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SUBJECT: Hazards Evaluation of Building 2527 and Very High Activity Waste Storage Facility - Vol. 7

TO: F. L. Culler

FROM: A. R. Irvine and J. P. Nichols

ABSTRACT

An evaluation of the hazards in Bldg. 2527 and the very high-activity waste storage facility indicates that the accident with the most serious consequences, a nuclear excursion which ruptures a vessel that contains the most radioactive process solution, would result in a probable radiation exposure to personnel in the secondary containment shell of approximately 0.1 rem and a maximum downwind exposure to ORNL personnel of approximately 1 rem. The fallout of radioactive particulate matter would be such that neither the hazardous ground concentration nor the required decontamination level would be exceeded.

Standards of construction and containment, assumptions made to evaluate the potential hazards of release of radioactive material, and methods of calculation used for development of this hazards analysis are given in ORNL-2956, Summary Report - Hazards Analyses of Radiochemical Processing and Waste Disposal at Oak Ridge National Laboratory, Sects. 4.0, 5.0, and 6.0.

NOTICE

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Part I. BUILDING 2527, FUEL STORAGE AND DISSOLUTION FACILITY

1.0 INTRODUCTION

1.1 Purpose and Uses

Building 2527 is a proposed extension of the ORNL radiochemical pilot plants and is for receipt, storage, and dissolution of power reactor fuels. It will be used in conjunction with the existing solvent extraction facilities in Bldgs. 3019 and 3505 to demonstrate processes for power reactor fuel recovery and recycle by solvent extraction. All types of radioactivity from long-decayed reactor fuel will be present at various points in the system. Radioactivity in the fuel elements as received will consist primarily of solid fission products and plutonium; after dissolution, gaseous fission products will be released and the remaining activity will be in liquid form.

1.2 Location and Distance from Other Facilities

Building 2527 will be located west of Third Street, between Central Avenue and Burial Ground Road (Fig. 1-1). Adjacent facilities include four buildings containing no activity and three containing varying amounts of activity, plus a proposed very high level waste storage facility. The distances of the various buildings from Bldg. 2527 are:

Bldg. No.	Name	Distance ft	Distance Direction	Activity Inventory, curies
2527	(Canal)			10^8-10^9 (mixed fp's in solids)
	(Cells)			10^6 (mixed fp's in solution)
	Waste Storage	25	East	3×10^8 (mixed fp's in solution)
2528	Thorium Oxide Preparation Plant	175	South	Negligible (α, in solids)
3505	Metal Recovery Plant	200	East	$<10^3$ (Zr, Nb, Ru, in solution)
3517	Multicurie Fission Product Plant	300	Southeast	10^6 (mixed fp's in solution)
2506	Stores, Paymaster	100	North	--
2525	Shop	100	Northwest	--
2516	Warehouse	100	West	--
2519	Central Steam Plant	200	Southwest	--

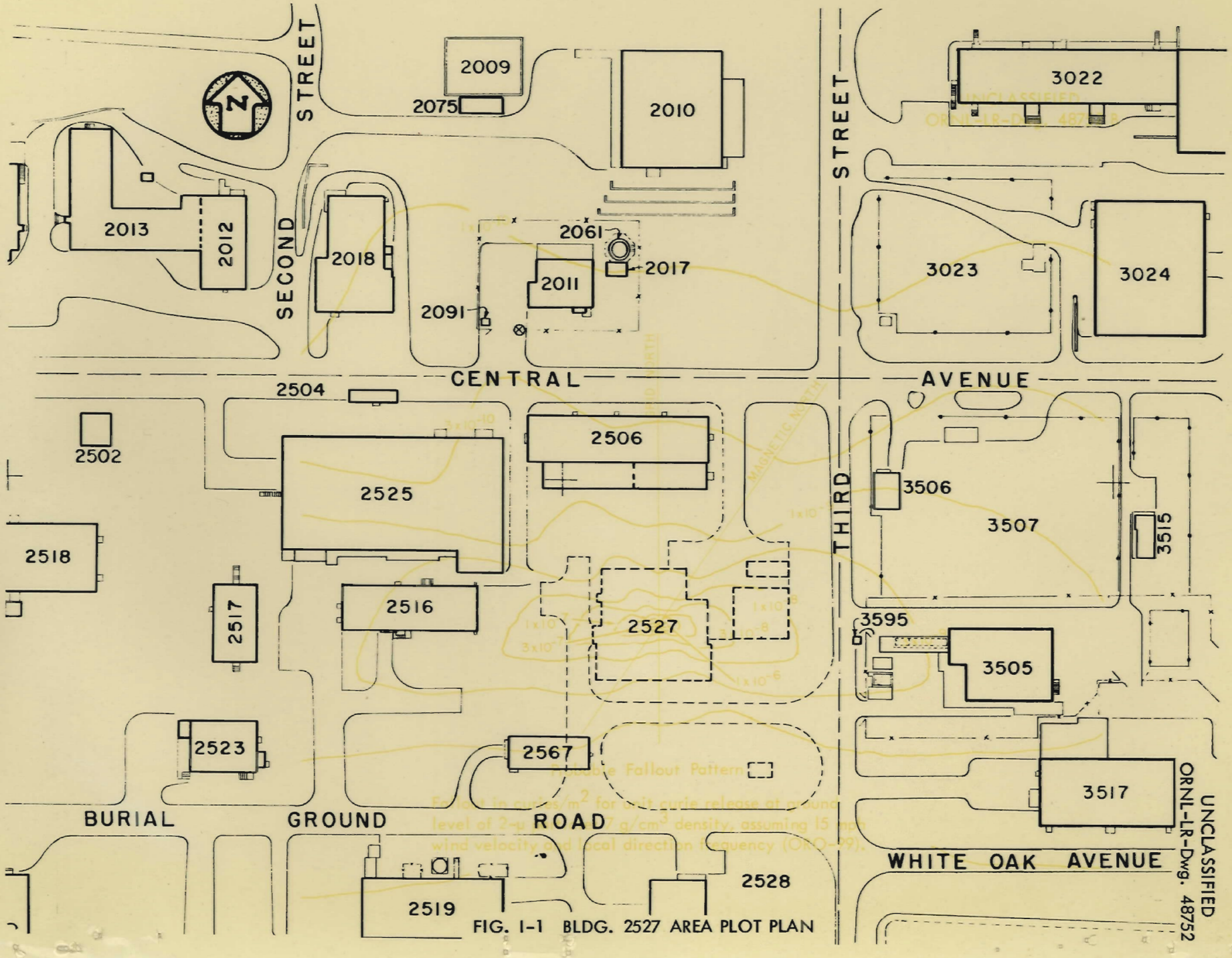

FIG. 1-1 BLDG. 2527 AREA PLOT PLAN

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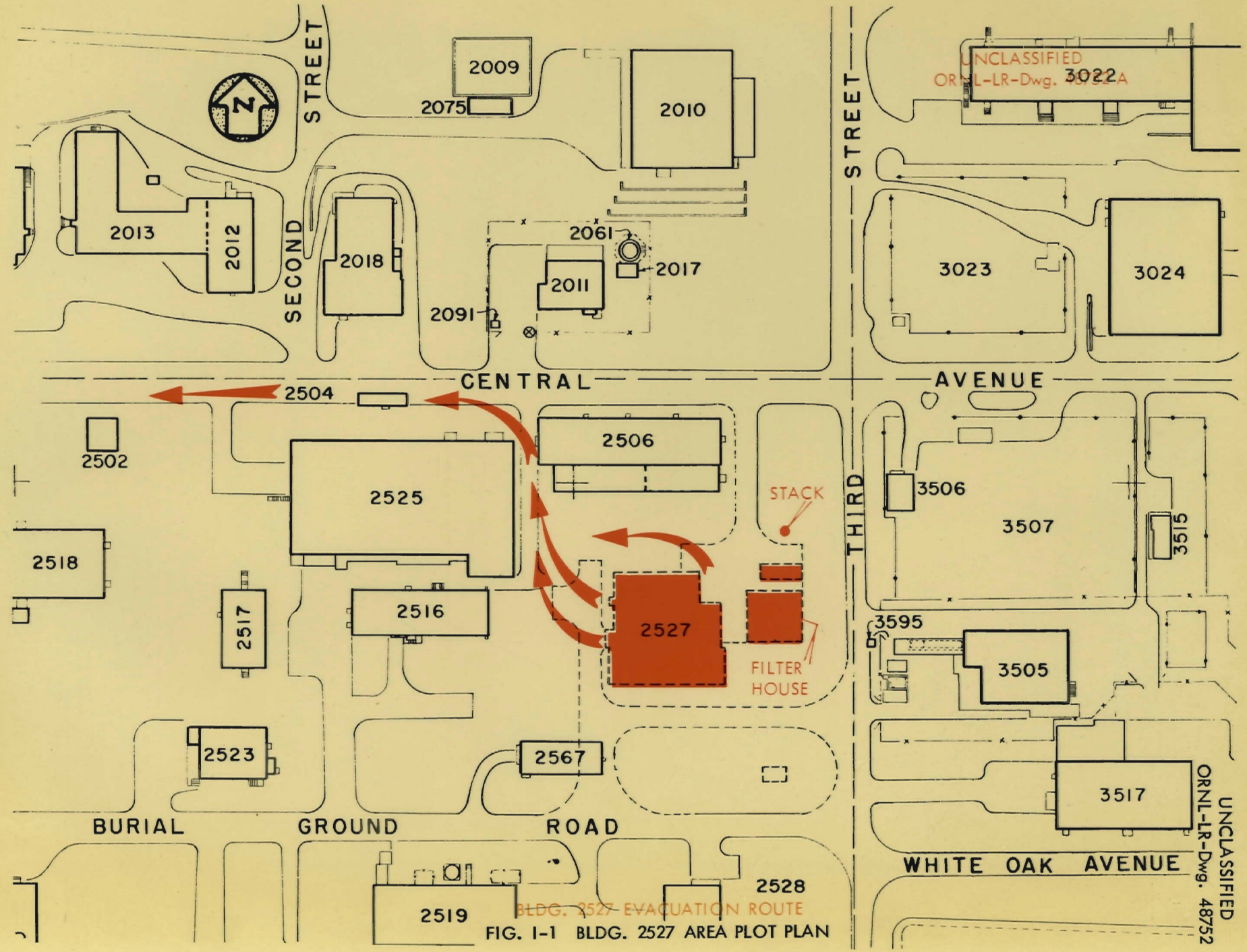

FIG. 1-1 BLDG. 2527 AREA PLOT PLAN

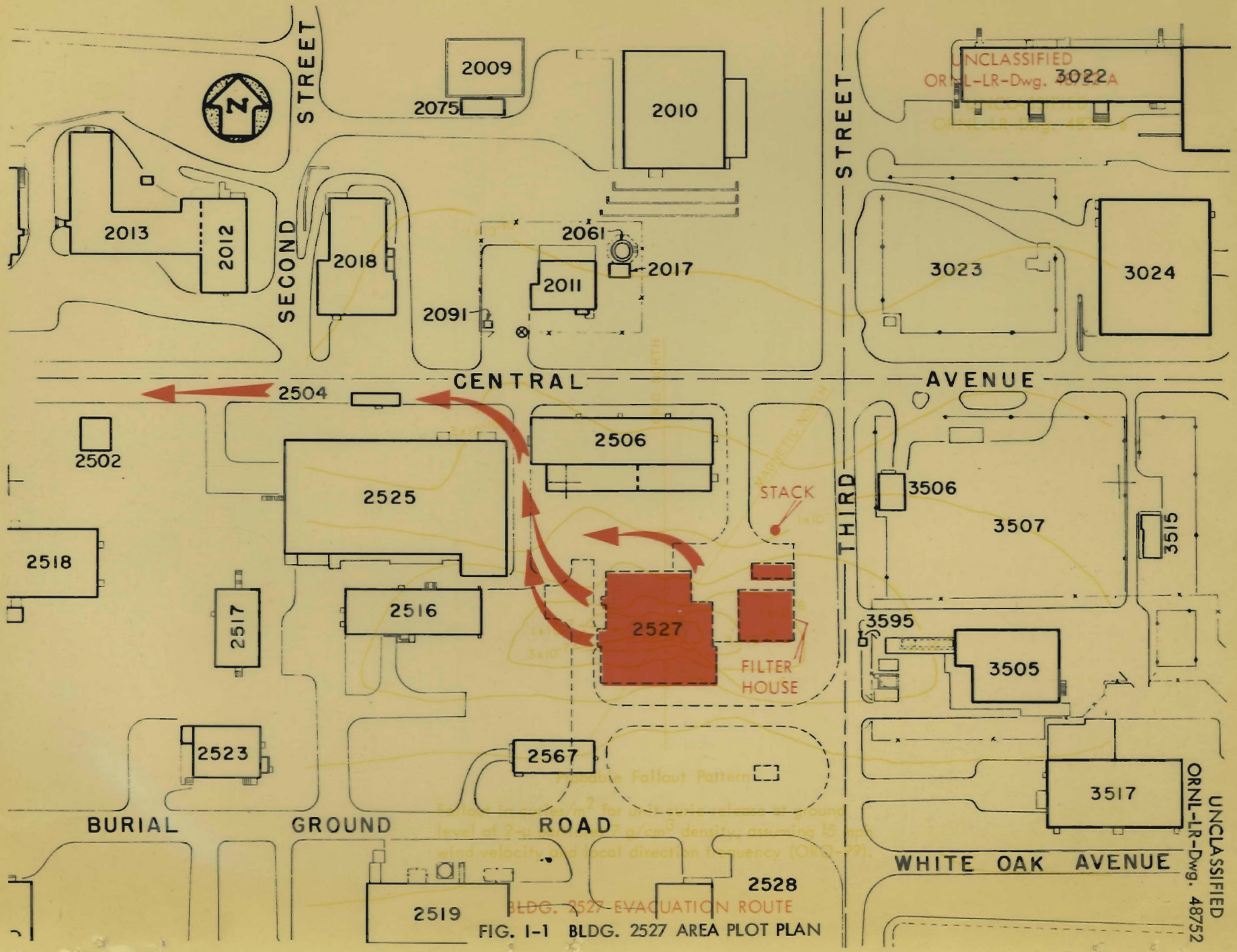

FIG. 1-1 BLDG. 2527 AREA PLOT PLAN

1.3 Building Description

The facility will consist of a deep canal for receipt and storage of fuel assemblies; two 6-ft-thick concrete-shielded cells; an area for cell auxiliaries; a vessel off-gas treatment system; and cell off-gas treatment and disposal equipment including a 100-ft stack. Figures 1-2, 3, 4, and 5 are plan and sectional views of the building with lines of containment and areas of possible contamination indicated on Fig. 1-2. The primary containment shell is to be the massive concrete wall in the case of the cells; in the case of the canal, heavy concrete walls on the sides and 12 ft of water above the fuel elements constitute the primary containment shell. Cell walls will be as leaktight as possible and blast-resistant. Secondary containment is provided by a leak-resistant building operated (in case of emergency) under negative pressure and having a filtered exhaust system.

1.4 Personnel Control

The numbers of people normally occupying Bldg. 2527 and adjacent facilities are:

	No. of People	
Bldg. No.	Weekdays	Nights and Weekends, per shift
2527	18	4
Waste Storage System	0	0
2506	28	0
2516	0	0
2519	7	4
2525	75	0
2528	3	1
3505	7	3
3517	12*	6

Personnel working in potentially contaminated areas of Bldg. 2527 normally will enter and leave the facility only by way of the change room, at which point they will be checked for radioactive contamination. Access to the equipment pit area will be required only for periodic maintenance; monitoring will be provided at the exit from this area as required.

*There may be up to 50 additional men during periods of construction.

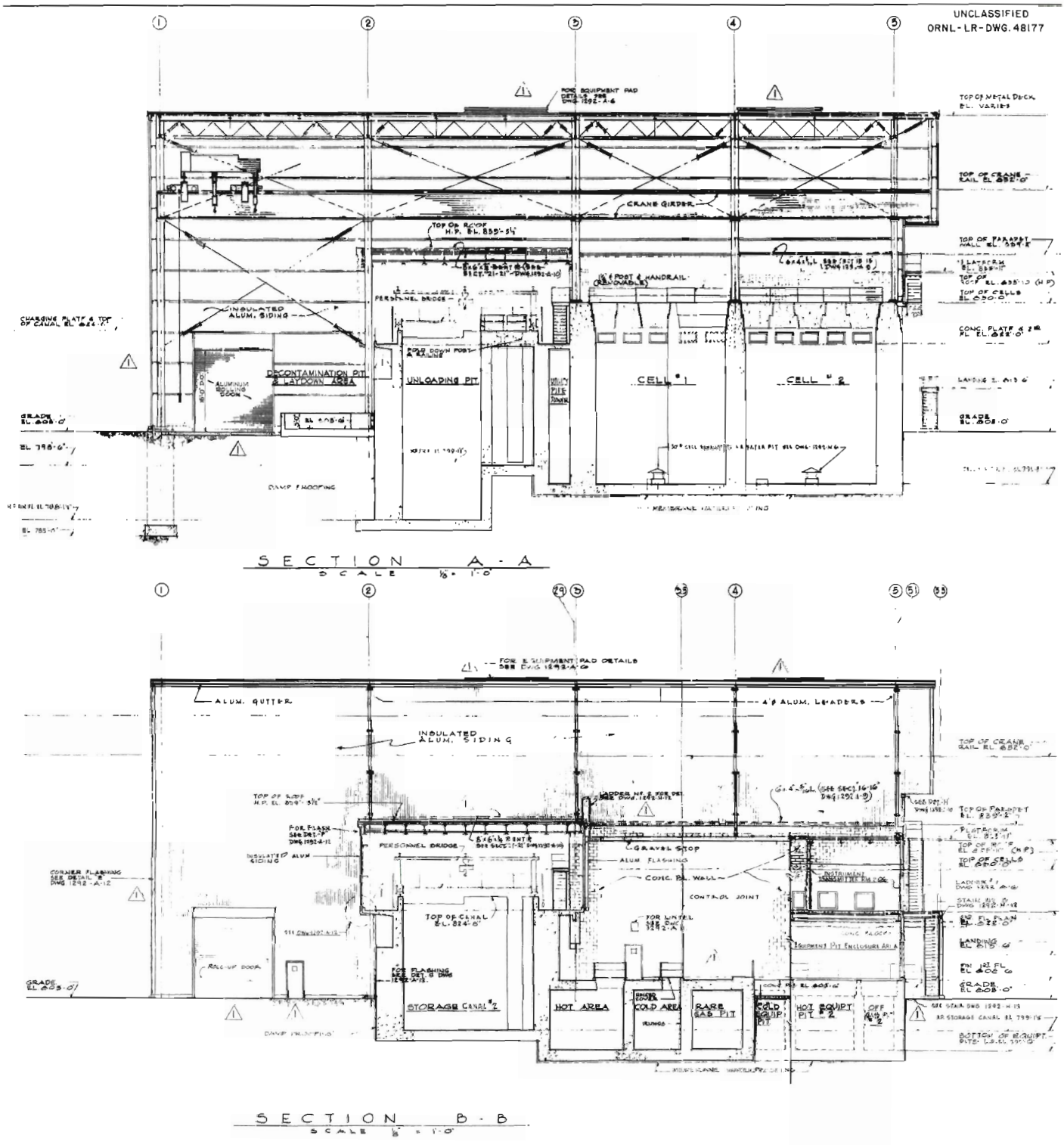

FIG. I-4 BLDG. 2527 - CROSS SECTIONS "A"-"A" & "B"-B"

