57	1.0	0.0	0.0	9.549	7.519	1.0000
5.35	1.0	0.0	0.0	6.966	5.485	1.0000
10.70	1.0	0.0	0.0	4.056	3.194	1.0000
10.70	0.5	0.5	2.745	4.956	3.902	1.0000
10.70	0.2	0.8	10.98	5.471	4.308	1.0000
10.70	0.1	0.9	24.70	5.662	4.458	1.0000
10.70	0.05	0.95	52.14	5.907	4.651	1.0000
10.70	0.02	0.98	134.2	6.579	5.180	1.0000
10.70	0.01	0.99	271.6	7.610	5.992	1.0000
10.70	0.3333	0	0.0	4.76	3.75	0.7071
10.70	0.3333	0.0667	0.5868	4.76	3.75	0.7354
10.70	0.3333	0.6667	5.868	4.76	3.75	0.9428

Table A-3. One-Dimensional Cylindrical Model of the ^{233}U Storage Hole

Region	Material	Cylinder Radius (cm)
1	33.33 vol % $^{233}UO_2$ -- 66.67 vol % H_2O	4.13
2	Aluminum	4.23
3	H_2O	4.29
4	Aluminum	4.39
5	H_2O	5.11
6	Stainless steel	5.71
7	Lead	10.16
8	Concrete	20.16

in the calculations of k_∞. The calculated k_∞ and k_{eff} values, with and without a perfect reflector outside the concrete, were 1.0909 and 0.8608, respectively. (ANISN-S_8 was used for these calculations.)

The KENO geometry routine was used to describe the storage arrangement for nine cells in concrete on a triangular pitch of 18 in. In a rectangular array of 54 concrete cells, the ^{233}U storage cells can be described by five cells of one type and four of another, differing only in the location of the cylinders within the cell. This geometry is depicted in Fig. A-3. The z dimension was 15 ft, the depth of the hole for this calculation. The cylindrical dimensions of the storage cell and materials were the same as those given in Table A-3. The k_{eff} was calculated to be 0.953 $\pm$ 0.008. Leakage from the concrete reflector was 0.0174.

Because there are no experimental data to provide validation of the calculational method for the conditions of low or no moderation of $^{233}UO_2$ oxide, the storage under these maximum conditions is not recommended. The assumed conditions for the calculations were:

1. All cylinders were filled with $^{233}UO_2$ having a density of 3.57 g/cm^3, one-third of the theoretical density.
2. All $^{233}UO_2$ voids were filled with H_2O (i.e., 66.67 vol % H_2O).
3. The space between aluminum cans was filled with H_2O.
4. The space between the outer aluminum can and the stainless steel hole liner was filled with water.

Although it is unlikely that all of these maxima will be met simultaneously in all storage cells, restrictions placed on the water content in the oxide would provide an adequate factor of safety to guarantee subcriticality. For the maximum oxide density of 3.57 g/cm^3 having the water content reduced from 67 vol % to 6.7 vol %, the calculated Δk_∞ was -0.21.

The storage arrangement should have the following restrictions:

1. The maximum inside diameter of the can shall be 3.25 in., and the maximum density of the $^{233}UO_2$ shall be 3.5 g/cm^3, thereby resulting in a maximum mass of 5.7 kg of $^{233}UO_2$ per foot.

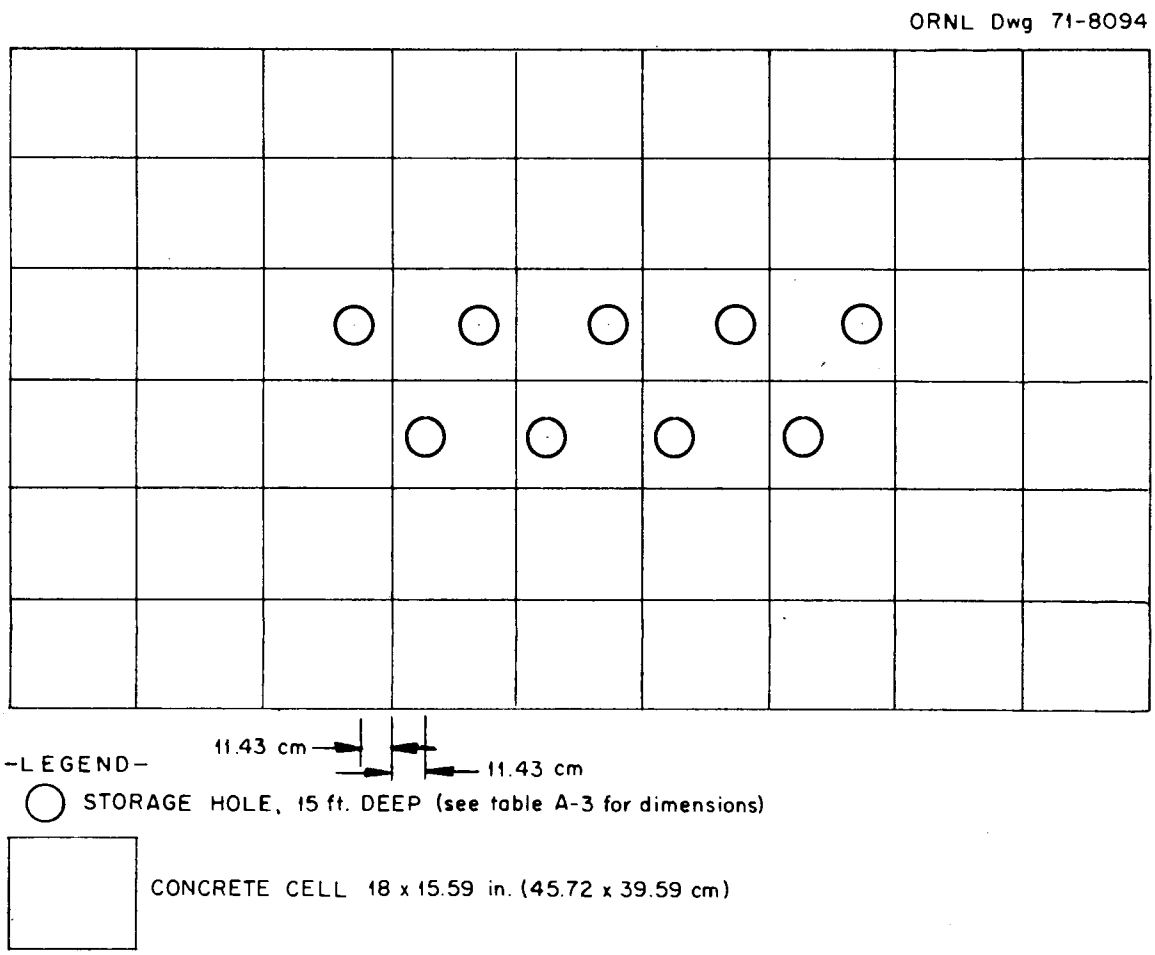

Fig. A-3. Geometry for KENO Criticality Calculations of Nine Storage Holes in Concrete.

2. No significant quantities of liquid water shall be permitted in the storage holes.

3. The H/U atom ratio in the oxide shall be less than 0.4.

Calculations were made for the storage holes in the concrete walls between cells which make a 9-unit linear array. The KENO Monte Carlo code with Hansen-Roach cross sections was used, and the results are given in Table A-4.