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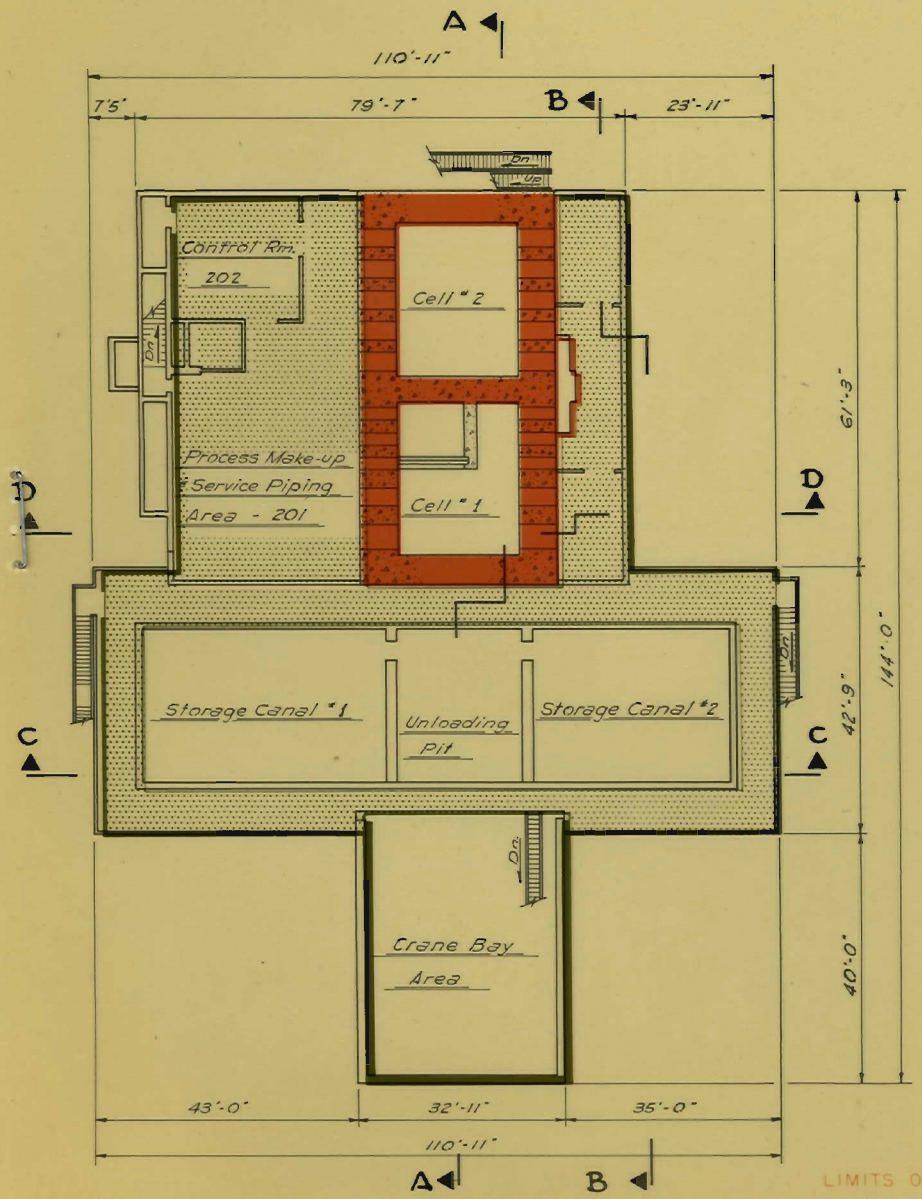

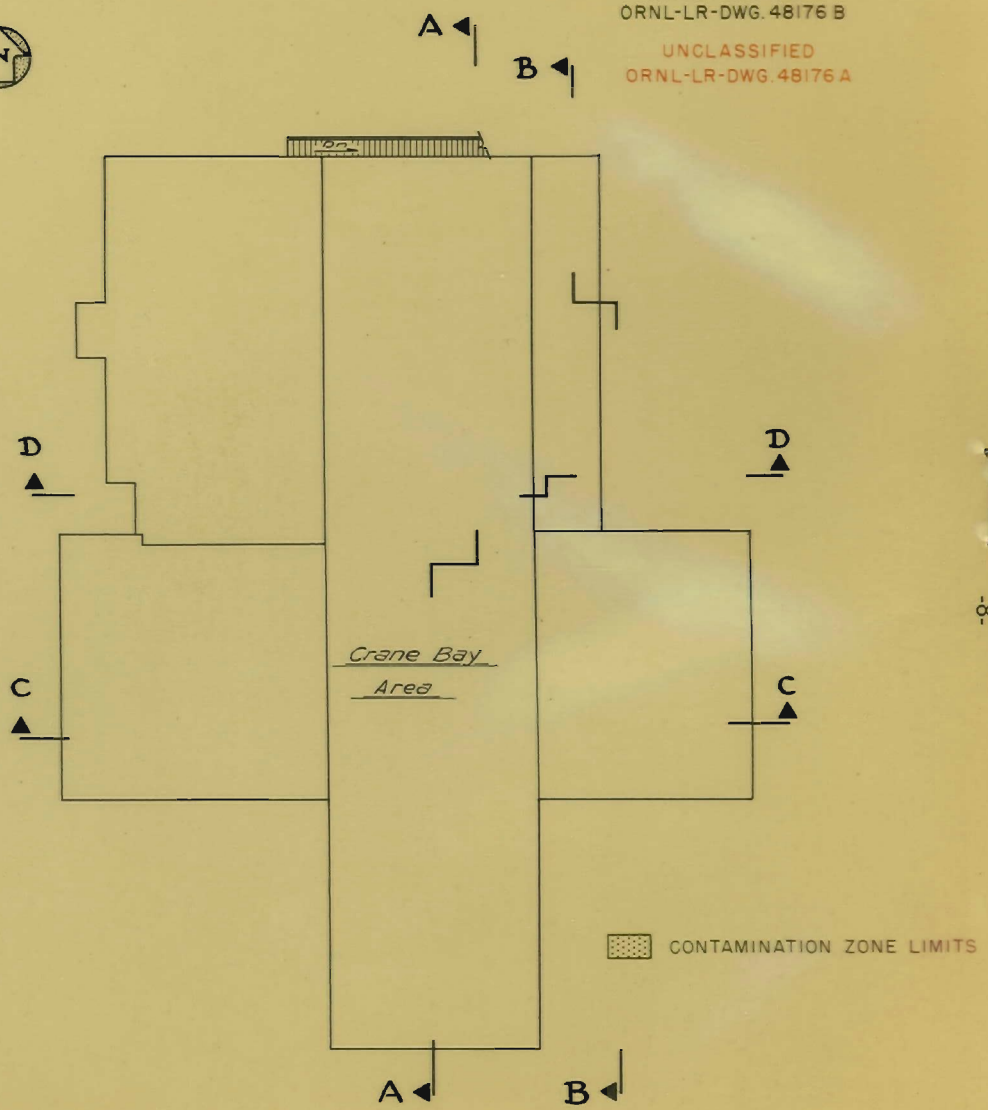

LIMITS OF PRIMARY CONTAINMENT
LIMITS OF SECONDARY CONTAINMENT

FIG. I-3 BLDG. 2527

SECOND FLOOR PLAN – ELEV. 822'-0"

ROOF PLAN

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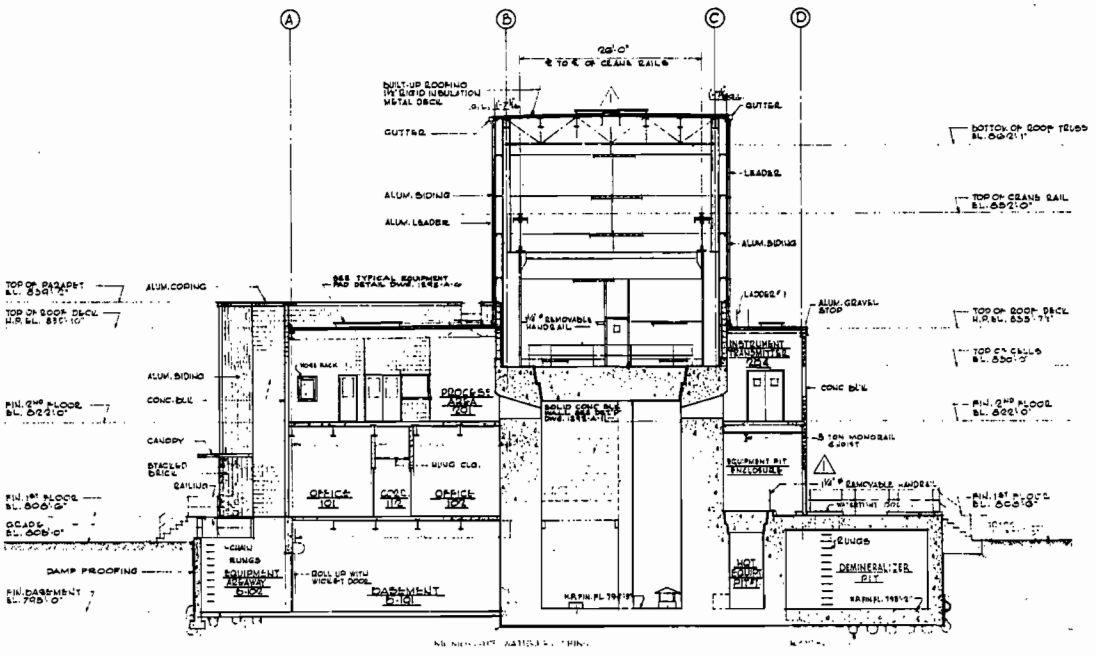
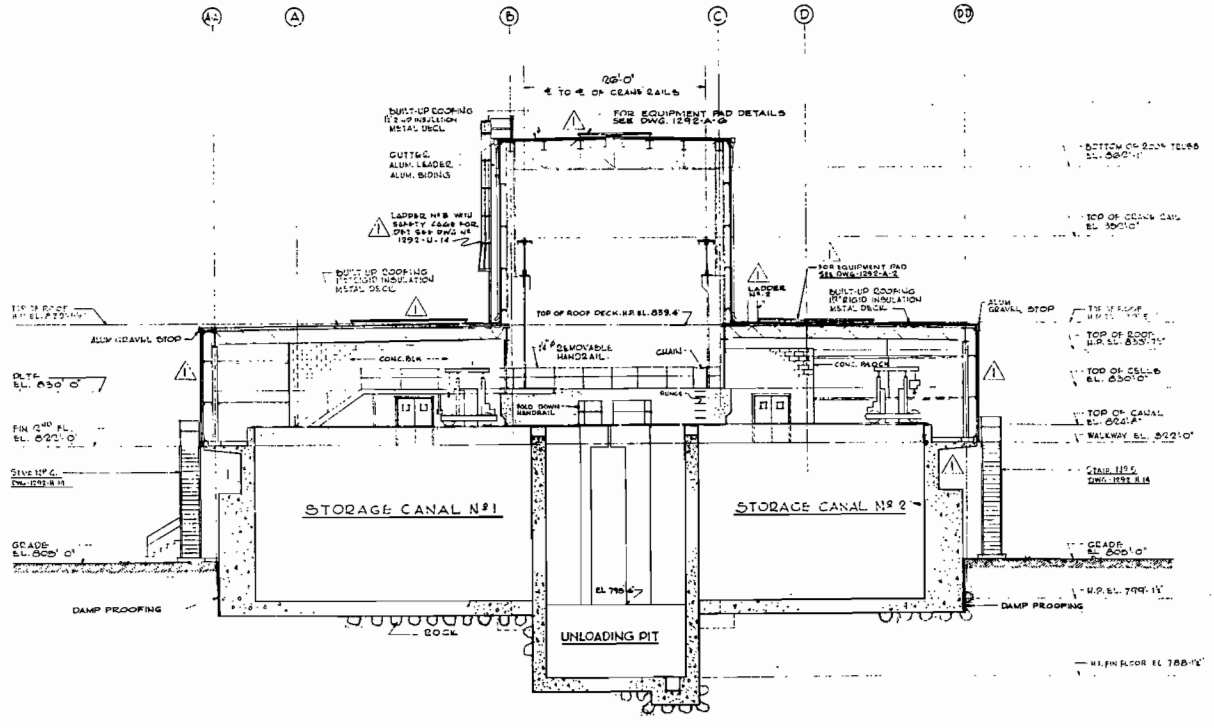


SECTION "D-D"
SCALE: 1/8"=1'-0"



SECTION "C-C"
SCALE: 1/8"=1'-0"

FIG. I-5 BLDG. 2527 - CROSS SECTIONS "C'"-"C" & "D'"-"D"

1.5 Process Description

The processes under consideration for demonstration in Bldg. 2527 are Darex, Sulfex, and Zirflex. Darex as applied to uranium metal and UO_2 fuels utilizes dilute aqua regia contained in a titanium system for dissolution of stainless steel cladding and core material. Residual chloride is subsequently removed by boiling in the presence of concentrated nitric acid. Darex (as applied to stainless steel—clad ThO_2-UO_2), Sulfex, and Zirflex are selective dissolution processes utilizing dilute aqua regia, sulfuric acid, and ammonium fluoride, respectively, to dissolve the cladding material without appreciably attacking the core material. The first two reagents are used to dissolve stainless steel, while the latter is used for Zircaloy cladding. A schematic equipment flowsheet is shown in Fig. 1-6.

1.6 Criticality

Nuclear safety (criticality) is a vital concern in this plant due to enrichment (up to 20%) and the quantity of fuel to be processed per batch (up to 22 kg of U^{235}-U^{233} mixture in 228 kg of thorium). Nuclear materials will always be situated behind a thick shield of water or concrete, which will minimize radiation exposure resulting from a nuclear excursion.

1.7 Liquid Waste Systems

Liquid wastes will be routed from the building according to their composition. Radioactive waste solutions are expected to be of low specific activity and will be sent to the concrete waste tanks. Normally activity-free steam condensate and cooling jacket water will be (a) recirculated through an intermediate cooler or (b) monitored with provisions for impounding or treatment if contamination is detected (see Fig. 1-7).

1.8 Gaseous Waste Systems

Gaseous wastes from the plant originate in the dissolvers, vessels, cells, and building. Figure 1-8 is a schematic flowsheet of the gaseous waste disposal system. The dissolver off-gas will be treated for removal of particulate matter, chemically hazardous materials (nitrogen oxides, hydrogen, and ammonia), and biologically hazardous materials (Kr^{85} and I^{131}). Nitrogen oxide will be removed by reaction with air, caustic scrubbing, and decomposition over a high-temperature rhodium catalyst bed. Hydrogen and ammonia will be reacted with air over a 400°C platinum bed under conditions carefully controlled to preclude formation of explosive mixtures. Iodine is to be removed by caustic scrubbing. Further cleanup will not be necessary unless decay times are materially decreased below the currently anticipated 180 days. Rare gas recovery is to be by absorption in Freon 12 at a pressure of 100 psi and a temperature of -120°C. Alternatively, the dissolver off-gas may be discharged to the atmosphere without treatment at a remote tall stack. Vessel off-gas will be passed through a condenser and an absolute burstproof filter before entering the plant vessel off-gas system for subsequent treatment before discharge. Ventilation air for the cells will be passed through a roughing filter and two absolute filter

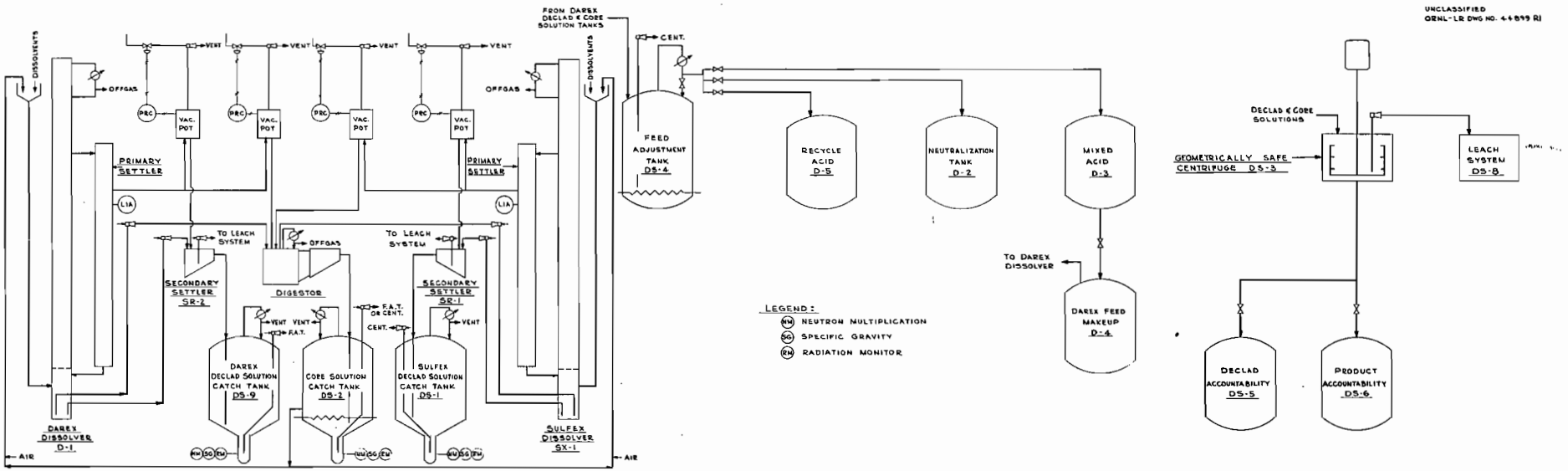


FIG. NO. I-6 PRFR HEAD END FACILITY EQUIPMENT FLOWSHEET

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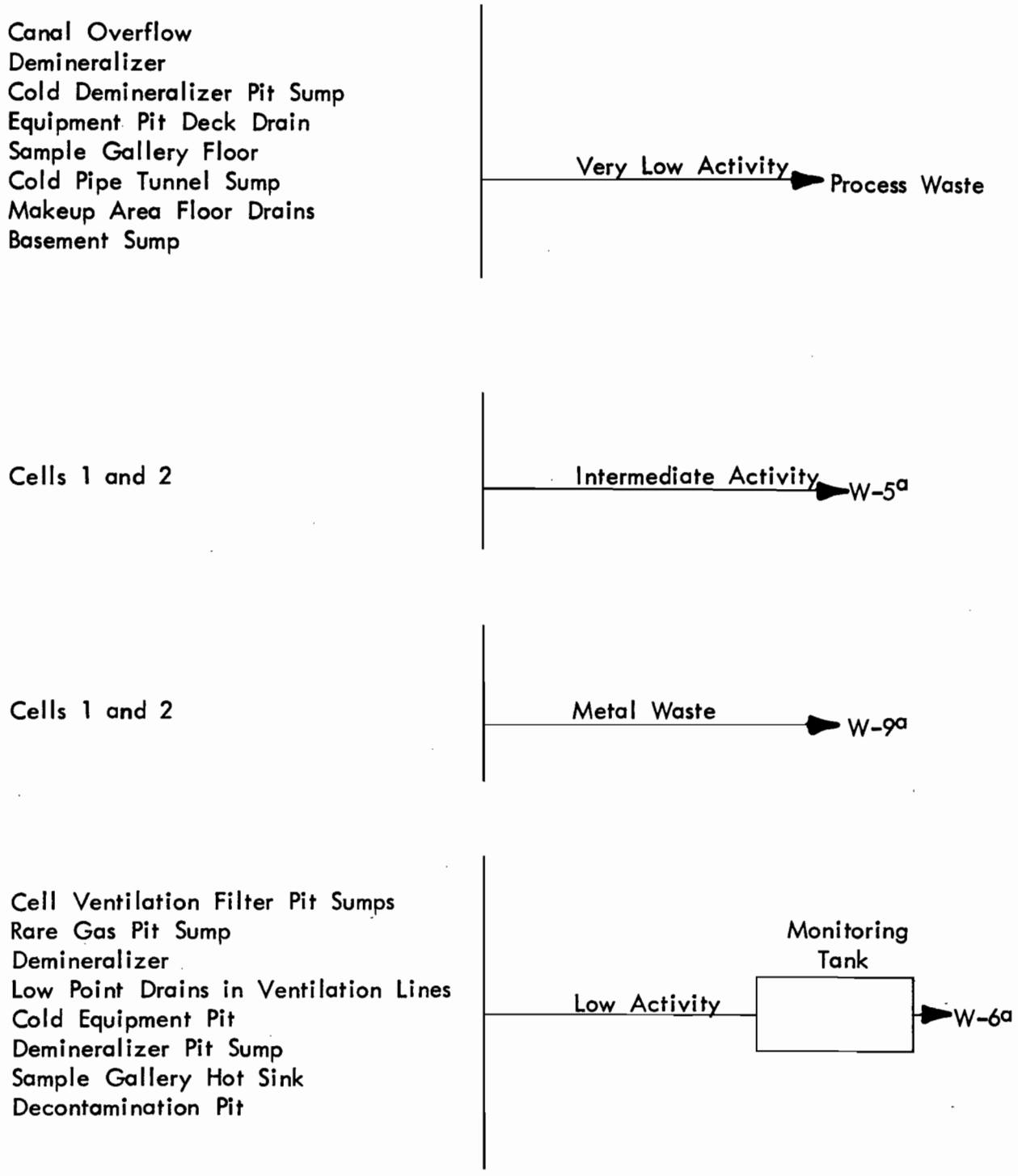

(a) All these tanks are existing uncooled 170,000 gallon capacity concrete tanks.

Fig. I-7. Liquid waste disposal system.

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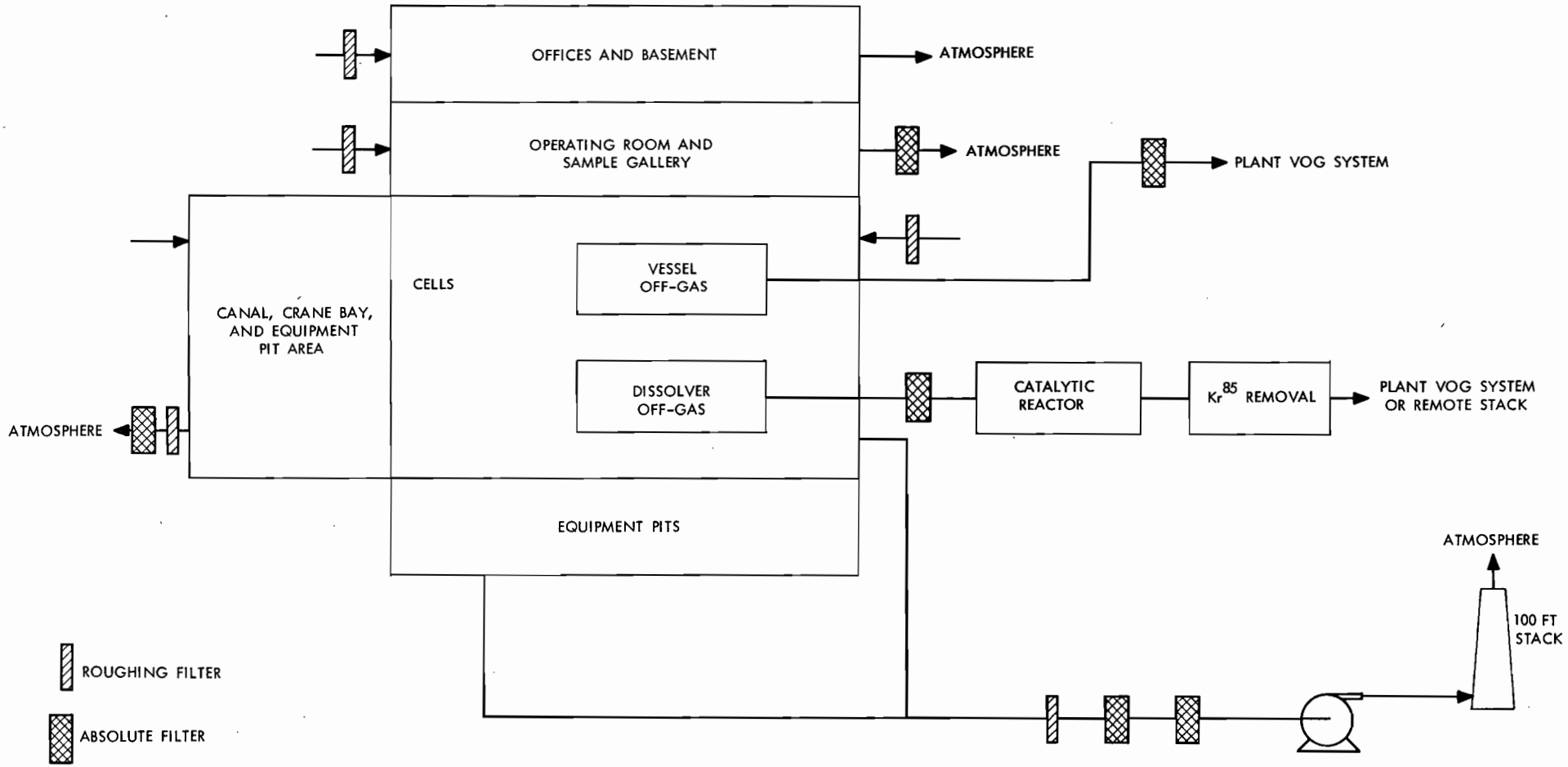

FIG. I-8 SCHEMATIC FLOWSHEET OF BUILDING 2527 GASEOUS WASTE DISPOSAL SYSTEM

systems in series prior to discharge to the environment via a 100-ft local stack. Ventilation air for occupied areas will be either discharged directly to the atmosphere or passed through an absolute filter prior to discharge in the case of potentially contaminated areas.

1.9 Monitoring System

Monitors will be provided in virtually all areas and effluent streams to warn of improper operating conditions. The locations and types of monitoring devices to be installed are:

Area or Gas Stream	Continuous Air Monitor	α	Neutron	Pressure	Other
Control room and makeup area	2	2		1	
Offices	1			1	
Basement	1			1	
Sample gallery	1	1		1	
Equipment pit area	1	1		1	
Crane bay	1	1		1	
Canal	1	1		1	
Cells			4	2	Hydrogen
Cell ventilation air	2*				
Dissolver off-gas				1	Hydrogen, oxygen
Vessel off-gas				1	Hydrogen
Coolant and steam condensate		1			

*One placed ahead of and one behind filtration system.

2.0 SUMMARY*

Building 2527 is to process irradiated fuels containing the following maximum amounts of radioactive materials:

Material	Design Capacity per Day	Maximum Inventory	
		In Plant and Canal	In Single Vessel
I^{131}, curies	0.020[a]	100	0.042[a]
Kr^{85}, curies	930[a]	120,000	1930[a]
Mixed nonvolatile fp's, curies	800,000[a]	100,000,000	1,700,000[a]
U^{233}, kg	1.35[a]	170	2.8[a]
U^{235}, kg	10.5[a]	1400	22[a]
U^{238}, kg	300[b]	10,000	250[b]
Th^{232}, kg	120[a]	15,000	~250[a]
Pu^{239}, kg	0.72[b]	50	0.6[b]

[a]These maxima will occur during the processing of CETR fuel. The listed inventory of U^{235} and U^{233} will not occur simultaneously.

[b]These maxima will occur during the processing of the PRDC radial blanket.

Accidents that can possibly occur in this facility which would result in the release of activity to the environment would be a nuclear excursion in a process vessel or a fire or explosion of the combustible gases generated during fuel dissolution. The probability of occurrence of such accidents will be maintained at an extremely low value by the use of multiple devices which are designed to detect and prevent the occurrence of the accidents. The release of activity to the environment in the event that such accidents occur would be minimal since the process equipment is enclosed within tight cells, which are shielded with 6 ft of concrete, designed to contain a 900-psf shock load, and are ventilated through a burst-resistant absolute filter system which discharges through a local stack.

2.1 Nuclear Safety Hazards

The nuclear safety hazards in Building 2527 are:

*Calculations and assumptions used are found in Vol. I of this report.

Location of Criticality Hazard	Maximum Quantity of Fissionable Material Involved	Shield	Type of Controls
Canal	17,000 kg $U^{235}O_2$-ThO_2[a]	12.5 ft H_2O	Subcritical arrays spaced 12 in. apart edge to edge
Dissolver	310 kg $U^{235}O_2$-ThO_2[a]	6 ft concrete	Geometry, 12 in. max dia
Solids separators	310 kg $U^{235}O_2$-ThO_2[a]	6 ft concrete	Geometry, 12 in. max dia
Dissolver product tank	310 kg $U^{235}O_2$-ThO_2[a]	6 ft concrete	Concentration control, prior solids separation, solids detection
Declad catch tank	310 kg $U^{235}O_2$-ThO_2[a]	6 ft concrete	Prior solids separation, solids detection
Feed adjustment tank[b]	310 kg $U^{235}O_2$-ThO_2[a]	6 ft concrete	Presence of soluble neutron poisons assured by multiple means

[a]The $U^{235}O_2$-ThO_2 mixture will contain a maximum of 7.1 wt % U^{235}.

[b]Site of the credible nuclear accident that would result in the most serious consequences.

The most serious nuclear safety hazard will occur during the processing of ceramic oxide fuels which are subject to fracture and breakage into small particles that can be carried with process streams from the geometrically safe dissolver into other nongeometrically safe vessels. Multiple means are to be provided to detect and/or prevent transfer of undissolved fuel particles to these nongeometrically safe vessels. During the processing of uranium-molybdenum fuels molybdenum-bearing precipitates will either be prevented or retained in limited-geometry equipment because of their capacity for carrying Pu^{4+}.

The maximum credible accident in Bldg. 2527 would be a nuclear reaction that completely shatters a process vessel containing a batch of irradiated CETR solution and scatters its contents throughout the cell. It is believed that such an accident could be caused by a single nuclear burst of 10^{18} fissions, which would release approximately 30,000 Btu and 100 cu ft of gas and steam in a small fraction of a second. The effects of such an accident have been evaluated and are:

	Aerosol Release	Gaseous Fission Product Release
Max downwind integrated dose from VOG release, rem	<0.001	0.154
Max downwind integrated dose from cell ventilation release, rem	0.006	0.650
2-min dose in secondary containment shell, rem	0.005	0.075
Max downwind integrated dose from building release, rem	<0.001	<0.001
Prompt neutron and gamma dose through shield, rem	0.012	

It is seen that the radiation exposures due to escape of long-decayed reactor fuel and "fresh" gaseous fission products to the environment and due to direct gamma and neutron dose through the shield would all be less than 1 rem.

A less serious and more probable type of nuclear accident could occur due to an initial burst of 10^{17} fissions which would not result in rupture of the process vessel. Such a reaction would tend to recur until moderator is boiled away, fissile material is expelled, or a shutdown device is actuated, with the result that aerosol and gaseous fission products would escape to the environment through the VOG system. The maximum downwind integrated dose from the VOG release in such an accident can be calculated from the above tabulation since the dose would be directly proportional to the number of fissions that occur before permanent shutdown of the reaction. It is improbable that more than 10^{18}-10^{19} recurring fissions would occur in a contained reaction in Bldg. 2527 since detection equipment will cause nuclear poisons to be added automatically to all suspected vessels in the cells soon after an initial burst.

2.2 Explosion and Fire Hazards

The explosive and combustible materials that are to be handled in Bldg. 2527 include nitrogen oxides, which can decompose to N_2 and O_2 with the liberation of heat, and hydrogen and ammonia, which can react violently with air. The lower explosive limits of hydrogen and ammonia in air are 4 and 16%, respectively. The maximum inventory of any of the explosive gases in the plant will be 40 cu ft. The inventory in the largest single vessel will be 10 cu ft. Since the energy release of these gases, diluted by air or other components to the extent that they could be in the plant is approximately 100 Btu/cu ft, the total potential energy release in the plant is ~4000 Btu and the potential energy release from the largest single tank is ~1000 Btu.

Equipment that is installed to prevent explosions includes nitrogen purging devices, H_2 monitors, an O_2 monitor, a catalytic reactor for NH_3 and H_2; a catalytic reactor

for nitrogen oxides; and a scrubber for nitrogen oxides. Explosive mixtures are prevented by nitrogen blanketing of the dissolver off-gas system and steam dilution of NH_3 and H_2 in the catalytic reactor. The many devices and administrative procedures that will be used to preclude an explosion are believed adequate to reduce the probability of explosion to a negligible value. Other facilities that produce hydrogen have been operated within the Commission without incident. Various commercial facilities producing explosive materials operate safely on a routine basis. It is therefore concluded that explosive gases can be processed without undue hazard. If the maximum credible explosion should occur, no walls would be shattered and no breaches in the containment or radiation shield would result. Activity could only escape by minor leaks through the cell walls or through the ventilation system.

Since no combustible materials other than the explosive gases previously discussed will exist in the cells, the probability of a fire is negligible and no fire prevention equipment, as such, will be included. The equipment that will be used in fighting a fire in a cell, if such should occur, will be an automatic sprinkler system and automatically actuated valves which will shut off the inlet air supply. The sealed cells and filtered off-gas systems would provide a high degree of containment of a fire.

2.3 Evaluation of Noncriticality Event Leading to Release of Radioactive Material

The effects of a noncriticality explosion in a process vessel filled with radioactive material would be equivalent to a criticality explosion with the exception that a radiation dose through the shield and release of gaseous fission products would not occur. The cell and vessels that could be involved in such an accident is described as follows:

Accident	Explosion in Tank That Contains Largest Amount of Radioactivity	Hydrogen Explosion During Decladding
Vessel name	Feed adjustment tank	Dissolver
Vessel volume, gal	1200	175
VOG purge rate, cfm	50	50
Cell volume, cu ft	15,000	15,000
Cell ventilation purge flow rate, cfm	1500	1500
Solution density, g/ml	1.9	Not applicable
Th^{232}, curies/mg of solution	10^{-10}	Not applicable
U^{235}, curies/mg of solution	10^{-9}	Not applicable
Mixed fp's, curies/mg of solution	6.8×10^{-3}	Not applicable
I^{131}, curies/mg of solution	10^{-10}	Not applicable
Kr^{85}, curies	nil	190
Total amount vaporized or suspended in the cell air after an accident		
Mixed fp's, curies	29.0	nil
curies/m^3 of cell air	0.068	nil
Kr^{85}, curies	nil	190
curies/m^3 of cell air	nil	0.44
Others	Negligible	Negligible

The effects of the accident are:

	Mixed fp's	Kr^{85}
Vessel off-gas release*		
Total amount, curies	0.082	190
Max downwind integrated dose, rem	<0.001	0.003
Distance downwind of max dose, m	580	580
Cell off-gas release*		
Total amount, curies	0.41	190
Max downwind integrated dose, rem	0.0062	0.019
Distance downwind of max dose, m	580	580
Release into secondary containment zone		
Total amount, curies	0.0032	0.021
Concentration, curies/m^3	3.0×10^{-7}	1.9×10^{-6}
2-min dose to building personnel, rem	0.0053	<0.001
Release from secondary containment zone* (assuming ventilation system works)		
Total amount, curies	5.0×10^{-5}	-
Max downwind dose, rem	<0.001	-
Distance downwind of max dose, m	150	150
Ground fallout at 20 m, curies/m^2	1.8×10^{-10}	-
Release from secondary containment zone* (assuming ventilation system fails)		
Total amount, curies	1.22×10^{-3}	-
Max downwind dose, rem	<0.001	-
Distance downwind of max dose, m	150	150
Ground fallout at 20 m, curies/m^2	4.5×10^{-9}	-

*The fallout of particulate matter from this source is such that the ground will not exceed either the hazardous or required decontamination level.