Table A-4. Calculated k_{eff} Values for a Linear Storage Arrangement in Concrete, 9 Units on 12-in.-Centers

Description	Oxide Density (g/cm^3)	H/U Atom Ratio	k_{eff} for 9-Unit Array
Double can with H_2O[a]	3.57	0	0.743 ± 0.005
Double can with H_2O[a]	2.14	0	0.634 ± 0.005
Double can with H_2O[a]	3.57	5.868	0.957 ± 0.006
Single can with H_2O[b]	3.57	0	0.826 ± 0.005

[a]See Table A-3 for the dimensions of the cylinders. There is no lead surrounding the stainless steel in these storage holes.

[b]Container has an OD of 3.874 in. (9.84 cm) and an ID of 3.748 in. (9.52 cm), with H_2O filling the storage hole.

These calculations confirm the previous results, which indicated that water mixed with the low-density oxide powder should not be tolerated. The dry oxide at expected densities no greater than 3.5 g/cm^3 and a diameter of 3.25 in., when suitably contained to assure that the oxide powder remains unmoderated, can be stored in all the holes. Oxides having higher densities shall be stored only when the total oxide mass is no greater than 5.7 kg per foot of storage.

A.8 Scrap Dissolver

Preliminary calculations were made for mixed oxide--water mixtures in which the ^{233}U content in the (Th + U) was 5.8 wt %. The critical radii, reflected by either 3 cm of stainless steel or 20 cm of water, were calculated as a function of the volume fraction of mixed oxide. The results of these calculations are given in Fig. A-4. It is seen that the minimum critical radius is obtained for a mixed oxide volume fraction of 0.20 when the ^{233}U concentration is 102.7 g/liter. This density of oxide, about 2.0 g/cm^3, is somewhat greater than the densities of some of the loose powders but much less than those to be expected in the pelleted oxides.

The calculational model or mockup of the dissolver in a concrete vault can be described as follows:

1. The dissolver consisted of a 10-ft length of 8-in. sched. 40 stainless steel pipe whose inside diameter was increased by 0.040 in., thereby allowing for three times the estimated corrosion of 6 to 7 mils.
2. The steam jacket consisted of a 10-ft length of 10-in. sched. 40 stainless steel pipe.
3. The return or cold leg consisted of a 10-ft length of 8-in. sched. 40 stainless steel pipe.
4. The center-to-center separation of the above components was 18 in.
5. The center of the dissolver was located 9 in. from the walls of one corner in a concrete vault, 6 x 7-1/2 x 10 ft; the cold leg was also located 9 in. from one of the walls.
6. The ^{233}U content of the (Th + U) mixture was 5.8 wt %.
7. The dissolver, steam jacket, and cold leg were filled with 20 vol % of UO_2-ThO_2 and 80 vol % of 5 M HNO_3 aqueous solution, the mixture that was found to have the smallest critical infinite cylinder diameter from previous ANISN calculations. This mixture contains 102.7 g of ^{233}U per liter.
8. Hansen-Roach 16-group cross sections were used.

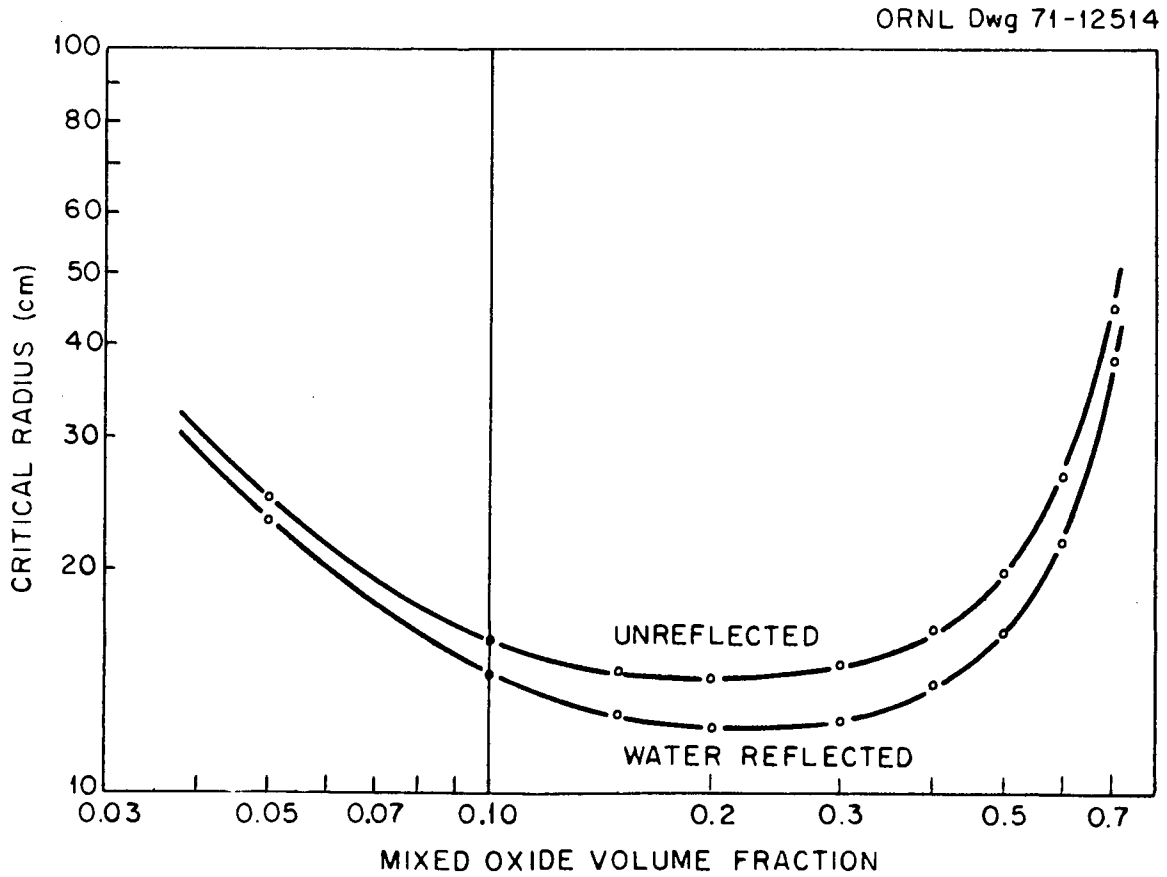

Fig. A-4. Calculated Critical Radius for 5.8 wt % ^{233}U in (U + Th) for Mixed Oxide--Water Mixtures.

9. The ^{233}U cross section set corresponded to the scattering of the mixture.

10. The thorium cross-section set was for unmoderated systems; this underestimates the resonance absorption.

11. The reflector albedo for 60-in.-thick concrete was used.

The k_{eff} for the above mockup of the dissolver was calculated to be 0.802 ± 0.006 by the KENO Monte Carlo code. This is sufficiently subcritical to ensure that the dissolver can be operated without restriction on the amount of oxide per batch, since the geometry is calculated to be subcritical by an adequate margin for the most reactive composition of the dissolver contents.

A.9 Dissolver S-15

The calculational model or geometric mockup of the dissolver S-15 in a concrete vault can be described as follows:

1. The dissolver was fabricated from a 6-ft length of 4-in. sched. 40 stainless steel pipe.

2. The steam jacket was made from a 6-ft length of 5-in. sched. 40 stainless steel pipe.

3. The return leg was fabricated from a 6-ft length of 3-in. sched. 40 stainless steel pipe.

4. The center of the dissolver was located at the horizontal mid-plane and in a corner of a 8 x 9 x 18 ft concrete vault, 27 in. from one wall and 15 in. from the other.

5. The center of the return leg was 20 in. and 36 in. from the above walls, respectively.

6. The aqueous solution, which contained 400 g of ^{233}U per liter, filled the dissolver, steam jacket, and cold leg.

7. The reflector albedo for 24-in. concrete was used instead of neutron tracking in the concrete of the vault walls, floor, and ceiling.

The k_{eff} was calculated to be 0.573 ± 0.004 using the KENO Monte Carlo code with Hansen-Roach 16-group cross sections.

For the above concentration of 400 g of ^{233}U per liter, a UO_2 batch charge of 5.5 kg is used. A double batch of 11 kg of UO_2 should not result in criticality because the density of the oxide is less than 2.0 g/cm^3. With this density and all voids filled with water, the calculated critical infinite cylinder diameter reflected by water is greater than 4 in. Additional factors that ensure subcriticality are:

1. Minimal reflection of steel pipe and steam jacket.
2. Neutron absorption in steel if additional water reflection is postulated.
3. Finite length of the S-15 dissolver.
4. Reduced moderation due to hydrogen displacement by nitric acid.
5. Neutron absorption in the nitric acid intermixed with UO_2 or U_3O_8.

A.10 Uranyl Nitrate--Thorium Nitrate--Nitric Acid Aqueous Solutions in Feed Tank S-2

In order to illustrate the changes in k_∞ that accompany concentration changes of the components, calculations have been made of the k_∞ values of various $^{233}UO_2(NO_3)_2$ solutions containing $Th(NO_3)_4$ and HNO_3. (The transport code, ANISN, in the S_8 approximation with 16-group Hansen-Roach cross sections was used for the calculations.) The results are given in Table A-5, and are shown in Figs. A-5, -6, and -7 as plots of the variation of k_∞ vs ^{233}U concentration for 0, 2, and 4 M HNO_3. (These data could also be plotted as a function of thorium or nitric acid content at constant concentrations of the other components.) For the ^{233}U solution containing no $Th(NO_3)_4$ or HNO_3, the limiting critical concentration (at $k_\infty = 1$) is calculated to be 11.25 g of ^{233}U per liter, which is identical to the value extrapolated from critical experiments in large but finite geometry. It is seen that, for 300 g of ^{232}Th per liter and 4 M HNO_3, the calculated critical concentration ($k_\infty = 1.0$) for ^{233}U is increased to 17.9 g/liter.

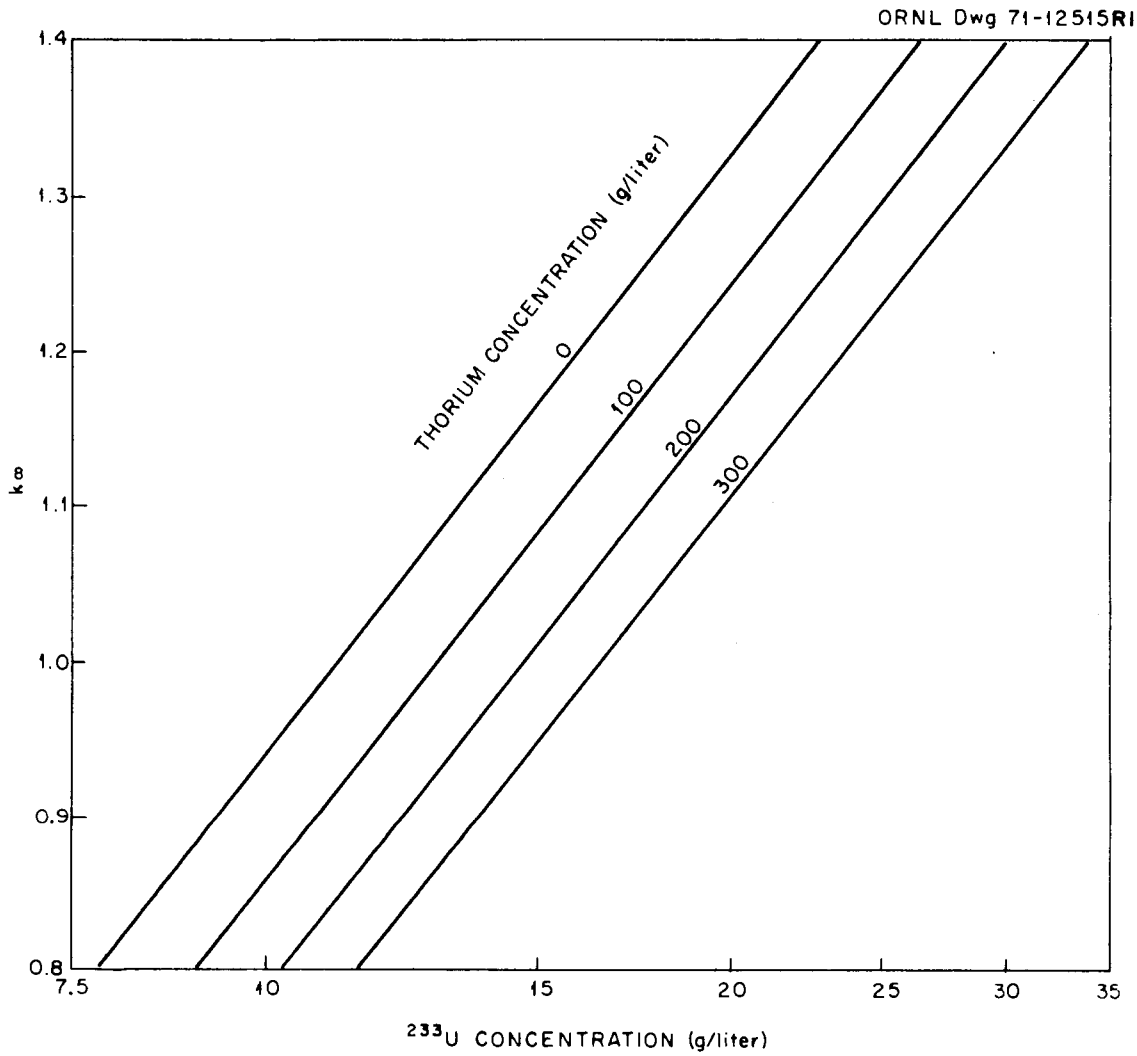

Fig. A-5. Calculated k_∞ Values as a Function of ^{233}U Concentration for 0 M HNO_3.