Part II. BUILDING 2527, VERY HIGH ACTIVITY WASTE STORAGE FACILITY

1.0 INTRODUCTION

1.1 Purpose and Uses

Some of the wastes produced in the Power Reactor Fuel Processing development program will be of very high specific activity and of sufficient volume that storage in existing or currently planned waste facilities is not feasible. The proposed waste storage facility is to be used only for the most active wastes. Other waste storage tanks for high-activity solutions are planned for installation at the Laboratory. In general, these tanks probably will be designed to accept wastes having a specific activity an order of magnitude less than will be found in the most active wastes from the Power Reactor Fuel Processing program. Wastes that do not present a heat generation problem will be routed to the concrete waste tanks. A full treatment of the waste storage system for the plant can be found elsewhere.

The very high activity waste storage facility is completely designed and awaiting Commission approval to proceed. The decision to do power reactor fuel processing demonstrations at ORNL is currently being reconsidered; consequently, construction of the storage facility and the head-end process development facility, Bldg. 2527, is being held in abeyance.

1.2 Location and Distance from Other Facilities

The location of the very high activity waste storage facility is west of Third Street between Central Avenue and Burial Ground Road (Fig. I-1 in Part I). Adjacent facilities are listed in Sect. 1.2 of Part I.

1.3 Building Description

The waste tanks will be housed in a concrete shielded vault as shown in Fig. II-1. The vault is ventilated and equipped with a stainless steel floor pan. The vault housing the three 15,000-gal tanks will not be opened after completion of construction except for small ports through which corrosion specimens may pass.

1.4 Personnel Control

The numbers of people normally occupying this waste storage facility and adjacent facilities are given in Sect. 1.4 of Part I. No personnel exclusion ordinarily will be required for reason of radiation hazards or contamination. Operator attention will be required occasionally to make jet transfers, etc., but this will be done in an area completely free of activity. If maintenance should be required on the 250-gal diversion

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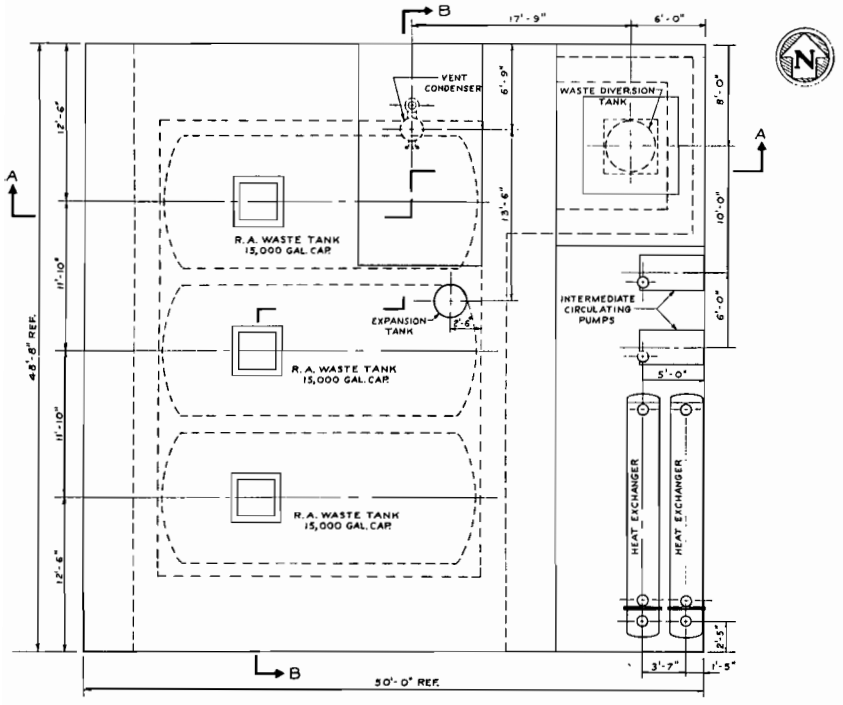
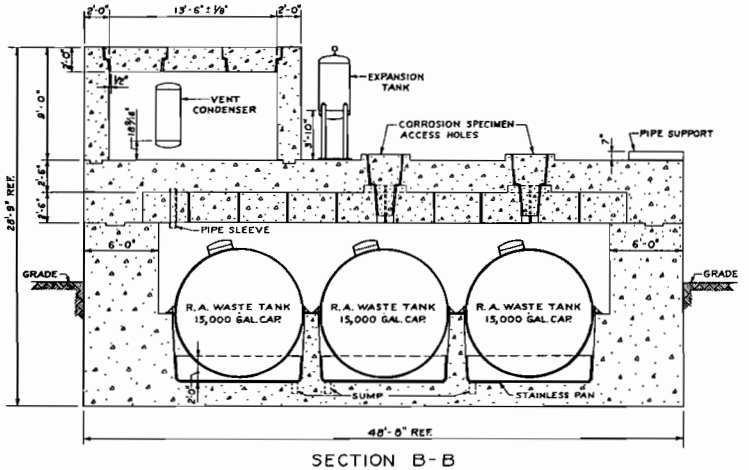
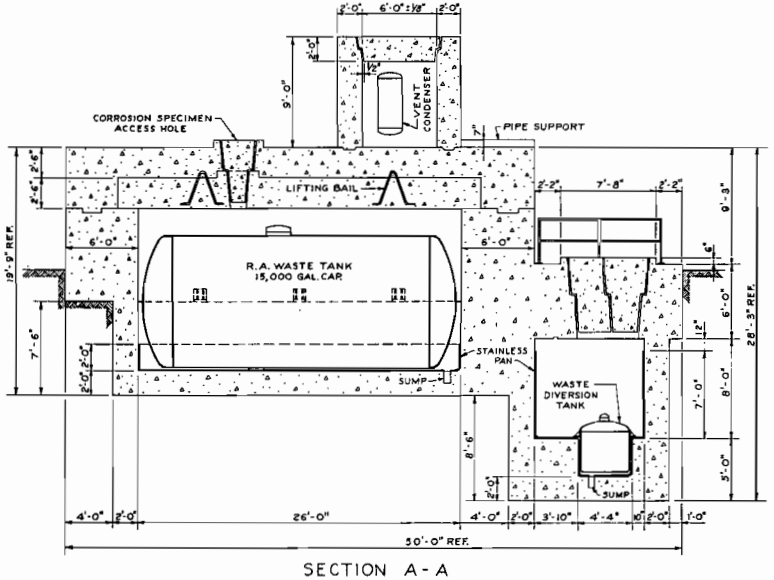


PLAN



SECTION B-B



SECTION A-A

FIG. II 1 SKETCH OF VERY HIGH WASTE STORAGE FACILITIES

tank system, temporary measures will be required to restrict personnel access and to limit the area of potential contamination. A permanent installation is not provided for this purpose because it may never be needed and because temporary construction should be fully adequate and much cheaper.

1.5 Process Description

No processing takes place in the waste tanks other than removal of heat and dilution plus purging of radiolytic hydrogen-oxygen mixtures.

The maximum anticipated heat generation of waste concentrates from six months' decayed fuel is ~600 Btu/gal, which would cause the solution temperature to rise in an uncooled system at the rate of ~65°F/hr. The average specific heat generation of the tank contents will be a function of filling rate and origin of waste. Wastes generated by processes that dissolve fuel and cladding material simultaneously cannot be concentrated by a large factor and will have a specific heat generation of the order of 20 Btu/gal. Each waste tank is to be provided with three parallel submerged cooling coils, each of which can remove 770,000 Btu/hr when the tank temperature is 160°F. There is to be one condenser of 2.5×10^6 Btu/hr capacity to serve all three tanks if any one of them should boil. Duplicate means for solution transfer from one tank to another are to be provided. Coolers are provided for the jet suctions in order that transfers can be made even when the tank contents are boiling. See Fig. II-2 for a simplified flowsheet of the system. Maximum corrosion rates are expected to be $\leq$ 2 mils/yr, giving a minimum tank life of 33 years. Hydrogen-oxygen mixtures generated by radiolytic decomposition of water will be monitored and diluted with air to below the lower explosive limit.

1.6 Criticality

No criticality problem is anticipated. The purpose of the solvent extraction facilities is to recover fissile materials. Their treatment is very thorough and will include chemical analyses prior to discharge to this, or any other, waste system. Only very small quantities and very dilute concentrations of fissile material will enter the waste system.

1.7 Liquid Waste Systems

No contaminated liquid waste will be produced by this waste storage facility unless a leak develops in the cooling system. Even in this case, cooling water should leak into the tank rather than waste entering the coolant stream. The worst eventuality is provided for by use of a completely enclosed coolant circuit, which is in turn cooled by a water-to-water heat exchanger.

1.8 Gaseous Waste Systems

Ventilation air for the shielded vault and vessel off-gas from the storage tanks are the two sources of gaseous waste. Both these streams are routed into the Bldg. 2527

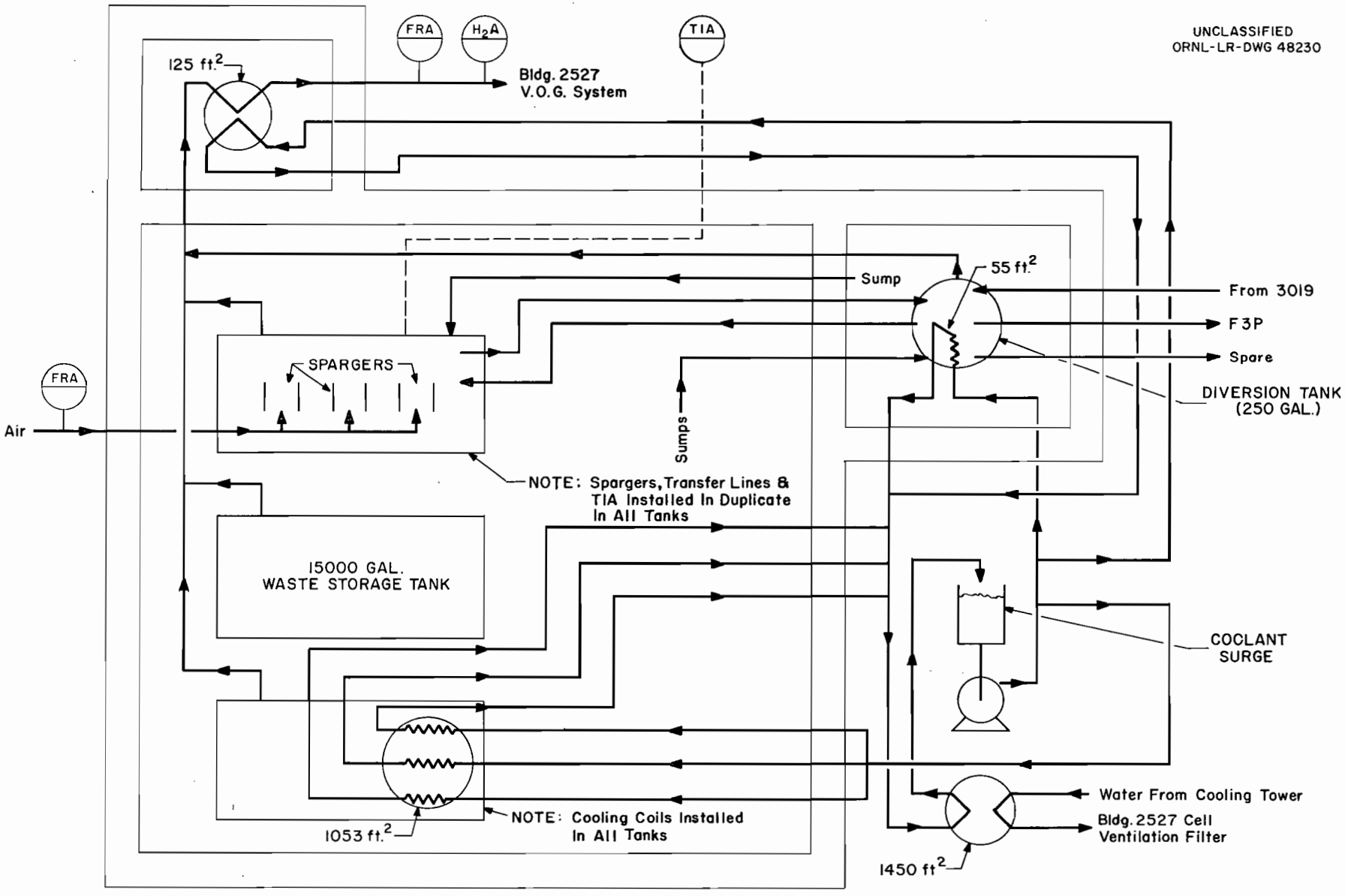


FIG. II-2 SIMPLIFIED FLOW DIAGRAM OF VERY HIGH ACTIVITY WASTE STORAGE FACILITY

off-gas systems. The vessel off-gas is passed through an absolute filter and then goes to the 3039 stack for additional treatment prior to discharge to the atmosphere. The cell off-gas passes through a roughing filter and two absolute filters in series and is then discharged from a 100-ft stack in the vicinity of Bldg. 2527.

1.9 Monitoring Systems

Monitors will be provided as listed in the table below:

Location or Stream	CAM*	Beta-gamma	Pressure	Flow	Hydrogen	Temperature
Ventilation air	1		1	1		
Vessel off-gas			1	1	1	
Air to storage tanks				1		
Outside vault	1	1				
Tank contents						8

* Continuous air monitor.

2.0 SUMMARY

The very high activity waste storage facility will contain large amounts of fission products, but the amount of fissionable material that will be intentionally discharged to the waste tanks is very small:

Material	Maximum Inventory	
	In Single Tank	In Waste System
Iodine-131, curies	Negligible	Negligible
Krypton-85, curies	-	-
Strontium-90, curies	6×10^6	12×10^6
Mixed nonvolatile fission products, curies	1.2×10^8	2.4×10^8
Heavy elements		
Pu-239, kg	0.38	0.38
U-235, kg	3.75	7.25
U-238, kg	75	75
U-233, kg	1.0	1.0
Th-232, kg	75	75
Am-241	Trace	Trace

Gaseous and short-lived fission products will be eliminated prior to storage by vaporization or decay. The amounts of fission products are estimates and will vary with the fuels processed. Strontium-90, which decays with a 28-yr half-life, is the controlling biological hazard. Quantities of heavy elements are based on an assumed loss of 0.1%.

More than 500 kg of fissionable material could be held in solution in one 15,000-gal tank without reaching criticality. Stringent measures are taken in the solvent extraction complex to assure absence of fissile material in the waste in order to maintain nuclear safety in that plant (see Hazards Evaluation of Bldg. 3019) and to prevent loss. Two successive gross errors are required for significant amounts of fissile material to reach the waste storage system: (1) failure of solvent extraction complex or (2) failure to analyze waste solution prior to transfer.

No combustible materials will be used in the construction of the waste storage facility and no combustible material will be stored there. Fires are not seen as a credible accident.

Hydrogen and oxygen are released from aqueous solutions of radioactive materials. In air, the lower explosive limit of hydrogen is 4%. The estimated maximum radiolytic yield of hydrogen is 4 cfh. If a tank filled with a stoichiometric mixture of hydrogen and oxygen exploded, the total energy would be ~220,000 Btu, or the equal of ~20 lb of TNT. Because it is impractical to design a system to contain an explosion of this order, control measures are taken to nullify the probability of such an occurrence:

1. Provide adequate dilution air to reduce hydrogen concentration to <1% in air.
2. Provide a low flow alarm on dilution air to all tanks.
3. Provide a low flow alarm on composite air from the three tanks.
4. Provide a hydrogen concentration monitor-alarm on composite air from the three tanks.
5. Electrically ground all equipment.
6. Provide flame arrestor in vessel off-gas line.

These measures are believed adequate to reduce the probability of a hydrogen explosion to the point of incredibility.

BUILDING 2527, FUEL STORAGE AND DISSOLUTION FACILITY. Part I.

3.0 PLANT AND PROCESS DESCRIPTION

3.1 Plant Description

The main building of the proposed addition will be a new shielded head-end pilot plant, Bldg. 2527, to be constructed generally of concrete, concrete block, and steel, with Class I roof and insulated aluminum siding enclosing the crane bay and canal. It will be composed of four major areas: (1) a concrete shielded cell area; (2) cell auxiliaries; (3) an enclosed canal area; and (4) a main operating area.

3.1.1 Concrete-Shielded Cell Area

The concrete-shielded cell area (Figs. I-2, 3, 4, 5, 9, 10) will consist of: two large cells, 25 x 20 x 30 ft high, with stainless steel-lined floors and walls and with 6-ft-thick concrete walls and 5-ft-thick ceiling having openings as required for equipment and service sleeves and removable roof plugs; and a third (centrifuge) cell located in the northwest corner of large cell 1. The inside dimensions of this cell will be 11 x 9 ft, walls will be 2-ft-thick solid concrete blocks, and floor and walls will be lined with stainless steel. These hot cells will house equipment for the Darex, Sulfex, and Zirflex processes.

The 6-ft-thick cell walls and 5-ft-thick roof, which provide primary containment, are designed to contain a shock wave pressure of 900 psf and a shock wave energy of 230 ft-lb/sq ft, which would result from the explosion of 3 lb of TNT in the center of the cells.

3.1.2 Cell Auxiliaries

Cell auxiliaries will consist of: a sampling gallery, approximately 11 x 62 ft, located immediately east of the cell area and containing a sample room and two instrument transmitter rooms; off-gas, and "hot" and "cold" equipment pits located immediately beneath the sampling gallery; pits for a water demineralizer system for the storage canals; and a rare gas recovery system located immediately east of the off-gas and equipment pit area (Figs. I-2, 3, 4, 5, 9, 10).

3.1.3 Enclosed Canal Area

The south portion of the facility will be an enclosed canal area (Figs. I-2, 3, 4, 5, 9, 10) approximately 33 ft wide by 110 ft long, containing water of varying depths, up to a maximum of 35 ft, to provide a receiving station (unloading and decontamination pits) and two storage canals for large power reactor fuel elements.

3.1.4 Main Operating Area

The main operating area will consist of two stories and a basement, approximately 36 x 64 ft in size, located west of the cell area (Figs. I-2, 3) and will provide space for offices, change and toilet room facilities, process makeup area, control room, and a basement equipment room.

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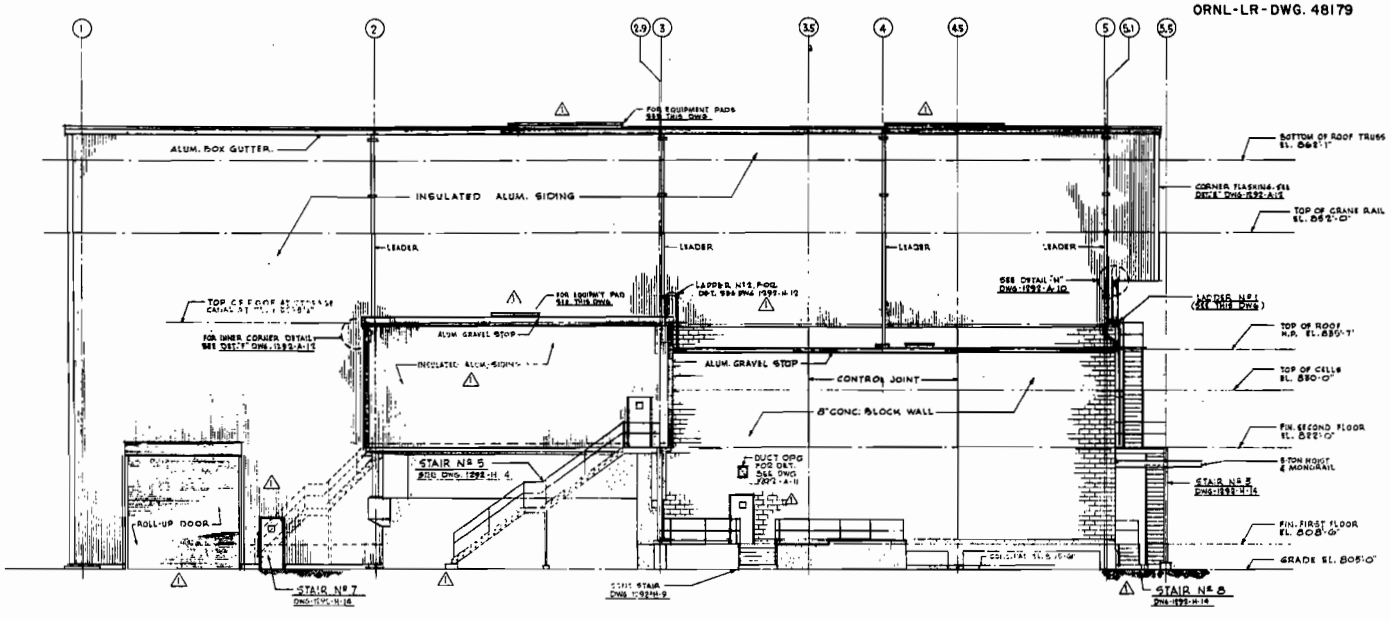
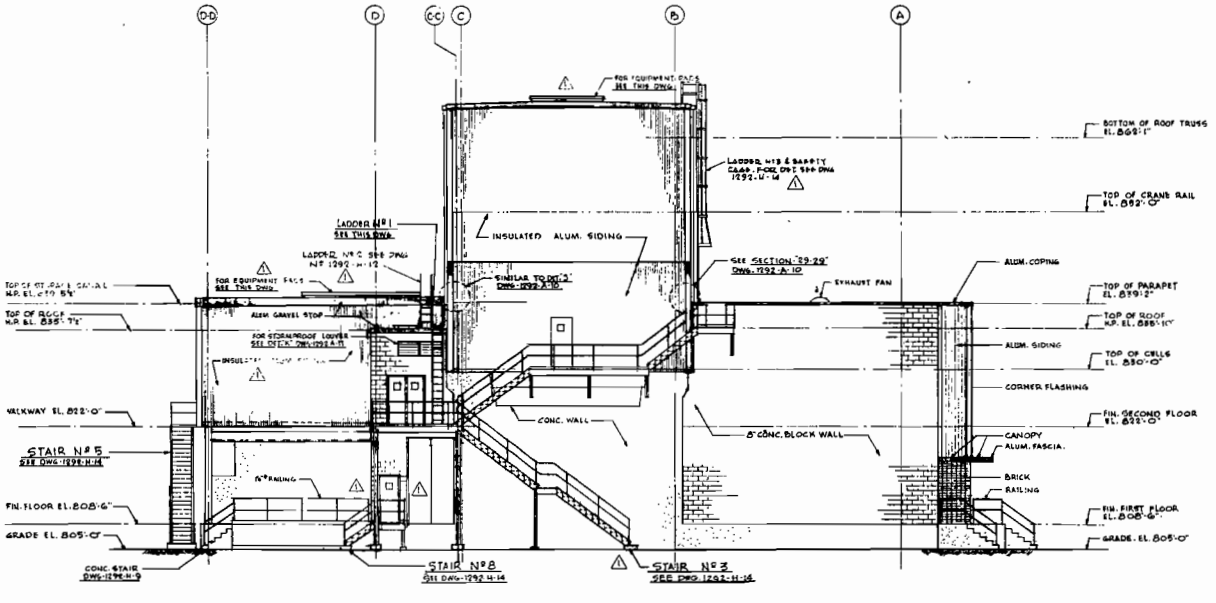


EAST ELEVATION
SCALE: 1/8"=1'-0"



NORTH ELEVATION
SCALE: 1/8"=1'-0"

FIG. I-9 BLDG. 2527-EAST & NORTH ELEVATIONS

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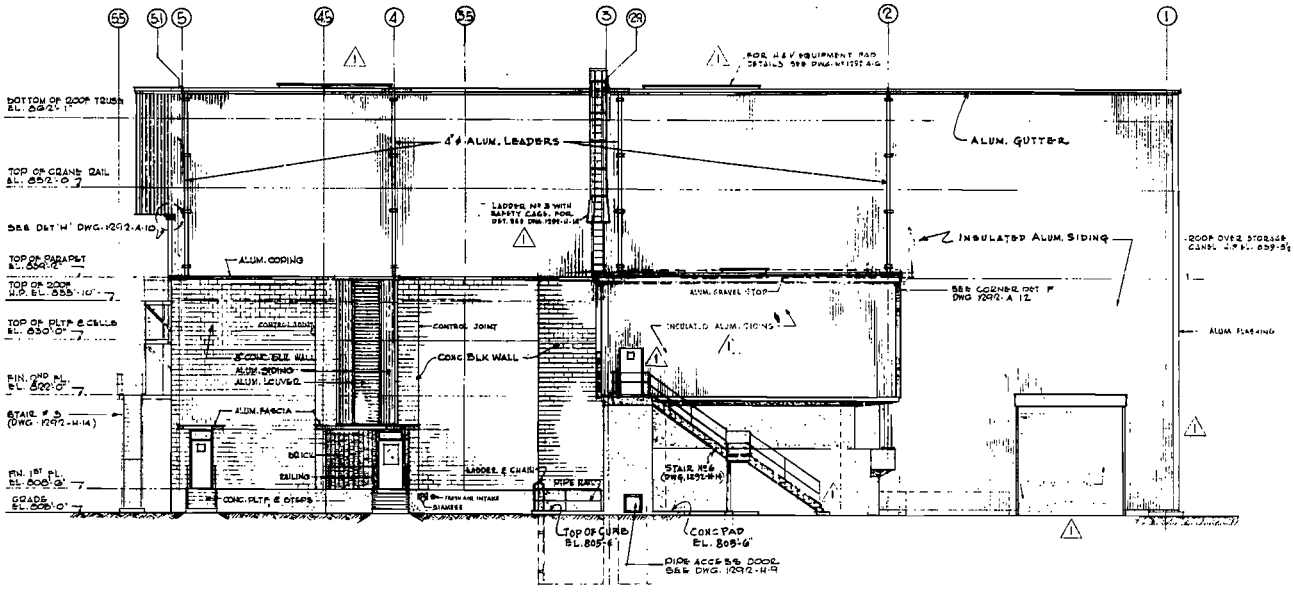
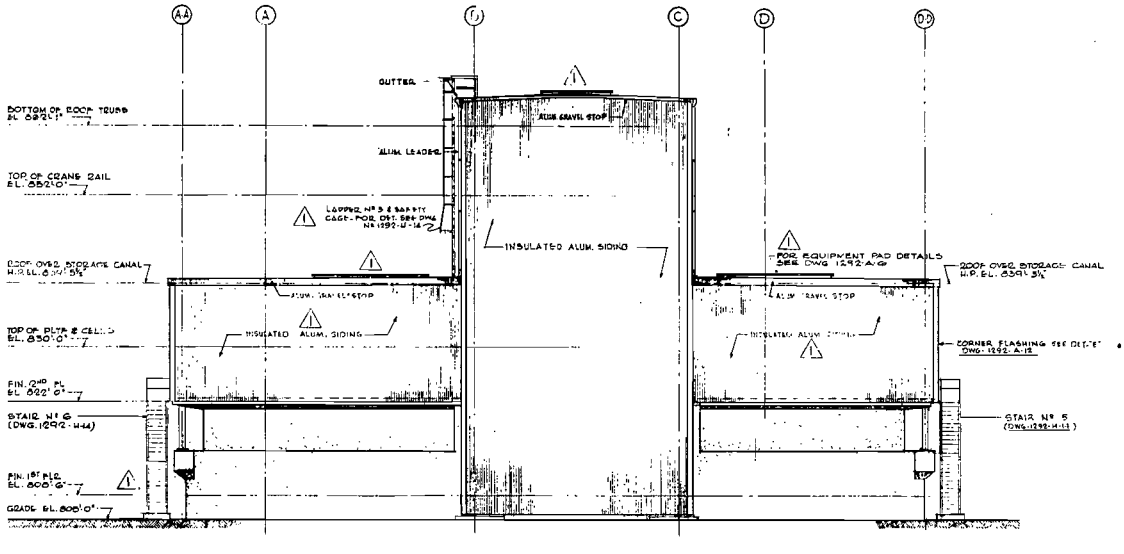


WEST ELEVATION



SOUTH ELEVATION

FIG. I-10 BLDG. 2527 - SOUTH & WEST ELEVATIONS

An enclosed craneway, approximately 144 ft long, will provide 75-ton and auxiliary 10-ton handling capabilities from the roadway located immediately south of the canal area, across the canal and over both cells.

3.2 Process Description

The process equipment to be installed in the ORNL Power Reactor Fuel Processing Pilot Complex is to demonstrate head-end processes applicable to the recovery of uranium and plutonium from spent power reactor fuels and to recover fissile material from all thorium fuels, stainless steel-clad low-enrichment uranium fuels, and nondomestic research reactor fuels. The processes that will be demonstrated are: (1) Darex, a total dissolution process applicable to stainless steel-clad uranium fuels of all enrichments or a selective dissolution process for stainless steel-clad thoria fuels; (2) Sulfex, a selective dissolution process using sulfuric acid for dissolution of stainless steel cladding and nitric acid for core dissolution; applicable only to medium- and low-enrichment fuels; and (3) Zirflex, a selective dissolution process similar to Sulfex except that ammonium fluoride is used to dissolve the Zircaloy cladding.

3.2.1 Darex Process

The Darex process accomplishes total dissolution of stainless steel-clad uranium-bearing fuels in boiling dilute aqua regia (5 N HNO_3-2 N HCl). Although this solution is very corrosive to most metals, titanium is almost completely passive to aqua regia. Dissolution of stainless steel in aqua regia is rapid, evolving appreciable quantities of nitrogen oxides but essentially no hydrogen. After the dissolution, chloride is removed from the dissolver product by evaporation in the presence of ~9 M nitric acid so that all subsequent operations may be carried out in stainless steel equipment. Dilution to workable concentrations follows chloride removal. Because appreciable quantities of silica are usually present in the cladding and/or core material, it probably will be necessary to clarify the product to make it suitable for solvent extraction feed. Figure I-6 illustrates the flowsheet of equipment to be installed in Bldg. 2527.

The Darex process can also be used for chemical decladding of stainless steel-clad thoria fuels due to the low dissolution rate of thoria in aqua regia. Losses of fissile material to aqua regia have been low in tests performed to date on both irradiated and nonirradiated fuels. Thoria is dissolved, after removal of the cladding solution, in fluoride-catalyzed nitric acid. Titanium seems to be an adequate material to contain the thoria dissolvent when suitable complexing agents are used.

3.2.2 Sulfex Process

The Sulfex process is applicable to uranium- or thorium-bearing fuels of medium- to low-enrichment (10% U^{235} maximum) clad in stainless steel. Dilute sulfuric acid (3 to 6 M) dissolves stainless steel at adequate rates, releasing approximately 8 cu ft of hydrogen per pound of stainless steel dissolved. The reaction between stainless steel and sulfuric acid can be initiated by contacting the fuel assembly with a specimen of active metal such as iron or zinc. When dissolution of the stainless steel cladding is almost complete, the solution

temperature is lowered and the supernatant liquor is transferred to a catch vessel. Rinsing follows to prevent contamination of the core dissolvent with decladding solution. The core can then be dissolved in the appropriate solution, 8 M HNO_3 for uranium or 13 M HNO_3-0.04 M HF-0.1 M $Al(NO_3)_3$ for thorium. The vessel must then be rinsed to prevent loss of dissolved core material to the decladding solution in the next batch. Tests to date on reactor grade oxide indicate that soluble losses of uranium from uranium oxide pellets are < 0.1%. In the handling of ceramic core fuels, the separation of fines from the solution of the clad may be required. The quantity or size of particles produced is not yet accurately known, although current indications are that this problem is not severe. A centrifuge or filter is planned for removal of solids from both the clad and core solutions.

Nionel, an International Nickel Company alloy, appears to be an adequate material of construction for containment of the declad solution and for all anticipated core dissolvents. Zirconium shows excellent corrosion resistance to Sulfex solutions provided the core dissolvent does not contain fluorides (necessary for thorium or thoria dissolution).

The quantities of radioactive and fissile material that will be prepared for solvent extraction in Bldg. 2527 are given in Table I-1.

3.3 Waste Systems Description

3.3.1 Process Liquid Wastes

Process liquid wastes are characterized by their normal activity content resulting from having been in contact with irradiated fuel, or having been a part of a fuel solution, or having removed activity from a waste stream or equipment piece. Approximate volumes and activity content of process waste solutions are:

Source	Liters per Operating Day	Curies per Liter
Decladding waste	1200	0.6
Evaporator condensate	1800	0.05
Miscellaneous	100	0.5
Metal waste	nil	1000
Caustic from scrubbers	400	0.04

These wastes will be routed from the building to the 170,000-gal concrete waste storage tanks (described elsewhere) via 2-in. stainless steel pipes. Metal waste will be sent to a separate storage tank from the other wastes. Wastes having negligible specific activity will be routed to the plant process waste systems. The origin and disposition of process waste is shown in Fig. I-7.

Table I-1. Quantities of Radioactive and Fissile Material Handled in Building 2527

Fuel	Head End Process	Maximum Fuel Irradiation	Batch Size[a]			Batch Cycle Time,[b] hr
			Fuel Material Content, kg	Maximum Fissile Material Content, kg	Maximum Activity Content,[c] curies	
Nondomestic RR	Total dissolution in HNO_3-Hg^{++}	20% burnup	10 U total	2.0 U^{235}	80,000	10
CETR	Darex total dissolution, or Darex or Sulfex declad-Thorex dissolution	20,000 Mwd/t	250 Th	22 U^{235}	17×10^5	50
Rural Coop	Darex total dissolution, or Darex or Sulfex declad-Thorex dissolution	5,500 Mwd/t	276 Th	11 U^{235}	5.5×10^5	55
Borax IV	NaOH-$NaNO_3$ plus HNO_3 declad-Thorex dissolution	5,000 Mwd/t	152 Th	9.8 U^{235}	4.6×10^5	30
NMSR	Darex total dissolution, or Sulfex declad-HNO_3 dissolution	7,400 Mwd/t	222 U total	10 U^{235}	3.7×10^5	18
FWC-EC	Darex total dissolution, or Sulfex declad-HNO_3 dissolution	10,000 Mwd/t	162 U total	3 U^{235}	4.9×10^5	19
EGCR	Darex total dissolution, or Sulfex declad-HNO_3 dissolution	10,000 Mwd/t	200 U total	5 U^{235}	4.7×10^5	16
CPPD	NaOH-$NaNO_3$ declad,[d] HNO_3 dissolution	3,000 Mwd/t	212 U total	7 U^{235}	2.8×10^5	17
PRDC Radial	NaOH-$NaNO_3$ declad,[d] HNO_3 dissolution	1,000 Mwd/t	252 U total	0.6 Pu	0.5×10^5	20

[a] The maximum quantity that will be contained in any one vessel.

[b] The residence time of a fuel batch in Bldg. 2527 equipment.

[c] The maximum concentration of activity in solution will occur in the feed adjustment tank during the CETR program. This maximum concentration at 1000 g Th/liter will be 6800 curies/liter ($\sim 6.8 \times 10^{-3}$ curie/mg of solution).

[d] It is assumed that this fuel is mechanically declad and reclad in aluminum in Bldg. 3026.

3.3.2 Cooling Water and Steam Condensate

Cooling water and steam condensate from heat exchangers and vessels that contain radioactive materials will be (a) recirculated through an intermediate heat exchanger cooled by cooling tower water or other economical means, or (b) will be monitored by sensitive monitoring instruments, with provisions for impounding and treatment if contamination is detected. The streams involved will normally be completely free of radioactivity and will be contaminated only infrequently by equipment failure.

3.3.3 Dissolver Off-gas Systems

Dissolver off-gases from Bldg. 2527 will be either scrubbed, filtered, and exhausted to the atmosphere through a remote stack or passed through a rare gas removal system and exhausted to the atmosphere through the ORNL vessel off-gas system. The choice of system will be determined by the availability of funds; the rare gas recovery system is the most desirable alternative, but it is questionable if the additional expenditure for the system is warranted in view of the apparently satisfactory remote atmospheric disposal system.