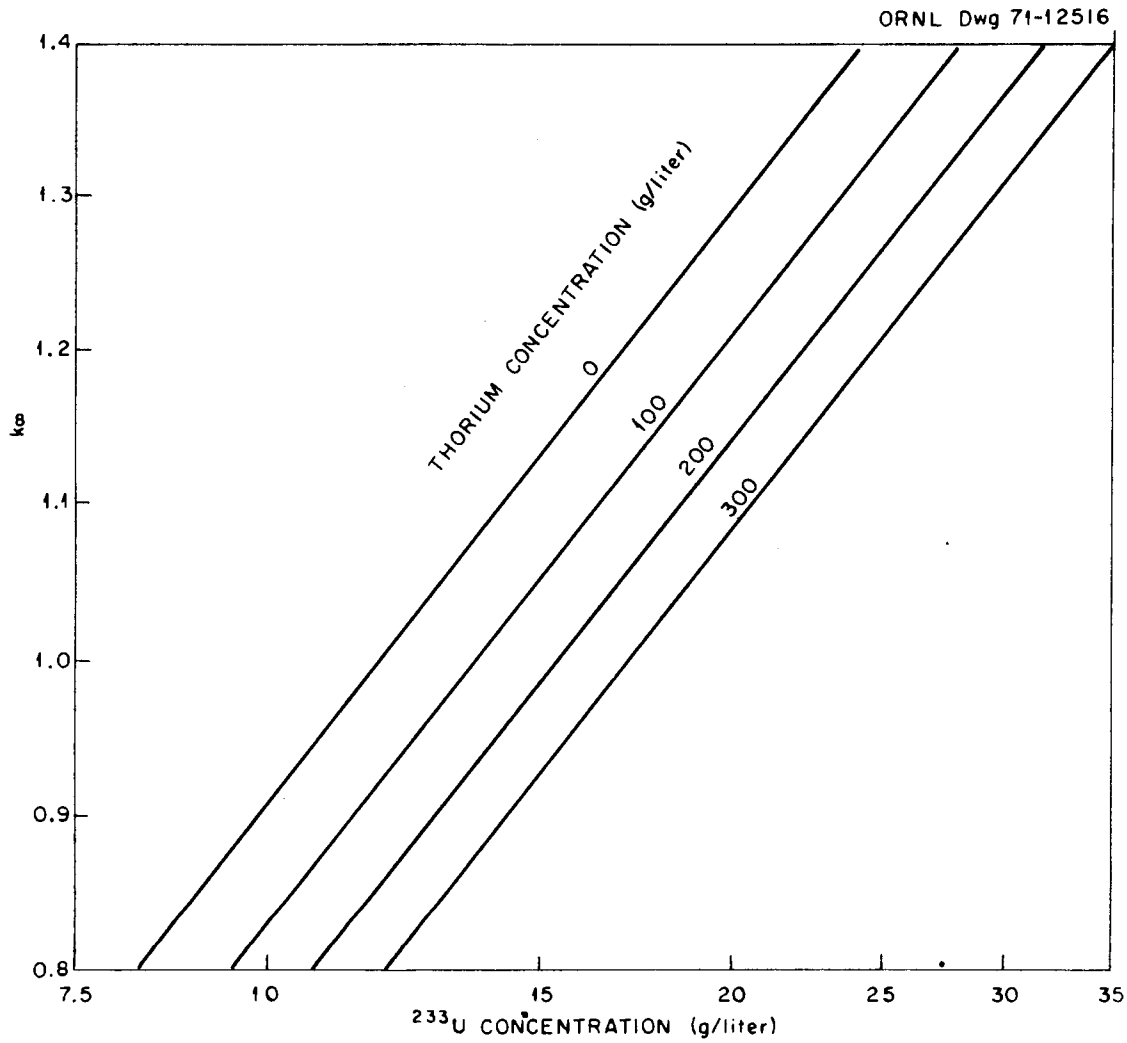

Fig. A-6. Calculated k_∞ Values as a Function of ^{233}U Concentration for 2 M HNO_3.

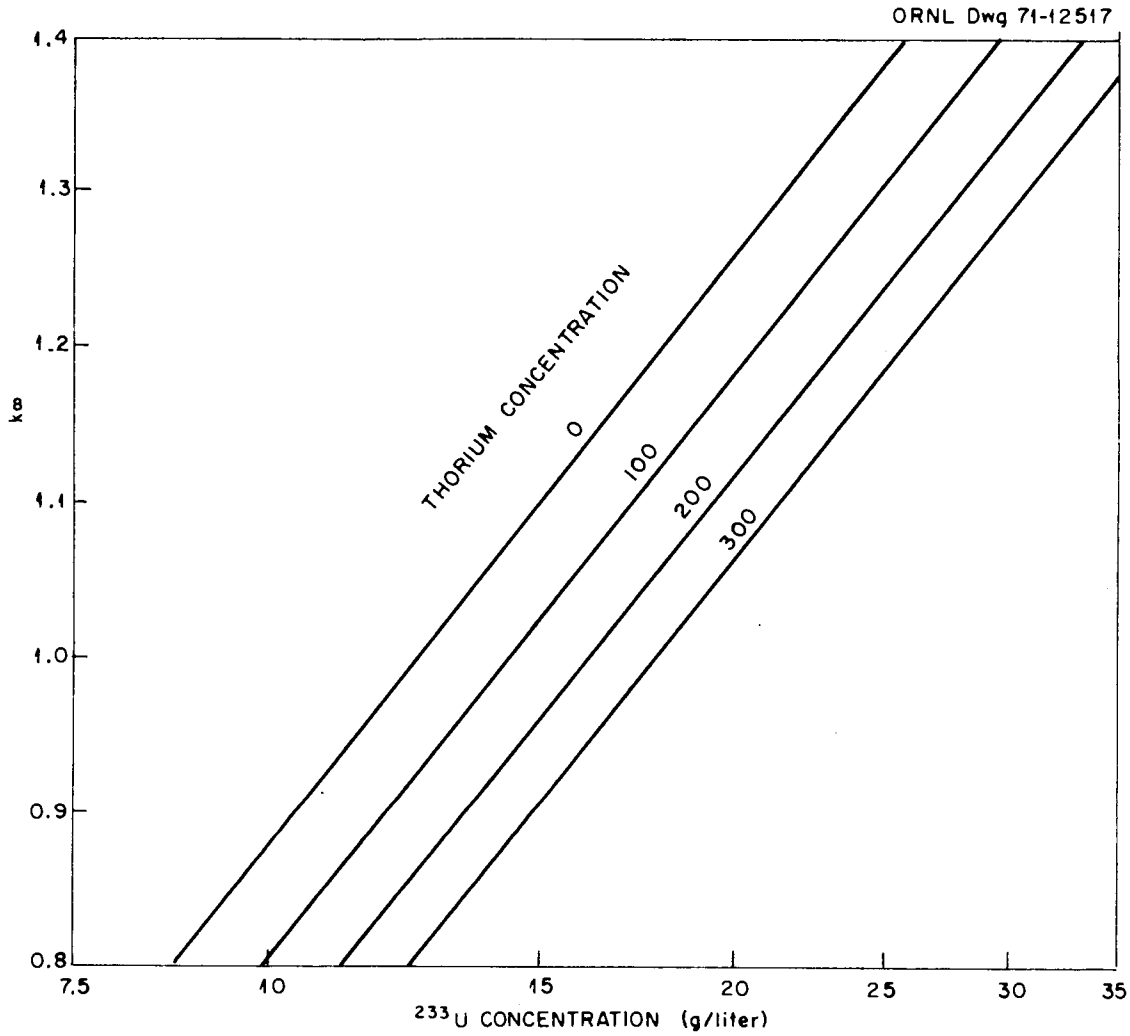

Fig. A-7. Calculated k_∞ Values as a Function of ^{233}U Concentration for 4 M HNO_3.

Table A-5. Values of k_∞ as a Function of ^{233}U Concentration in Aqueous Solutions Containing $^{233}UO_2(NO_3)_2$, $Th(NO_3)_4$, and HNO_3

0 g Th/liter		100 g Th/liter		200 g Th/liter		300 g Th/liter	
^{233}U (g/liter)	k_∞	^{233}U (g/liter)	k_∞	^{233}U (g/liter)	k_∞	^{233}U (g/liter)	k_∞
			0 M HNO_3				
8	0.8166	9	0.8016	10	0.7897	11	0.7803
9	0.8796	10	0.8572	11	0.8397	12	0.8254
10	0.9373	11	0.9090	12	0.8863	13	0.8680
11	0.9906	12	0.9569	13	0.9301	14	0.9081
13	1.0853	14	1.0435	16	1.0462	16	0.9818
16	1.2040	16	1.1197	20	1.1727	20	1.1072
20	1.3296	20	1.2463	24	1.2753	24	1.2102
24	1.4287	24	1.3477	28	1.3601	28	1.2960
			2 M HNO_3				
9	0.8479	10	0.8300	11	0.8159	12	0.8043
10	0.9048	11	0.8811	12	0.8620	13	0.8464
11	0.9575	12	0.9285	13	0.9054	14	0.8861
12	1.0063	13	0.9730	14	0.9460	15	0.9237
14	1.0938	16	1.0904	16	1.0207	17	0.9929
16	1.1702	20	1.2171	20	1.1469	20	1.0841
20	1.2965	24	1.3192	24	1.2498	24	1.1871
24	1.3967	28	1.4030	28	1.3350	28	1.2732
						32	1.3463
			4 M HNO_3				
10	0.8745	11	0.8549	12	0.8390	13	0.8260
11	0.9267	12	0.9019	13	0.8820	14	0.8652
12	0.9749	13	0.9460	14	0.9223	15	0.9025
13	1.0201	14	0.9872	15	0.9604	16	0.9377
16	1.1383	16	1.0627	17	1.0304	20	1.0622
20	1.2650	20	1.1894	20	1.1222	24	1.1650
24	1.3661	24	1.2919	24	1.2252	28	1.2513
28	1.4486	28	1.3764	28	1.3110	32	1.3248
				32	1.3834		

For the operating conditions of 10 g of ^{233}U and 170 g of ^{232}Th per liter in feed tank S-2, the margin for safety is believed to be adequate because k_∞ is calculated to be less than 0.8. Although the same concentration would be subcritical without the thorium, it is recommended that

operations without thorium be limited to approximately 7.5 g of ^{233}U per liter, for which k_∞ is calculated to be 0.8. Higher concentrations may be allowed in instances where more stringent administrative procedures ensure that the critical concentrations will not be exceeded.

A.11 Pipe Storage Arrays X-11 and X-14

The calculational model for the pipe storage arrays can be described as follows:

1. The storage pipes were 4-in. sched. 40 stainless steel pipes, 20 ft long on 12-in. centers.
2. The concrete vault was 20 x 8 x 20 ft, with eight storage pipes centered on one of the 20- by 20-ft walls. For an interaction calculation, eight more storage pipes were added to the opposite wall.
3. The reflector albedo for 24-in. concrete was used instead of neutron tracking in the concrete of the vault walls, floor, and ceiling.

The k_{eff} values were calculated to be 0.679 ± 0.004 and 0.698 ± 0.004 for solutions containing 250 and 400 g of ^{233}U per liter, respectively, for filling the eight storage pipes. The KENO Monte Carlo code and Hansen-Roach 16-group cross sections were used to make these calculations. For a solution containing 250 g of ^{233}U per liter, the 16-pipe storage arrangement had a k_{eff} of 0.814 ± 0.004. The calculational model for the X-11, X-15, X-14, and X-17 tanks (the latter two were filled with 35% glass) assumed that these vessels were located in a concrete vault 20 x 8-3/4 20 ft and contained uranyl nitrate solutions having a ^{233}U concentration of 250 g/liter. The k_{eff} was 0.704 ± 0.004.

Other potential interaction problems were not found by examination of the layout of the tanks and process equipment in Building 3019 shown in Figs. 20-25.

A.12 Evaporator

The calculational model for the evaporator can be described as follows:

1. The evaporator was a 6-ft length of 5-in. sched. 40 stainless steel pipe.
2. The steam jacket was a 6-ft length of 6-in. sched. 40 stainless steel pipe.
3. The evaporator was located at the horizontal midplane and in the corner of a 16 x 9 x 18 ft concrete vault, 5 in. from one wall and 21 in. from the other. The evaporator and steam jacket were filled with an aqueous solution containing 400 g of ^{233}U per liter.
4. The reflector albedo from 24 in. of concrete was used instead of neutron tracking in the concrete of the vault walls, floor, and ceiling.

The k_{eff} was calculated to be 0.799 $\pm$ 0.005 using the KENO Monte Carlo code and Hansen-Roach 16-group cross sections.

A.13 UO_2 Preparation Facility

Only two equipment items in the oxide production line have the potential for accumulating masses of solid large enough to suggest that a calculation be made. They are described below.

Centrifuge Bowl. — The Teflon-lined centrifuge bowl has a capacity of 7.1 liters. It is not believed to be credible that the catch basin can become flooded with solution or with water. Therefore, the bowl is essentially unreflected. The unreflected spherical critical volume is greater than 13 liters at a ^{233}U concentration of 100 g/liter and is greater than 8 liters at a concentration of 500 g/liter. Thus, because the bowl has a sufficiently small volume and a nonspherical shape, subcriticality is ensured.

A double batch of precipitate (2 kg) in the centrifuge bowl is also subcritical by a large margin. However, there must be assurance that the

bowl and all replacement bowls are made from Teflon and not from hydrogen-containing plastics.

Oxide Blender. — The geometry of the blender volume, two perpendicular cylinders, is amenable to description by the Monte Carlo geometry routines. However, a calculation can be made in spherical geometry and the same mass in the blender geometry will have a lower k_{eff}. The critical radii for $^{233}UO_2$ at various densities (100, 50, 25, and 12.5% of theoretical) were calculated, reflected by 2 in. of lead. Both the calculated critical masses and the radii are plotted in Fig. A-8 as a function of oxide density. The proposed batch limit of 20 kg at an oxide density of 1.3 kg/liter is certainly subcritical, but administrative controls must be provided to preclude the presence of water on other forms of hydrogen.