The gaseous activity that will be generated in the Bldg. 2527 dissolvers in the interim processing of power reactor fuels is tabulated in Table I-2. The maximum activity that will be released in a single dissolution will be 1930 curies of Kr^{85} from a batch of CETR fuel. Iodine and xenon isotopes will have decayed to such an extent before the fuel is processed that they will be biologically insignificant compared with the Kr^{85}, and particulate activity will be filtered below the significant level. The gas will also be passed through a caustic scrubber, which will further remove approximately 99.9% of the iodine in the DOG.*

Remote Stack Disposal of Dissolver Off-gases. The peak release rate of Kr^{85} to the dissolver off-gas system will occur during the decladding of CETR fuel when up to 1200 curies of Kr^{85} may be evolved during a 20-min period (this release rate is at least a factor of 10 higher than that expected). Using this maximum emission rate, 1 curie/sec, calculations were made to determine the peak, maximum average, and yearly average ground dose rates that would result from the release from the existing 3039 stack and conceptual remote stacks located on Haw Ridge and Melton Hill. Properties of these stacks are listed in Table I-3. The results of the calculations are tabulated in Table I-4.

The peak ground dose rates were calculated by applying Sutton's continuous elevated point source diffusion equation with a negligible stack height and vigorous lapse rate. This method was suggested by Gifford.** Atmospheric stability parameters and diffusion coefficients from AECU-3066*** were used.

*K. V. Seyfrit, "The Removal of Iodine from Dissolver Off-gas by Caustic Scrubbing," HW-50259 (1957).

**Gifford, U.S.W.B.O., Oak Ridge, Tenn., personal communication, January 1960.

***AECU-3066, U. S. Dept. Commerce, "Meterology and Atomic Energy," (July 1955).

Table I-2. Gaseous Activity That Will Be Released through Building 2527 Dissolver Off-gas System

Reactor Fuel	Fuel Batch Size, kg	Batch Cycle Time, hr	Activity per Batch, curies			Released to Atmosphere per Batch, curies					
						From Rare Gas Recovery			From Remote Stack		
			Kr^{85}	I^{131}	Xe^{133} x 10^5	Kr^{85}[a]	I^{131}[b] x 10^5	Xe^{133}[a] x 10^7	Kr^{85}	I^{131}[b] x 10^5	Xe^{133} x 10^5
CETR	250 Th	50	1930	0.042	2.4	19.0	4.2	2.4	1930	4.2	2.4
Rural Coop	276 Th	55	720	0.024	1.3	7.2	2.4	1.3	720	2.4	1.3
NMSR	222 U	18	680	0.010	0.61	6.8	1.0	0.61	680	1.0	0.61
FWC-EC	162 U	19	610	0.015	0.85	6.1	1.5	0.85	610	1.5	0.85
CPPD	212 U	17	280	0.011	0.64	2.8	1.1	0.64	280	1.1	0.64
PRDC Radial	252 U	20	94	0.003	0.17	0.94	0.3	0.17	94	0.3	0.17
EGCR	200 U	16	780	0.008	0.45	7.8	0.8	0.45	780	0.8	0.45

[a] The rare gas recovery system is designed such that it will recover 99% of the xenon and krypton.

[b] It is assumed that 99.9% of the iodine will be removed in the caustic scrubber (Reference HW-50259).

Table I-3 Characteristics of Stacks

Stack	Stack Height, ft	Effective Sea Level Elevation of Stack, ft	Average Annual Wind Speed at Top of Stack, mph
X-10, 3039	250	1100	5.0
Haw Ridge	500	1500	9.0
Melton Hill	100	1456	8.5

Table I-4. Ground Dose Rates Resulting from Continuous Emission of 1 curie/sec of Kr^{85} from Stacks

Stack	Ground Location	Differential Elevation, ft	Distance, ft	Wind Frequency, %	Ground Dose Rate,* mr/hr		
					Peak	Average Max.	Average Yearly
3039	X-10	300	460	30	250	2.5	0.05
	TSF	40	11,500	2	0.77	0.63	0.038
	HRT	260	3,500	8	7.3	3.4	0.17
	EGCR	240	12,500	10	0.65	0.54	0.10
Haw Ridge	X-10	700	2,000	10	9.8	0.25	$< 10^{-5}$
	TSF	440	10,000	7	0.56	0.33	0.023
	HRT	660	2,000	10	9.8	0.29	$< 10^{-5}$
	EGCR	640	11,000	15	0.50	0.31	0.025
Melton Hill	X-10	656	7,000	8	1.1	0.31	0.0013
	TSF	396	4,000	8	3.0	0.85	0.005
	HRT	616	4,000	15	3.0	0.35	$< 10^{-5}$
	EGCR	596	13,500	15	0.34	0.20	0.063

*To obtain the ground concentration in curies/m^3 or the dilution factor in sec/m^3, multiply the dose rate values by $1.2 \times 10^{-5}/2.5$.

These peak dose rates represent what is thought to be the maximum ground dose rate that will ever occur. Near the stacks, where the peak dose rates are much higher than the average, the peaks will have a short duration (~1 sec) and a small frequency of occurrence (several times per hour). Far away from the stacks where the peaks approach the average, the peaks will have a somewhat longer duration and larger frequency of occurrence.

The maximum average ground dose rates, which represent the maximum ground dose rates averaged over about 1 hr, were calculated by Sutton's equation.

The average annual dose rate for a particular location was calculated from a modified Sutton equation, taking full credit for distance from the stack, stack height, and wind directional frequency. A neutral stability parameter with corresponding diffusion coefficients was used. Wind roses for the X-10 area from ORO-99* were used. In general, conservative values of wind frequency were used in order to compensate for cross wind diffusion.

The calculated peak ground dose rates in the X-10 area due to 1 curie/sec of Kr^{85} being evolved from the 3039, Haw Ridge, and Melton Hill stacks will be 250, 10, and 1 mr/hr, respectively. The maximum average dose rate for the three stacks, in the same order, will be 2.5, 0.25, and 0.31 mr/hr, while the yearly average dose rate for all three stacks will be less than 0.05 mr/hr. It is calculated that the peak ground dose rates due to activity from any of the stacks will not exceed 10 mr/hr at the HRT, 3 mr/hr at the TSF, and 0.7 mr/hr at the EGCR.

The calculations clearly indicate that since, in reality, Kr^{85} would be dispersed at a rate of 1 curie/sec for more than 20 min per batch of fuel (about 70 minutes per week), direct disposal of the dissolver off-gases to a remote stack would be safe. Direct disposal of the gases to the 3039 stack would probably not be feasible, however, since the high peak dose rates, up to 250 mr/hr, would disturb ORNL low-level counting experiments and Health Physics monitoring procedures.

Dissolver Off-gas Disposal through Rare Gas Removal System. The conceptual rare gas removal system (Fig. I-11) can be divided into two main sections: the gas preparation section upstream of the main compressor and the rare gas removal section beyond the compressor.

Two gas preparation systems will be used. If the off-gas contains oxides of nitrogen (which are present in Darex off-gas and in off-gas from nitric acid dissolution of uranium and UO_2), air will be added to the surge tank to convert the NO to NO_2. Subsequently, the gas will be scrubbed in caustic scrubber A for NO_2 removal. The gas stream will then be heated, filtered to remove particulate matter, and sent to an N_2O reactor to decompose the N_2O into N_2 and O_2 over a 400°C rhodium catalyst bed. After being cooled, the gases will be sent to the main compressor. Off-gases containing either hydrogen (from Sulfex) or ammonia (from Zirflex) will be caustic-scrubbed and filtered as before, but air will not be added to the surge tank. After filtration, the hydrogen or ammonia will be by-passed to other equipment, where it will be reacted with air over a platinum catalyst at 400°C in the presence of sufficient steam to dilute the H_2-O_2 or

*U.S.W.B.O., Oak Ridge, Tenn., "A Meteorological Survey of the Oak Ridge Area," ORO-99 (November 1953).

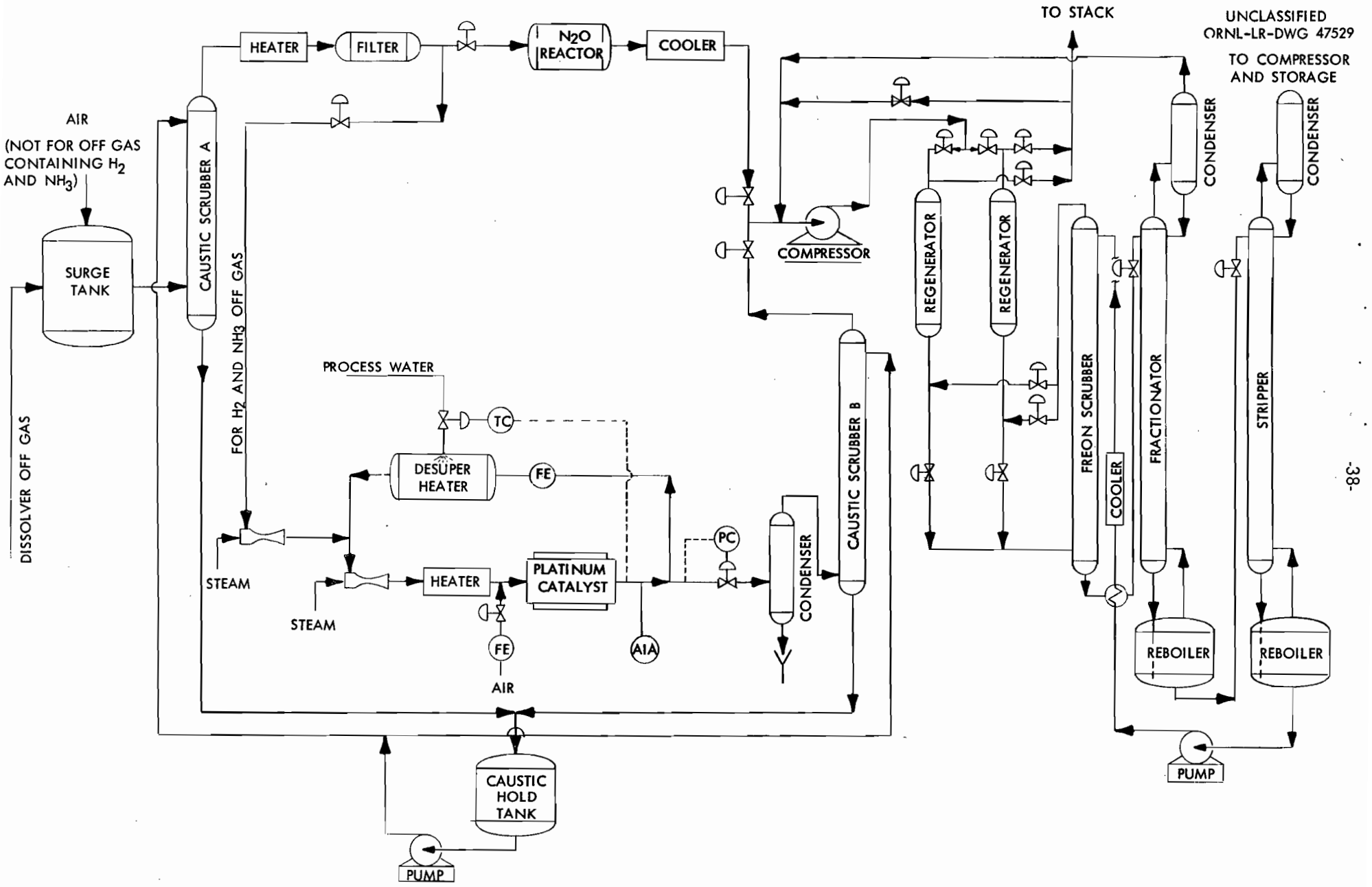


FIG. I-11 Rare gas removal – H_2 and NH_3 combustion flowsheet.

NH_3-O_2 mixture below its lower explosive limit. A recirculating steam system motivated by a steam jet will be used to reduce the over-all volume of steam required for this dilution. Temperature control of the system will be accomplished by desuperheating the recycle steam with process water. Excess steam, rare gas, and NO formed during the combustion of NH_3 will be vented through a pressure controller to a condenser. Caustic scrubber B will then remove the NO_2 formed by the reaction of NO with the excess oxygen in the system. Therefore the gas streams entering the main compressor from either scrubber B or the N_2O reactor will contain essentially N_2, O_2, and the rare gases, and will be saturated with water vapor.

Gases entering the rare gas removal system will be compressed to about 100 psi. Cooling, condensation of water vapor, and residual nitrogen oxides removal will occur in the regenerator system. The rare gases will then be absorbed in Freon-12 at about -120°C in the liquid Freon scrubber. The decontaminated gas stream will return to the regenerator system to provide refrigeration for the feed gas and to pick up deposited water and nitrogen oxides for disposal at the stack. A separation of the rare gases from absorbed N_2 and O_2 will be carried out in the fractionator, but the overhead from this column will contain sufficient rare gas to necessitate its recycle to the feed gas stream. After being stripped from the solvent, the rare gases will be compressed and then stored in shielded cylinders.

The estimated number of curies of activity that will be released to the atmosphere per day with the rare gas removal system during processing of the ORNL-committed power reactor fuels, decayed six months, are shown in Table I-2. Particulate activity has not been included because, after the DOG has been passed through a scrubber and filters, this type of activity will occur as a smaller fraction of the permissible ground concentration than the Kr^{85}.

Average ground concentration will be highest during the 75-day CETR processing program. If average annual weather conditions exist during this campaign, the 3039 stack plus atmospheric dilution will be 1.6×10^{-5} sec/m^3 and the corresponding maximum average ground concentration will be 4.3×10^{-9} μc/cc. This ground concentration, which would occur in the X-10 area, corresponds to 0.04% of the maximum permissible concentration of Kr^{85}, the controlling radioisotope, in air for 40 hr/week occupational exposure, 1.2×10^{-5} μc/cc. The dilution at a distance 5 miles from the stack, during average annual weather conditions, would be 2×10^{-8} sec/m^3, which corresponds to an average ground concentration 0.00003% of the mpc. The maximum average ground concentration during the time required for decladding CETR fuel would be 1% of the mpc, with instantaneous peak values as high as the mpc.

3.3.4 Vessel Off-gas System

Process vessels in Bldg. 2527 will be vented to the 3039 stack through a 6-in. stainless steel pipe off-gas line. The line will contain a condenser and absolute burst resistant deep bed filter in the proximity of Bldg. 2527 to remove steam and the bulk of the particulate material from the air streams prior to discharge of the air to the stack. The only air that will be released through the vessel off-gas system will be that being used in the process to sparge vessels

or airlift solutions plus relatively small amounts of instrument air and in-leakage.

Experience in evaporating radioactive solutions has shown that the activity of evaporated water or steam from the solution is negligible compared with the activity of the liquid particles in the gas stream. These liquid particles, which vary in size from a fraction of a micron to several hundred microns depending on the velocity of the gas stream, have essentially the same composition as the solution from which they have escaped. Using particle size distributions obtained in cooling tower experiments,* Thorex Pilot Plant sparging data,** and settling velocities given in the Chemical Engineers Handbook,*** it was possible to derive a relationship which yields the maximum particulate concentration in an air stream as a function of the minimum superficial velocity of the stream. This relationship is shown in Fig. I-12. For superficial velocities less than about 0.15 ft/sec, an aerosol formed by vigorous mixing of a solution with air is metastable and contains particles with a mean particle size of approximately 10 microns and has a concentration in the order of 10-15 mg/m^3.† After several changes of direction in the off-gas line, the mass mean particle size will be in the order of 3 microns and the concentration will be about 10 mg/m^3. The particle size distribution at this point will contain equal weight fractions of particles in the ranges of $< 0.4\ \mu$, 0.4 to $1.3\ \mu$, 1.3 to $3\ \mu$, 3 to $5\ \mu$, and $> 5\ \mu$.†† This particle size distribution is shown on logarithmic probability paper in Fig. I-13.

The concentration of liquid particles that pass through an absolute filter may be calculated from the data of Fig. I-2 and characteristics of the filter. It is conservatively assumed that a CWS or Cambridge type absolute filter will remove all particles greater than $5\ \mu$, 99.95% of particles in the range 5-$0.3\ \mu$, 95% of particles between 0.3 and $0.1\ \mu$, and 87% of particles less than $0.1\ \mu$. Using these assumptions and the particle size distribution given in Fig. I-2, the concentration of liquid particles in filtered air would be

$$10\ (mg/m^3)\left[1 - (0.9995)(0.835) - (0.95)(0.085) - (0.87)(0.080)\right] = 0.14\ mg/m^3$$

The maximum permissible concentration in air of radioactive aerosols that contain typical PRFP solutions, six months' decay, were calculated from data in NBS-69††† If the maximum permissible concentrations are weighted, using the fraction of activity of each type in a typical PRFP solution containing 6.8 β-γ

*K. K. McKelvey and M. Brooke, "The Industrial Cooling Tower," Elsevier Publishing Co., New York, 1959.

**J. R. Parrott, ORNL, personal communication, Oct. 20, 1959.

***J. H. Perry, "Chemical Engineers Handbook," McGraw-Hill Book Co., Inc., New York, 1950.

†P. L. Magill et al., "Air Pollution Handbook," McGraw-Hill Book Co., Inc., New York, 1956; S. K. Friedlander et al., "Handbook on Air Cleaning," USEAC, 1952.

††J. C. Suddath, ORNL, personal communication, Oct. 21, 1959.

†††NBS Handbook 69, "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure," June 1959.