A.14 References for Appendix A

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2. G. E. Whitesides and Nancy Cross, KENO, A Multigroup Monte Carlo Criticality Program, CTC-5, Computing Technology Center, Oak Ridge Gaseous Diffusion Plant (1969).
3. Gordon E. Hansen and William H. Roach, Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies, LAMS-2543, Los Alamos Scientific Laboratory (1961).
4. J. P. Nichols, C. L. Schuske, and D. W. Magnuson, Use of Borosilicate Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material, Y-CDC-8, Oak Ridge Y-12 Plant Criticality Data Center (1971).
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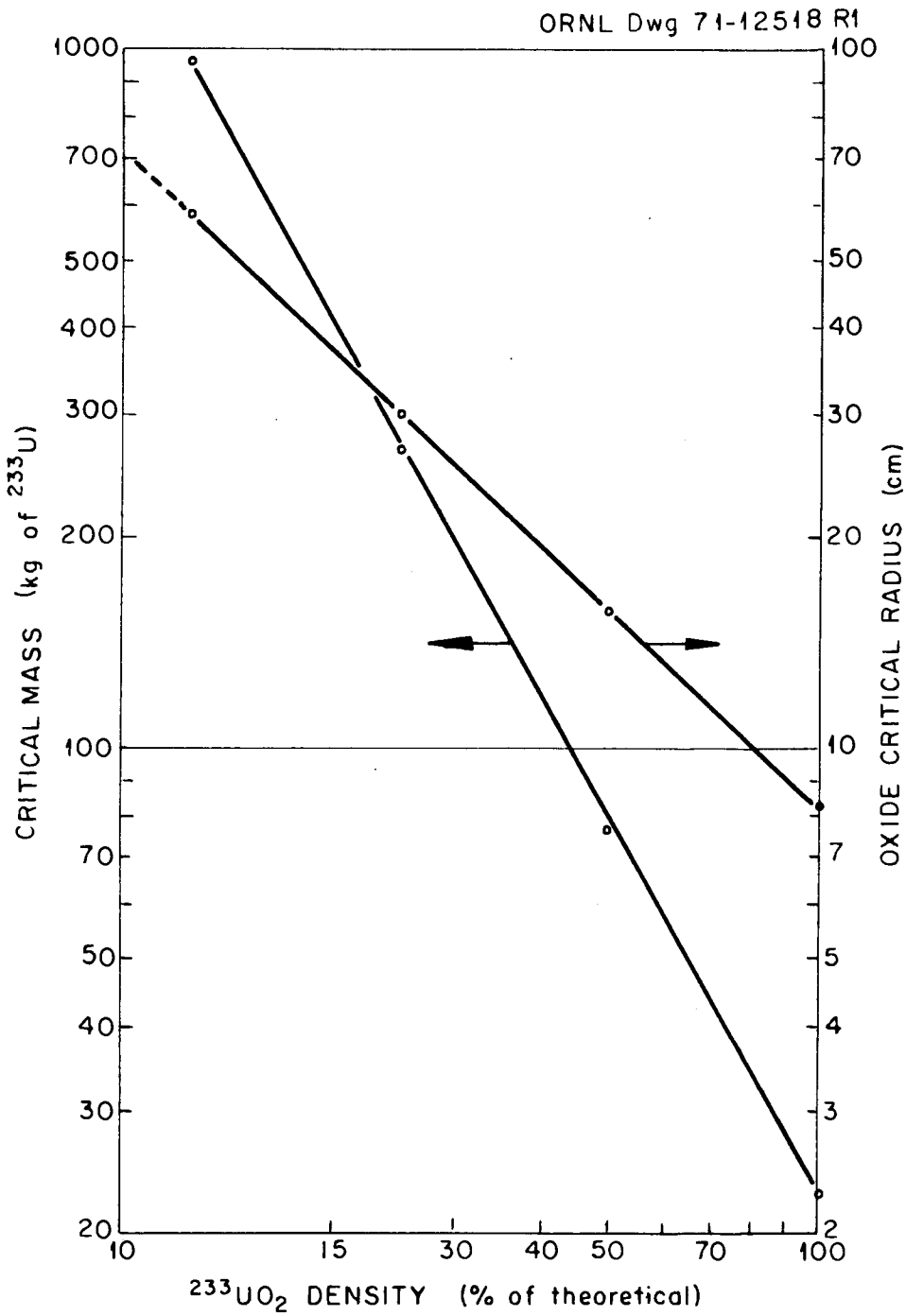

Fig. A-8. Critical Mass and Radius as a Function of Oxide Density, Reflected by 2 in. of Lead.

10.2 Appendix B: LWBR Support Training Program

The following subjects have been selected to cover the program effectively:

(1) General Program

a. LWBR Program
b. ORNL Participation
c. Organization

(2) Plant Safety

a. Plant Criticality
b. Conventional Safety and Emergencies
c. Routine Plant Surveillance
d. Building Containment
e. Health Physics

(3) Process Control

a. Analytical Requirements, Sample Preparation
b. Quality Assurance
c. Material Balances and Accountability
d. Run Sheets
e. Preparation of Plant Records

(4) Purification System Operation

a. Chemical Flowsheet
b. Equipment Flowsheet
c. Feed Material Handling, Charging
d. Storage Systems
e. Dissolution
f. Solvent Extraction System
g. Ion Exchange System
h. Solvent Wash System
i. Plant Evaporators
j. Unloading Solutions

(5) Oxide Conversion Line

a. Chemical Flowsheet
b. Equipment Flowsheet
c. Box A Operations - Liquid Waste Handling
d. Box B Operations - Precipitation and Centrifugation
e. Box C Operations - Microwave Oven
f. Box D Operations - Furnace Tray Loading
g. Box E Operations - Furnace
h. Box F Operations - Furnace Tray Unloading
i. Box G Operations - Powder Inventory and Packaging
j. Box L Operations - Isotopic Dilution
k. Product Shipping
l. Standard Practices for Glove-Box Operation

(6) Control of Plant Effluents

a. Cell Off-Gas System
b. Vessel Off-Gas Systems
c. Glove-Box Ventilation
d. Miscellaneous Waste Systems
e. Low and Intermediate-Level Liquid Wastes
f. Solid Waste Handling

(7) Plant Support

a. Analytical Chemistry
b. Services Including Waste, Air, Water, Steam, etc.
c. Maintenance

10.3 Appendix C: Operational Procedures Used in the LWBR Support Program

All processes in the LWBR support program in the Building 3019 Pilot Plant will be operated according to the written, tested procedures listed below. Each procedure progresses according to stepwise instructions given in detail on a run sheet that provides for supervisory acknowledgments (by signature) of critical steps and items.

Procedure 3.1 (Pure Oxide Dissolution) is included following the procedure list. It is typical of the operating procedures used in the LWBR assistance program.

RPPP Procedures

1.0 Receiving and Storing ^{233}U

- 1.1 Solution Receiving and Storing
- 1.2 Solids Receiving and Storing

2.0 Stored Concentrated Solution Transfer

- 2.1 Removal for Shipping
- 2.2 Transfer Between Storage Tanks
  - 2.2.1 Between Glass Ring-Filled Tanks
  - 2.2.2 From U-1 - U-10 to S-2
  - 2.2.3 From R-2 to S-2

3.0 Dissolution of Solids

- 3.1 Pure U-Oxide Dissolution
- 3.2 Scrap Dissolution

4.0 U Purification

- 4.1 Solvent Extraction Operations
- 4.2 S-2 Boildown
- 4.3 Preparation of DIBAN
- 4.4 Concentration of Product
- 4.5 Preparation and Cleanup of Organic Phase
- 4.6 Isotopic Dilution
- 4.7 Cleanup of Spills
- 4.8 Sampling

5.0 Thorium Removal

- 5.1 Ion Exchange Operations
- 5.2 Acetate Cleaning of Resin

6.0 ^{233}U Oxide Production

- 6.1 Standby Conditions (all systems)
- 6.2 Feed Preparation

6.3 Precipitation and Centrifugation

6.4 Drying

6.5 Boat Loading

6.6 Firing

6.7 Stabilizing

6.8 Blending

6.9 Product Solids Packaging for Shipment

6.10 Waste Handling

- 6.10.1 Solution
- 6.10.2 Solids

6.11 Weekend Cleanup; Spills

6.12 Furnace Operation

- 6.12.1 Start-up
- 6.12.2 Normal Operation Checksheet
- 6.12.3 Shutdown

6.13 Gas Handling

- 6.13.1 H_2 System
- 6.13.2 Argon System
- 6.13.3 NH_3 System

Approved:
Operations Group Ralph G. Nicol 1
Technical Group W. J. McWaffee
Project Leader J. Rausch

Rev. Date 12-1-71

Date: _____

Run No. BETO-_____

Dissolution No. _____

3.1 Pure U-Oxide Dissolution Run Sheet

3.1.1 Preliminary Data:

(1) Criticality specs: Max. ^{233}U/charge = 5500 g., max. ^{233}U conc. = 400 g/liter in S-15.

(2) Uranium content of 1.9-liter heel in S-15 _____g. (See previous run.)

(3) Description of material to be added to S-15:

Source: Transfer No. _____; Material Identity _____

Well No. or Drum No.	Can No.	U (g)	^{233}U (%)	^{232}U (ppm)

Total weight U _____ (max. - 4500 g U if no heel; 4000 g U if heel in S-15).

Total weight of U to be in S-15 (2+3) _____gm

(4) Minimum solution volume = 13 liters for efficient thermocycling.

(5) Expected U concentration _____ g/ℓ.

XXX Supervisory review (comparison with criticality and solubility limits)

by _____, Date _____

Technician _____, Date _____

2

Rev. Date 12-1-71

Date: _____

Run No. BETO-_____

Dissolution No. _____

3.1.2 Check Dissolver System:

(1) LR-S15 _____, DR-S15 _____, PR-S15 _____

TR-S15 _____, LR-R35_____, DR-R35 _____

Check calibration/purge flow of pneumatic instruments _____.

XXX Supervisory approval to continue, by _____.

(2) Check that R-35 contains enough water or dilute UN solution so accidental transfer of UN from S-15 will produce a solution containing no more than 250 g U/liter in R-35 (see calculation, p. 9).

Criticality limit: 300 g U/liter in R-35; 250 g U/liter in R-25.

(If valves leak, solution from S-15 could collect in R-25.)

a) Vol. solution in R35 _____, wt. U in R35 _____,

b) Sol. solution in S15 _____, wt. U in S15 _____.

c) Total weight of U that could collect in R35 _____.

Remember, some U remains in 1.9 liter heel in S-15.

d) Volume of solution required in R35 so max. conc. U is 300 g/liter _____.

e) Determine water to be added to R-35 _____ (see calculation, p. 9).

XXX Supervisory approval to continue, by _____.

f) Add water to R-35 through charging pot in makeup room, liters _____.

Technician _____, Date _____

Rev. Date 12-1-71

Date: _____

Run No. BETO-_____

Dissolution No. _____

g) Record:

LR-R35 _____, DR-R35 _____, Vol. R35_____.

XXX Supervisory approval to continue, by _____.

3.1.3 Transfer shipping drum or can (see p.1) from _____ to charging area in penthouse _____. Have HP coverage _____.

3.1.4 Determine resistivity of demineralized water _____ (must exceed 3.5 x 10^5 ohms).

3.1.5 Add dissolvent to S-15 (see calculation, p.10).

(1) 15.8 M HNO_3 (ACS spec.) _____ liters,* Demin. H_2O _____ liters. (Total = 11.1 liters).

(2) Record LR-S15 _____, TR-S15 _____ DR-S15 _____ (constant readings required).

(3) Turn on air to draft tube (5 psig) _____.

(4) Heat solution to 50°C _____.

XXX Supervisory approval to continue, by _____

3.1.6 Bag material into Charging Glove Box one can at a time (not more than 1 kg of U/can). Glove box criticality limit: 1 can of U-oxide, no U-bearing solutions.

(1) Can No. _____ _____ _____ _____ _____

_____ _____ _____ _____ _____

(2) Have HP coverage _____.

(3) Place can in shielded carrier _____. Set carrier at glove box charging port _____. Place can into glove box. Weigh: Fill out weigh-in sheet.

*To maintain purity of product, break manufacturer's seal on bottle at time of addition. Dispose of any unused acid so it cannot be used in the next dissolution.

Technician _____, Date _____

Rev. Date 12-1-71

Date: _____

Run No. BETO-_____

Dissolution No. _____

Open can _____, examine material and can _____.*

(4) Sample (if necessary) Code: MOS-_____.

Request: Coulometric U, isotopic U, ^{232}U, mass spec. for impurities.

Oxide Weighing-in Sheet

Can No.	ORNL weight, grams Gross	Tare	Net	Vendor's net wt., grams	Remarks

Note: Maximum net weight of oxide per dissolution: 5500 g, if no U-bearing heel in S-15 from previous run. 5000 g, if U-bearing heel in S-15 from previous run.

3.1.7 Prepare to Charge Oxide.

(1) Charging chute for S-20 must be covered with a plate locked in position and the key must be stored outside of glove box _____.
Criticality spec: S-20 is not critically safe for dissolution of U-oxide.

XXX Supervisory approval to continue, by _____, Date _____

*Notify supervisor of condition of material. Record findings under "Remarks"

Technician _____, Date _____

Rev. Date 12-1-71

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Date: _____

Run No. BETO-_____

Dissolution No. _____

(2) Open top isolation (ball) valve (to S-15) _____.

(3) Inspect charging chute above bottom isolation valve for foreign materials _____.

(4) Tilt holder and pour powdered oxide into charging chute _____; reinspect charging chute _____.

(5) Close top isolation valve _____.

(6) "Cutie Pie" reading at isolation valves _____.

(7) Open bottom isolation (ball) valve _____.

(8) "Cutie Pie" reading at isolation valves _____.

(9) Observe action of S-15 instruments: changes, are they normal? _____.

(10) Close bottom isolation valve _____.

(11) Weigh empty can _____.

(12) Repeat steps 3.1.6 and 3.1.7 until charging is finished _____.

(13) Recheck can numbers and weights _____.

(14) Close top isolation valve _____.

(15) Bag out empty containers and clean up glove box _____.

XXX Supervisory approval to continue, by _____.

3.1.8 Dissolve material.

(1) Check that LR-S15 _____, DR-S15 _____, PR-S15 _____, and TR-S15 _____ are normal. (Record final readings following charging, above.)

Technician _____, Date _____

Rev. Date 12-1-71

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Date: _____

Run No. BETO-_____

Dissolution No. _____

(2) Heat S-15 to 70° - 80°C _____.

(3) Record information hourly on Data Sheet during dissolution operation (attached).

(4) Maintain solution volume at 13 to 14 liters _____.

(5) When dissolution is complete (solution density constant for 2 hours), close steam supply valve and cooling water valve _____.

(6) Sample solution in S-15 _____; Code MDO-_____.
Request H^+ (∿0.1 M) _____ and Sp. g. (∿1.35) _____.

(7) Calculate amount of dissolved U _____. (Formula, p. 10 or curve).

(8) Recheck LR-R35 _____ and PR-R35 _____.

XXX Supervisory approval to continue, (a) dissolution or (b) transfer, by _____.(circle approved plan.)

3.1.9 Transfer S-15 product to R-35.

(1) Open valve, TOG-152 (KL) _____.

(2) Check change in LR-S15 _____ and LR-R35 _____.

(3) When transfer complete, close valve, TOG-152 (KL) _____.

(4) Sparge solution in R-35 for 30 min. _____.

3.1.10 Sample Solution in R-35. Code MDOA-_____.
Request: U (coulometric), H^+, sp. gr. Record: LR-R35_____,
DR-R35 _____, Vol. R-35 _____.

Technician _____, Date _____

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Rev. Date 12-1-71

Date: _____

Run No. BETO-_____

Dissolution No. _____

Uranium Accounted for _____ gm. (Note: Max. U conc. = 250 g/liter to go to U-storage tanks, criticality limit.)

XXX Supervisory approval to continue to Part 3.1.11 _____ or 3.1.12_____,

by _____ (circle one.)

3.1.11 Transfer R-35 product to U-storage.

Receiving tank to be specified by supervisor _____.

(1) Open TOG-151 (KL) _____.

(2) Open vent valve (TOG-R26V) for R-26 _____.

(3) Check that drain valve for R-25 (TOG-R25/R26) is closed _____.

(4) Open TOG-103 _____ until LR-R35 decreases and again becomes constant _____.

(5) Close TOG-103.

(6) Open TOG-R26M-1/2 and 1 and 2 _____.

(7) Pressurize R-26 (TOG-R26A) _____ with TOG-R26V closed _____ until U-20 liquid level alarm sounds, indicating completion of transfer.

(8) Open vent valve and close air supply valve to R-26 _____.

(9) Close TOG-R26M-1/2 _____.