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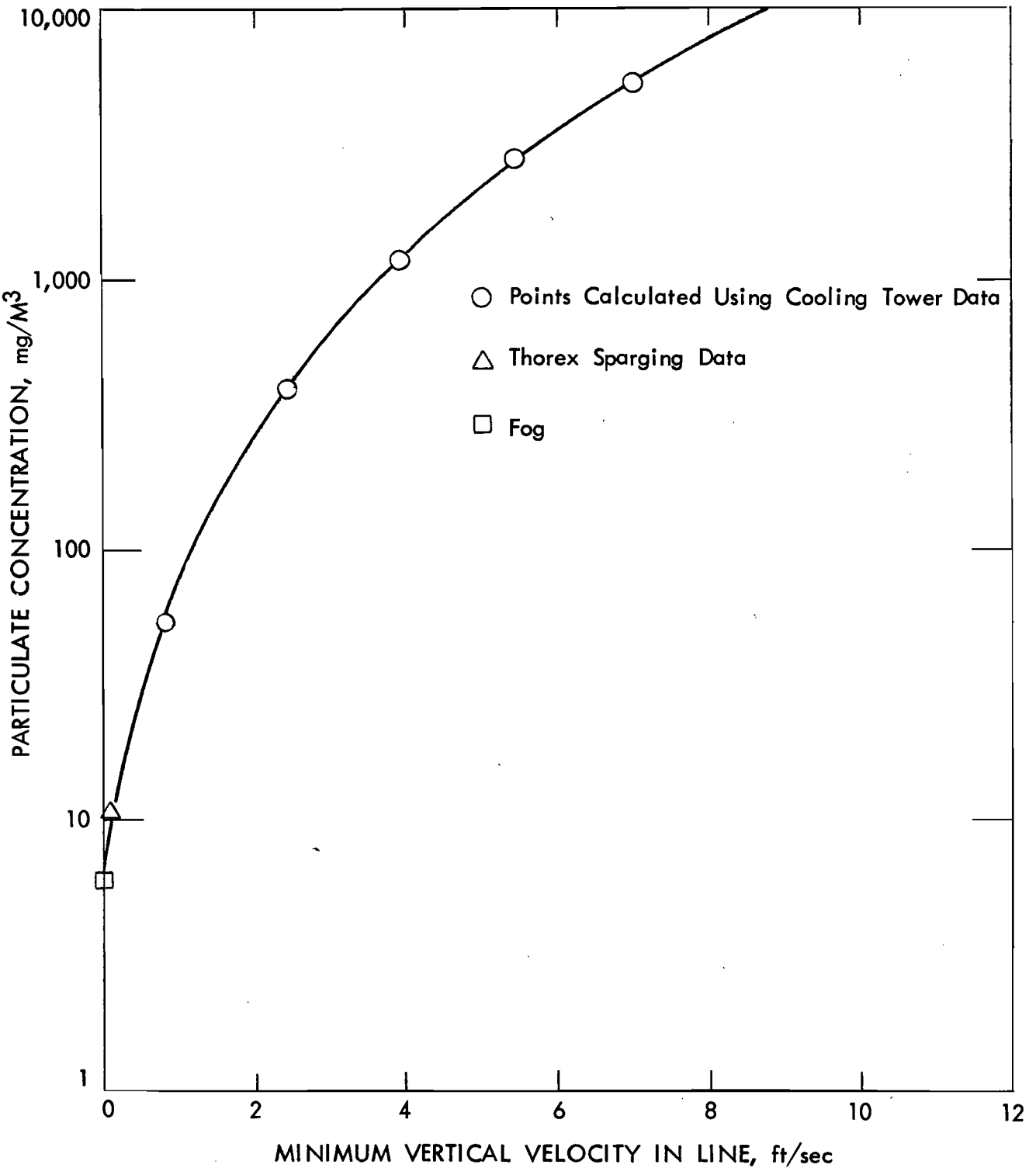

Fig. I-12. The effect of minimum superficial velocity in an off-gas line on the concentration of liquid solution particles resulting from very vigorous mixing of a solution with air. (Solution Density = 1 g/cc)

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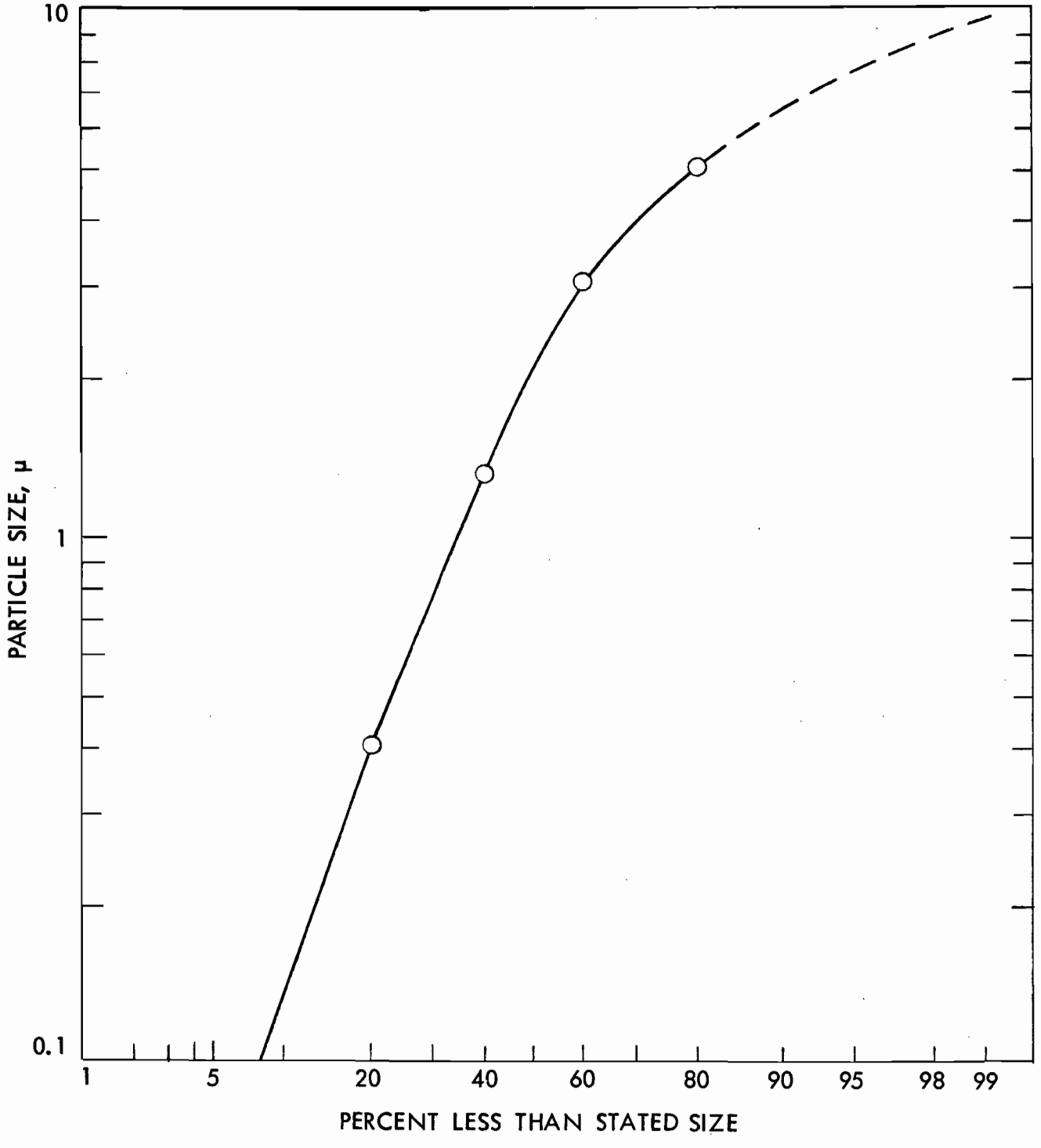

Fig. I-13. The particle size distribution of a stable aerosol which has encountered several changes of direction in a pipeline.

curies/ml, and a correction is made for the probable plutonium concentration, the gross β-γ mpc for occupational (40 hr/week) exposure is 6.6×10^{-9} μc/cc. The maximum ground concentration resulting from sparging and airlifting typical head-end solutions and from the maximum credible nuclear excursion in a process vessel have been estimated from this value of mpc and the aerosol data. The following assumptions were made to calculate the ground concentration resulting from sparging a vessel:

1. Air sparge rate = 1.3 scfm/ft^2 of vessel sectional area
2. Average annual weather conditions (dilution in 3039 stack and atmosphere) = 1.6×10^{-5} sec/m^3
3. PRFP vessel diameter = 5 ft

The air flow rate in the vessel off-gas from a sparger then would be

$$\frac{\pi(5)^2(1.3)}{(4)(60)(35)} = 1.2 \times 10^{-2} \text{ m}^3/\text{sec}$$

This flow rate, the solution activity (6.8×10^{-3} curie/mg), the particulate concentration (0.14 mg/m^3), and the stack dilution may be used to calculate the maximum average ground concentration:

$$(1.2 \times 10^{-2})(6.8 \times 10^{-3})(0.14)(1.6 \times 10^{-5}) = 1.83 \times 10^{-10} \ \mu\text{c/cc}$$

This maximum average ground concentration corresponds to 2.8% of the mpc. The percentage will vary in direct proportion to the number of spargers that are operating at any particular time. During normal plant operation it should not be necessary to use more than two spargers simultaneously.

The maximum average ground concentration resulting from the operation of an airlift was calculated by methods similar to those used for sparging. These calculations, in which average annual weather conditions and a conservative air flow per airlift, 1.5 scfm, were used, showed that each airlift will cause a maximum average ground concentration of 0.17% of the mpc. During normal head-end plant operation it should not be necessary to use more than two or three airlifts at the same time.

3.3.5 Cell Ventilation System

Ventilation air from the Bldg. 2527 cells will be passed through burst-resistant filters, combined with sufficient air to make a total of 13,000 cfm, and blown to the atmosphere through a 100-ft-high 3-ft-dia stack. The pre-filtered ventilation air will enter the cells through intake lines near the ceiling and exit through 30-in. ducts located near the floor. The air exit is baffled to prevent air from being splashed into the ducts. Approximately 0.5 in. H_2O negative pressure will be maintained in the cells by the exhaust fan. This negative pressure will cause a normal air flow of 1500 cfm per cell and an airflow of 10,000 cfm during the infrequent occasions when a cell roof plug is removed. All air that enters the cells will normally leave through the filters and stack, since the air intake lines will have check valves to prevent backflow and other cell openings will be tightly sealed to minimize leaks.

The filters are burst-resistant in the sense that an orifice will be provided in the ventilation line ahead of the filter to limit the air flow such that a pressure drop across the filters cannot reach the bursting pressure, 6-8 in. w.g. For example, if the orifice is designed for a pressure drop of 0.1 in. w.g. at the normal cell air flow of 1500 cfm and an incredible pressure of 20 in. of water suddenly developed in the cell, 18 in. w.g. pressure drop would occur at the orifice and the pressure at the filter would not exceed 2 in. w.g. This orifice would also provide a pressure drop of 4.4 in. w.g. for the 10,000 cfm that is required when one of the roof plugs is removed. This difficulty can be overcome by (1) providing a blower with sufficient capacity or (2) removing the orifice during shutdown when the cells do not contain appreciable activity and it is desired to open the roof plugs.

The only way whereby appreciable activity can get into the cell ventilation air is through the rupture of, or leak from, a vessel. An accident of this type is treated as a maximum credible accident in Sect. 4.5.

4.0 HAZARD DESCRIPTION

4.1 Radiation

The maximum activity of fuel to be handled in Bldg. 2527 is greater than has been routinely encountered previously, with the possible exception of some special purpose runs that were made in order to recover radiolanthanum (Ra-La) or Pa^{233} from short-decayed irradiated thorium (short decay Thorex). It is expected that most fuels will be processed only after ~180 days' decay. Presumably most fuels will not be shipped from the reactor site prior to 120 days' decay. The fuel assemblies from the Consolidated Edison Thorium Reactor are the most active of any scheduled for processing; the activity of a 250-kg batch (two assemblies) and the shielding thickness required to reduce the radiation field to 0.25 mr/hr are:

Activity per 250-kg batch	1.7×10^6 curies
Shield thickness for 0.25 mr/hr	
Canal (water)	12.5 ft
Dissolver	6 ft
Solvent extraction feed tank	5.3 ft

The canal and operating areas are therefore adequate for full-time occupancy and the roof area of the cells (with 5-ft shielding) is adequate in the worst case for limited occupancy. Under actual installation conditions, the activity level over the cells probably will not exceed 0.25 mr/hr; and under the worst conditions, the activity would be only 2.5 mr/hr at one or two spots.

Process solution samples will be taken by fully remote means, remotely loaded into a shielded carrier, and transported to the analytical facility. Most of the samples will be transported to the existing Bldg. 3019 analytical facility. A

maximum sample activity of 35 curies of gross beta-gamma with negligible alpha will occur when 5 ml of the most radioactive process solution is transported. The radiation dose rate at the surface of the sample carrier will be limited to 10 mr/hr.

Fuel assemblies will be loaded into the dissolver singly (in the case of large assemblies) or possibly as a cluster of assemblies held together by an expendable cage. They will be drawn vertically through a shielded tube in the canal loading area into a vertical charger located on a pad slightly above water level. The carrier will be shielded with 11 in. of lead, which will limit the radiation field at contact to 10 mr/hr. Dissolver closures and shielding will be so arranged that at no time will radiation escape directly from the dissolver or fuel assembly.

4.2 Criticality Considerations

This facility is intended for processing, under semiproduction conditions, fuels enriched to 20% U^{235}. Fuel types range from the ceramic oxide type (typified by CETR fuel made of ThO_2-UO_2) clad in stainless steel to the sodium-bonded uranium alloy type, through the aluminum-clad uranium-aluminum alloy class. Enrichments vary from the maximum stated above to depleted in some of the uranium metal fuels. The enrichment and batch size of one dissolver charge for various fuel assemblies is:

Reactor Fuel	Batch Size, kg		Enrichment, %
	Total	U-235	
CETR	250	22	9
Rural Cooperative	276	11	4.2
NMSR	222	10	4.6
FWC-EC	162	3	2
CPPD	212	7	3.2
PRDC Radial Blanket	252	nil	depleted
EGCR	200	5	2.5
Nondomestic Research	10	2	20

Undissolved fuel constitutes the greatest criticality hazard because in this condition high concentrations of fissionable material are readily attained. Special precautions are therefore required in the fuel storage canals, in the dissolver, and in any other pieces of equipment that may receive significant amounts of undissolved fuel. Homogeneous solutions encountered in processing of power reactor fuels can also reach criticality, but simple control measures, such as dilution, are usually applicable.

4.2.1 Nuclear Safety Control in Fuel Storage Canal

In the fuel storage canal individual fuel elements will be stored in criticality-safe racks which are one element wide and approximately 15 ft long. The racks will be made of heavy metal for structural integrity and will be separated from one another by at least 12 in. of water to prevent interaction. Fuel elements will be transported via a crane from a shipping carrier to the

racks or from the racks to the dissolvers, usually one element at a time but always in a batch such that it could not cause a nuclear reaction in the case of a crane or operator accident.

It is thought that procedures and interlocks that will be designed into the canal operations will preclude any nuclear reactions. Even if a nuclear reaction should occur, the gaseous fission products would probably not be released from the fuel, and the direct radiation dose rate at the surface of the canal would be even lower than the dose rate through 6 ft of concrete (see Sect. 4.2.2).

4.2.2 Nuclear Safety Control in the Process

The general plan for criticality control in the process is limitation of geometry in the dissolver, settling equipment, and other equipment pieces that will in the normal course of operation contain solid fissile matter; use of instrumentation, procedures, etc., to detect and/or prevent accumulation of significant amounts of fissile solids; and use of soluble nuclear poisons and concentration control for homogeneous solutions. Figure I-6 is a schematic representation of the proposed process equipment, which shows the geometrically safe dissolvers and solids separation equipment. Neutron, specific gravity, and radiation monitors that will be used to detect solids in the nongeometrically safe vessels which follow the dissolver are also shown. Other monitors will be used to ensure that the required amount of soluble neutron poisons are in solution. The decision to use a conditionally safe system was based on the premise that administrative procedures and instrumentation can decrease the probability of a nuclear incident to a very low value and that the facility is adequately designed to prevent excessive exposure of personnel to radiation and to prevent any significant spread of contamination to the environment if a nuclear reaction should occur. The nuclear safety procedures and equipment are described more completely elsewhere.*

Two rather difficult problems in criticality control result from the unusual nature of the fuel being processed. In the first case, the ceramic oxide fuels are subject to breakage and fracture due to thermal stress, mechanical shock, etc., and therefore some fraction of the fuel may be transferred from the dissolver to subsequent vessels with the various dissolvents. Since subcritical amounts of fuel might always be removed from the dissolver, progressive accumulations in subsequent vessels might cause a nuclear incident. Therefore precautions must be taken to prevent removal of undissolved fuel from the dissolver complex or to prevent accumulation of a critical mass in a subsequent vessel of unsafe geometry. Several means are available which can accomplish these results. In the second case, molybdenum contained in the CPPD and PRDC fuels precipitatesunder certain conditions when the fuel is dissolved in nitric acid. This presents a criticality problem due to the "carrying"

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*J. P. Nichols, "Proposed Methods for Maintaining Nuclear Safety in the Darex-Sulfex Head-End Process," ORNL-CF-59-10-40 (October 1959); A. R. Irvine et al., "A Review of Factors Bearing on the Nuclear Safety of Equipment for the PRFR Head-End Facility," ORNL-CF-60-1-102 (January 1960); A. R. Irvine, "Management Criticality Review, Power Reactor Fuel Processing," ORNL-CF-60-1-82 (January 1960).

capacity of the precipitate for a large fraction of the Pu^{4+}. It is therefore necessary to either positively prevent formation of a precipitate or to arrange to maintain any precipitate in a geometry or batch safe condition.

It is planned that detection and automatic poisoning systems will be included in the facility to signal evacuation orders and shut down any nuclear reaction. The detection system probably will consist of BF_3 counters strategically located in the cells and selectively shielded to permit adequate discrimination of neutrons from gamma radiation. The reaction shutdown system probably will consist of cadmium solutions stored under a high pressure, which will discharge into process vessels in the event that a nuclear reaction is detected.

Experience has shown that the most probable nuclear reaction in a Bldg. 2527 process vessel would consist of an initial and most powerful surge of approximately 10^{17} fissions. Such has been the case in approximately 15 prompt critical excursions which have occurred, several in heterogeneous systems, in critical assembly laboratories, and production plants. In solutions and water-moderated lattices the energy release in 10^{17} fissions is apparently sufficient to temporarily shut down the nuclear reaction due to thermal expansion and bubble formation and may allow time for the action of a permanent shutdown procedure to prevent recurrence of the reaction. If a shutdown device is not actuated after the initial surge, the reaction may tend to recur and gradually boil away the moderator or expel fissile material from the reaction vessel until permanent subcriticality is obtained. This type of accident is typified by the recent Idaho incident in which approximately 4×10^{19} fissions occurred in a period of approximately 7 min before permanent subcriticality was achieved.

It is expected that the most probable nuclear reaction, of the type described in the preceding paragraph, would not result in the rupture of a process vessel. In the previous accidents the individual excursions occurred in periods of approximately 0.1 sec or longer and did not result in vessel ruptures even though some of the vessels had restrictive vents. Excursion times of the order of 0.1 sec are relatively long by physical standards, and the energy of an excursion may be released without formation of a high-pressure blast or shock wave. The consequences of such a nuclear reaction that would be contained in a process vessel would be that plant operating personnel would receive direct exposure from prompt neutrons and gammas which passed through the cell wall, and activity would be released to the environment through the vessel off-gas system.