(10) Drain U-20 to U storage tank (See UN Transfer Run Sheet 2.1) _____.

(11) Repeat steps 4 through 10 until R-35 is empty.

3.1.12 If analysis shows less than desired amount of U in solution (<95%):

(1) Resparge R-35 solution _____.

Resample _____: Code MDOA-_____.

Technician _____, Date _____

Rev. Date 12-1-71

Date: _____

Run No.BETO- _____

Dissolution No. _____

(2) If analytical results satisfactory, proceed to part 3.1.11 _____.

(3) If analytical results low, proceed to part 3.1.13 _____.

3.1.13 Cleanout Dissolving of Oxide in S-15.

(1) Transfer solution in R-35 to U storage (part 3.1.11) _____.

(2) Check that valve TOG-152 (KL) is closed _____.

(3) Charge 9 liters of water and 4 liters of 15.8 M HNO_3 to S-15 _____.

(4) Heat S-15 to 70°-80°C with sparging.

(5) Continue for 2 hours; sample _____, code MDOC-_____.

(6) Determine U content from H^+ and sp. gr. (formula, p.10 or curve)

_____ g.

(7) (a) If missing U is accounted for, cool to 50°C, drain to R-35, and resample (parts 3.1.9 and 3.1.10) _____.

(b) If missing U is not found, repeat clean-out dissolving _____.

(8) This product containing a high acid concentration must be transferred to the solvent extraction system (S-2).

(a) Transfer solution from S-15 to R-35, as above _____.

(b) Transfer from R-35 to U-20, as above _____.

(c) Transfer from U-20 to S-2.*

*See Run Sheet No. 2.2.2. S-2 must contain enough $Th(NO_3)_4$ solution so final wt. ratio Th/U = 17/1 after transfer (criticality spec.).

Technician _____, Date _____

Rev. Date 12-1-71

Date: _____

Run No. BETO-_____

Dissolution No. _____

Sample Calculations - Material to be charged is UO_3 from NFS and SR in 1970.

Maximum charge to S-15 = 4.5 kg U (criticality spec: 5.5 kg U, max.)

Maximum U concentration: S-15 = 400 g/ℓ; R-35 = 300 g/ℓ; R-25 = 250 g/ℓ

(criticality specs.)

Minimum volume of solution = 13.0 liters

1. Water heel in R-35 for first dissolution in S-15 (no heel in S-15).

   Solution draining to R-35 = 13.0 - 1.9 (S-15 heel) = 11.1 liters

   U conc. in S-15 = $\frac{7 \text{ cans x } 619 \text{ g U/can}}{13.0 \text{ liters}} = \frac{4333 \text{ g}}{13.0 \ \ell} = 333$ g/liter

   U draining to R-35 = 333 x 11.1 = 3696 g

   Initial heel in R-35 (for U conc. of 250 g/ℓ) =

   $\frac{3696}{250} - 11.1 = 14.8 - 11.1 = 3.7 \ \ell$

2. U in S-15 for second and succeeding dissolutions

   U in heel of S-15 (from 1st run) = 1.9 ℓ x 333 g/ℓ =633 g

   U in new charge to S-15 = 6 cans x 619 g U/can = 3714 g

   Total charge in S-15 (U) = 4347 g

   U conc. in S-15 = $\frac{4347 \text{ g}}{13.0 \ \ell} = 334$ g/ℓ

   R-35 heel remains as shown in 1, above.

Technician _____, Date _____

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Rev. Date 12-1-71

Date: _____

Run No. BETO-_____

Dissolution No. _____

3. Dissolvent solution required in S-15: 11.1 liter

Heel: 1.9 liter

15.8 M HNO_3:

	No Heel	Heel
moles to be dissolved = $\frac{4333}{233}$ = 18.6		$\frac{3714}{233}$ = 15.9
moles acid required 2 x 18.6 = 37.2		31.8
moles acid (excess) = 0.2 M x 13/ℓ = 2.6		.2 x 11.1 = 2.2
total moles acid	39.8	34.0
volume of acid = $\frac{39.8 \text{ moles}}{15.8 \text{ M}}$ = 2.5 liter		$\frac{34.0}{15.8}$ = 2.2 liter
Water: 13 - (2.5) = 10.5 liter		11.1 - 2.2 = 8.9 liter
Total solution volume = 13.0 liter		13.0 liter

4. Formula for Determining U-sp. gr. and acid molarity given.

Sp. Gr. = 1.001 ± 0.318 (M U) + 0.031 (M HNO_3), when Th concentration nil.

Technician _____, Date _____

Date: _____

Dissol. # _____

Run # _____

Dissolution and Clean-Out Data Sheet (S-15)

Time	LR %	DR %	p %	T°C	By

10.4 Appendix D: Emergency Procedure

Shown on the next few pages is a copy of the "Building 3019 Emergency Procedure" issued each individual assigned to the Building 3019 Pilot Plant.

10.4.1 Definition of an Emergency

An emergency is defined as the occurrence of an unplanned event or combination of events requiring immediate evacuation of an area of building by the occupants.

10.4.2 Responsibility of Employees

One of the most critical factors in combating emergencies is the ability of the person discovering an emergency to act immediately, with knowledge of what action to take and where to obtain assistance rapidly. Each person normally assigned to Building 3019 will be thoroughly familiar with the following:

(1) Location and use of all emergency equipment.

(2) Assembly points for all types of emergencies.

(3) Emergency alarms.

(4) Name of Local Emergency Director.

(5) Emergency equipment shutdown procedures.

(6) Route of evacuation.

(7) Assembly areas to which all Building occupants will move to receive instructions.

10.4.3 Responsibilities of Person Discovering and Declaring Emergency

The person discovering and recognizing an emergency will:

(1) Report immediately to the Local Emergency Director.

(2) Obtain assistance.

(3) Seek advice.

10.4.4 Responsibility of Local Emergency Personnel

A local emergency squad consists of the Building Supervisor, his delegated alternate, wardens, and other squad members. The responsibility of each is outlined below.

Building Supervisor. — The Building Supervisor shall organize and train building personnel with regard to the action required for various types of emergencies, ensure that local plans are consistent with overall laboratory plans, and maintain emergency equipment and evacuation plans.

In the event of a local emergency, he will direct his squad and ensure that:

(1) Building personnel have been evacuated.

(2) The Laboratory Emergency Control Center has been notified.

(3) Emergency service units are assembled, briefed on the situation, and directed to the scene.

(4) Equipment and processes are shut down as necessary for safety.

(5) The safety of the assembly point is considered.

(6) The extent of the emergency is determined.

(7) Pre-entry into the area and building is controlled.

(8) Personnel and area monitoring for radioactivity by Health Physics and Safety Group is begun.

Wardens. — Wardens are designated, trained Building personnel responsible for ensuring that all areas of the building have been evacuated in the event of an emergency.

10.4.5 Emergency Signals

The following signals indicate the existence of an emergency:

(1) Building sirens.

(2) Building air horns.

(3) Public address system.

(4) Flashing lights.

10.4.6 Laboratory Evacuation Signal

The Laboratory evacuation signal is a siren-like wail over the ORNL public address system. All Building 3019 personnel will leave the building following established evacuation routes (indicated by signs in corridors and at doors) and assemble in the area at the northwest corner of Building 3019. Further instructions will be given on the public address system.

10.4.7 Emergency Equipment

The Local Emergency Director will inventory and maintain an emergency supply cabinet at the local assembly point.

10.4.8 Evacuation Routes

The evacuation routes are marked by signs in strategic locations throughout the building.

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