The integrated direct radiation dose through 6 ft of concrete that might be received by Bldg. 2527 personnel before they evacuate in the event of a nuclear reaction is:

Number of Fissions in Nuclear Reaction	Integrated Dose at Outside of Cell Wall, rem
10^{17}	0.0012
10^{18}	0.012
10^{19}	0.12
10^{20}	1.2

The dose rate that personnel would receive from the initial surge would probably be of the order of 1.2 mrem with credible exposures as high as 1.2 rem in the event that personnel did not evacuate the building and reaction shutdown mechanisms did not operate.

4.2.3 Activity Release from a Contained Nuclear Reaction

If a contained nuclear reaction occurred in a process vessel, activity would be released to the vessel off-gas system and thence to the environment. The activity would consist of fresh gaseous fission products and solution particles.

Particles of the process solution would be carried into the vessel off-gas system by the steam and small amount of gas generated in the nuclear reaction. Most of the steam would be condensed in nearby vessels, the vessel off-gas condenser, and in the VOG lines near the process building. The only activity that would escape to the 3039 stack cleaning system through the burst-resistant VOG filters would exist as extremely small solution particles entrained in any air that might be flowing through the line. The mpc of particulate activity resulting from any conceivable nuclear reaction would be equal to the mpc for normal PRFP solutions, since short-lived activity resulting from fresh fissions is biologically insignificant compared with activity from long-decayed power reactor fuel. It may be concluded that, since particulate activity from an accident would be carried through the filters only in a prevailing air stream and since any air flowing in the VOG system would already have the stable aerosol concentration and particle size distribution from sparging or airlifting, a nuclear excursion would probably not increase the activity of the filtered air, and thus the ground concentration, by any significant amount over that already present. The consequences due to the release of gaseous fission products that would be formed in the maximum contained nuclear accident may be calculated in a manner analogous to that described in Sect. 4.5 since the maximum downwind dose from a VOG release is directly proportional to the number of fissions that occur in the reaction.

It is very unlikely, though not completely inconceivable, that a nuclear reaction could occur which would rupture a process vessel and scatter its contents throughout a process cell. The consequences of such an accident, in addition to direct radiation exposure and release of activity to the vessel off-gas system, would be that activity could be released to the atmosphere through the cell ventilation system and cracks in the cell wall. This type of accident is treated as the maximum credible accident in Sect. 4.5.

4.3 Chemical Hazards

No unusual hazards due to chemicals are anticipated. Chemicals to be used are H_2SO_4, 60% HNO_3, aqueous HCl, 50% NaOH, Borax, $Al(NO_3)_3$, $Cd(NO_3)_2$, and NH_4F. Facilities for handling the above chemicals have not yet been designed; standard safety devices will be provided. The products of reaction of several of the chemicals with the fuel or its cladding can react violently with air (see below).

4.4 Fire and Explosion Hazards

Since Bldg. 2527 will be used only to prepare aqueous solutions of irradiated fuel and will not normally contain organic solvents, there will be relatively few fire and explosion hazards in the building. Exceptions are flammable gases generated in dissolution, radiolytic gas generated in radioactive solutions, and some decontaminating solutions.

Possible explosion hazards common to the dissolver off-gas and rare gas recovery system will include the explosion of hydrogen or ammonia caused by inadvertently leaking air into the system, the exothermic decomposition of NO_2 or N_2O, and the vaporization of a liquefied gas in a closed vessel (adequate safety valves will be provided to protect against this hazard). Air inleakage to the system filled with hydrogen or ammonia will be prevented by nitrogen purging before and during operation. In addition, continuous monitoring of the off-gases will warn of hazardous oxygen concentrations before the mixture reaches its explosive limits. During combustion of the hydrogen or ammonia and air, the mixture will be diluted with steam to well below its explosive limits.

Complete removal of the oxides of nitrogen upstream of the rare gas system will eliminate the hazards associated with these constituents. The NO_2 will be scrubbed out with caustic followed by a heated catalytic bed for the decomposition of the N_2O.

The maximum credible stable detonation pressure for a hydrogen-oxygen explosion has been determined to be about 30 atmospheres for a mixture initially at 1 atmosphere pressure. This would be comparable to a peak shock wave pressure from about 3 lb of TNT 3 ft from the center of the blast.* At 20 ft the peak shock wave pressure would be about 400 psf, which is less than the design pressure of 900 psf; so the explosion should be contained in the cell.

The maximum credible release of activity from the dissolver off-gas and rare gas recovery system would occur if a line ruptured due to a hydrogen-oxygen explosion during the decladding of CETR fuel. Such an accident would result in a maximum release of 190 curies of Kr^{85}, 10% of the Kr^{85} in the fuel, to the cell ventilation system before the decladding operation could be halted. The calculated effects of such an accident are given in Table I-5. Release of the rare gases to the atmosphere from the stack would not present grave decontamination problems, since the gases would be diluted by the prevailing wind and would not produce fallout.

Potentially explosive mixtures of hydrogen and oxygen are formed by the radiolytic decomposition of water in vessels that contain solutions of radioactive material. The yield of hydrogen in typical solutions that will occur in Bldg. 2527 will vary from 0.03 to 0.1 molecule per 100 ev of β-γ radiation

*J. P. Nichols, "The Effects of a Detonation in a Process Cell," ORNL-CF-59-11-115 (November 1959).

Table I-5. Effects of Accidental Release of Kr^{85} from Building 2527

Vessel Off-gas Release	
Total amount, curies	190
Max downwind integrated dose, rem	0.003
Distance downwind of max dose, m	580
Cell Off-gas Release	
Total amount, curies	190
Max downwind integrated dose, rem	0.019
Distance downwind of max dose, m	580
Release into Secondary Containment Zone	
Total amount, curies	0.021
Concentration, curies/m^3	1.9×10^{-6}
2-min-dose to building personnel, rem	<0.001
Release from Secondary Containment Zone (assuming ventilation system works)	
Total amount, curies	-
Max downwind integrated dose, rem	-
Distance downwind of max dose, m	150
Ground fallout at 20 m, curies/m^2	-
Release from Secondary Containment Zone (assuming ventilation system fails)	
Total amount, curies	-
Max downwind integrated dose, rem	-
Distance downwind of max dose, m	150
Ground fallout at 20 m, curies/m^2	-

depending on the solution nitrate concentration. It is planned that vessels that may generate potentially explosive amounts of radiolytic gas will be sparged with sufficient air to dilute the radiolytic gas below its minimum explosive limit. Radiolytic gas is also formed in a nuclear excursion with a yield of approximately 0.005 molecule per electron volt of energy evolved in the excursion. This is ordinarily an insignificant hazard, since the energy evolved in the excursion is approximately 100 times the energy that would be evolved if the resultant radiolytic gas burned to form water.

4.5 Maximum Credible Accident

The maximum credible accident that could occur in a Bldg. 2527 process cell is a nuclear reaction that completely shatters a process vessel containing a batch of irradiated CETR fuel solution and scatters its contents throughout the cell. This fuel solution, which could contain as much as 1.7×10^6 β-γ curies of fission products, is the most hazardous solution that will be handled in the plant, including solutions of α emitters. In the maximum credible single energy release, approximately 100 cu ft of gas and steam could be released to the cell in a small fraction of a second. This release could occur as a result of the maximum credible single nuclear burst, 10^{18} fissions, which is a factor of about 10 higher than what is considered to be the most realistic single burst. Approximately 30,000 Btu of energy and 2.0 std cu ft of radiolytic gas are released in 10^{18} fissions; approximately 100 cu ft of steam could be produced if it is assumed that 10% of the energy is used for the production of same. Since the Bldg. 2527 cells are designed to contain a shock wave equivalent to that from 3 lb of TNT (energy release = 5700 Btu with 50% appearing as a shock wave), they would certainly contain the nuclear burst since in this type of explosion a significantly smaller fraction of the total energy would appear as shock.

The effects of the maximum credible accident would be that the cell air pressure would be rapidly increased to approximately 4 in. w.g. positive pressure and a large fraction of the solution of fission products would be dispersed in the cell (and vessel) air as an aerosol. This air, which is laden with a very radioactive aerosol, could then be released to the environment through the vessel off-gas system, the cell ventilation system, and leaks in the cell wall and secondary containment shell. Fresh gaseous fission products from the reaction would be released in these same air streams. The calculated effects of an accident of this type in the worst conceivable vessel, the feed adjustment tank during CETR processing, are given in Table I-6.

The effects of a maximum credible noncriticality accident would be equivalent to the maximum credible criticality accident with the exception that there would be no direct radiation dose or release of gaseous fission products. The effects of this type accident are given in Table I-7.

Table I-6. Effects of a Nuclear Reaction of 10^{18} Fissions in Feed Adjustment Tank Followed by Rupture of Tank

	Aerosol Release	Gaseous Fission Product Release
Max downwind integrated dose from VOG release, rem	<0.001	0.154
Max downwind integrated dose from cell ventilation release, rem	0.006	0.650
2-min dose in secondary containment shell, rem	0.005	0.075
Max downwind integrated dose from building release, rem	<0.001	<0.001
Prompt neutron and gamma dose through shield, rem	0.012	

Table I-7. Effects of Accidental Release of Radioactive Material from Maximum Credible Noncriticality Accident in Building 2527

Vessel Off-gas Release	
Total amount, curies	0.082
Max downwind integrated dose, rem	<0.001
Distance downwind of max dose, m	580
Cell Off-gas Release	
Total amount, curies	0.41
Max downwind integrated dose, rem	0.0062
Distance downwind of max dose, m	580
Release into Secondary Containment Zone	
Total amount, curies	0.0032
Concentration, curies/m^3	3.0×10^{-7}
2-min-dose to building personnel, rem	0.0053
Release from Secondary Containment Zone (assuming ventilation system works)	
Total amount, curies	5.0×10^{-5}
Max downwind integrated dose, rem	<0.001
Distance downwind of max dose, m	150
Ground fallout at 20 m, curies/m^2	1.8×10^{-10}
Release from Secondary Containment Zone (assuming ventilation system fails)	
Total amount, curies	1.22×10^{-3}
Max downwind integrated dose, rem	<0.001
Distance downwind of max dose, m	150
Ground fallout at 20 m, curies/m^2	4.5×10^{-9}

5.0 OPERATING AND EMERGENCY PROCEDURES

Because Bldg. 2527 is a proposed facility rather than an operating system, detailed procedures for operations and for emergencies have not been prepared. Detailed operating procedures will be prepared for the process system during the final stages of design and construction and will specify the step-by-step operations to be performed. Detailed emergency procedures have not been formally prepared. Some of the possible emergencies and appropriate procedures, many of which are carried out by automatic controllers, are:

Emergency	Means of Detection	Procedure
Nuclear chain reaction in cell	Neutron monitor	1. Add nuclear poison to all probable points of reaction 2. Evacuate building 3. Determine source of reaction by observation of pressure charts, samples, etc., and proceed only with management approval
Explosion causing rupture of vessel	Cell pressure	1. Close air inlet to cells 2. Close air entry points to secondary containment areas and maintain these areas under 0.3 in. H_2O negative pressure 3. Determine cause of explosion and proceed only with management approval
Fire in cells	Cell temperature indicator	1. Close air inlet to cell 2. Admit water to cell via decontamination sprays 3. As last resort, CO_2 could be admitted via the cell air entry system
Nuclear chain reaction in canal	Neutron monitor	1. Evacuate building 2. If reaction blows assembly apart, determine cause and further separate fuel assemblies 3. If reaction continues, determine gamma and neutron field; if satisfactorily low, mechanically place cadmium strips in vicinity of reacting assemblies; if not low enough, add massive amounts of $Cd(NO_3)_2$ to canal

Part II. BUILDING 2527, VERY HIGH ACTIVITY WASTE STORAGE FACILITY

3.0 SYSTEM DESCRIPTION

Three 15,000-gal waste tanks and a 250-gal diversion tank are to be housed in a concrete vault located partially below grade near proposed Bldg. 2527. The storage tanks will be in a common room 26 x 36 x 13 ft high, which has a stainless steel pan on the floor and reaching 2 ft up the walls. A diversion tank will be in a separate cuibcle, ~8 x 10 x 11 ft high, fitted for contact maintenance procedures. The diversion tank will serve as a "switching station" for all solution transfers; that is, any time a solution transfer is made, into or out of the system or between tanks, the solution will pass through the diversion tank. The shielding walls will be 6 ft thick and the roof 5 ft of normal concrete. The vault is to be made of reinforced concrete and will be permanently sealed with the exception of the diversion tank cubicle, the corrosion specimen removal ports, and an air entry port. The enclosure is designed to contain a shock wave having a pressure of 900 lb/sq ft and energy equal to 230 ft-lb/sq ft. A slight negative pressure is to be maintained in the enclosure, although no ventilation will be needed under normal operating conditions. Ventilation air will pass through the filtration system provided for Bldg. 2527, consisting of one roughing filter (80% NBS) and two absolute filters (99.95% DOP) in series. Vessel off-gas flow will be monitored for volume and hydrogen content and will be filtered in the vessel off-gas system serving Bldg. 2527 prior to going to the 3039 stack for additional treatment and dispersal. Figure II-1 shows the general layout of the storage system and its components.

These tanks are to contain acidic wastes. Fission product decay heat is to be removed by means of cooling coils installed in a minimum of three parallel banks, each of which is capable of removing 770,000 Btu/hr when the tank contents are at 160°F. As heat removal capacity is a direct function of the temperature difference between coolant and the tank contents, each coil will remove ~1.5×10^6 Btu/hr when the wastes are at the boiling point. In the event of coil failure, the defective coil can be detected and isolated. Contamination of the environment via a coil failure is prevented by use of an enclosed coolant circuit, which is in turn cooled by cooling tower water in a water-to-water heat exchanger. A 125 sq ft condenser is provided for the off-gas from all three tanks. This condenser is adequate to condense the steam from any single tank if all its cooling coils should fail simultaneously. The major features of the waste tanks are enumerated in Table II-1.

It is planned that sufficient tank capacity will be held in reserve to permit movement of solution from a leaking tank to an adjacent one. Duplicate routes are available to perform any necessary service.

4.0 HAZARDS

The hazards in storage of waste derive primarily from the fact that large quantities of highly active wastes are retained over an indefinite period. Volumes and activity level of the more active wastes to be stored in these tanks are:

Source of Waste	Volume, gal per metric ton of fuel	Specific Activity, β-γ curies/gal	Heat Generation, Btu/gal
Declad CETR	105	37,000	600
Declad NMSR	52	25,000	420
Darex dissolved NMSR	1220	1,100	18

The primary concern is with containment. For the systems currently under consideration, the corrosion rate is low (⩽ 2 mils/year) with a minimum corrosion allowance of 66 mils. Corrosion specimens will be installed and inspected at intervals. If a coil fails, only part of the tank's cooling capacity is lost. If a leak develops in a tank, means are available under the worst conditions to transfer solutions to another vessel and to pick up the leaked solution as well. Duplicate means of transfer are available for all process streams.

The probability of chemical explosion or fire is limited to problems with radiolytic hydrogen. The amount of hydrogen liberated per unit of decay heat decreases as the nitrate concentration in the solution increases and is therefore expected to be quite low. Airlift type circulators are to be used to enhance heat transfer and the air sweep provided will prevent accumulation of hydrogen. Flow alarms will be installed on each air inlet stream as well as on the combined vessel off-gas flow. In addition, a hydrogen monitor will show the hydrogen concentration in the effluent air, thereby providing another safeguard. The tanks will each be electrically grounded, and a flame arrestor will be provided on the vessel off-gas line.

A hydrogen explosion under the worst conditions would be equivalent to ~20 pounds of TNT, which no doubt would rupture the storage tank and breach the shielding walls. Explosions of hydrogen-oxygen mixtures can be prevented by dilution and by exclusion of sparks. Dilution air will be supplied to each tank by means of air spargers. The flow to each tank will be monitored and provided with an alarm; in addition, a flow alarm on the vessel off-gas will provide a check on the total air flow to the system. A hydrogen monitor-alarm system will provide additional assurance of adequate dilution air. Sparks will be eliminated from the system by electrical grounding and the use of flame arrestors in the off-gas line.

Criticality is not a significant hazard in this system. Normal operation will cause ~4 kg of fissile matter to be discharged to each 15,000-gal tank, which would be a concentration of ~0.08 g/liter. A concentration of > 10 g of fissile matter per liter is required for criticality. Stringent measures are taken in the solvent extraction complex to prevent entry of fissile matter into the waste system in order to prevent losses and to ensure nuclear safety in that plant. In addition to careful operation of the extraction columns, wastes are sampled and analyzed prior to transfer to the waste tanks.

5.0 OPERATING AND EMERGENCY PROCEDURES

Because this is a proposed facility, detailed procedures have not been prepared. Instrument panel boards and alarms will be installed both locally and in the control rooms of Bldg. 2527. Normal operating procedures would provide for routine surveillance of all instruments and for periodic removal of corrosion

specimens to determine the rate of attack on the tank. One tank will be held in reserve at all times.

The following table lists some emergencies that might arise and the appropriate remedial procedure:

Emergency	Remedial Measure
Leak in cooling coil	Pressure-test each coil and install blinds to isolate defective coil
Failure of all coils in a single tank	Turn cooling water to jet suction coolers and transfer tank contents to spare tank
Leak in one tank	Transfer contents of tank to spare tank; rinse floor of vault with water and transfer to spare tank (Note: floor drains empty into diversion tank)
Failure of electrical supply	Use emergency electrical supply; if this does not work, use process water routed directly through coils; monitor effluent on emergency basis
Failure of cooling water system	Use process water routed directly through coils; monitor effluent on emergency basis

Table II-1. Specifications of High-activity Waste Tanks

Capacity, gal, each	15,000
Material of construction	304L stainless steel
Operating temperature, maximum, ^{o}F	160
Number of parallel pipe coils	3
Pipe size, in.	2
Pipe schedule	40
Cooling water temperature, ^{o}F	
Inlet	100
Outlet	110
Heat transfer coëfficient, Btu/hr.sq ft.^{o}F	40
Heat transfer area per coil, sq ft	350
Heat removal capacity per coil, Btu/hr	770,000
Condenser surface area, sq ft	125
Condenser heat removal capacity, Btu/hr	2.5×10^{6}
Corrosion rates, ipy at 176°F in 2 M HNO_3—1.82 M stainless steel	0.0023
Corrosion allowance, in.	0.066
Peak hydrogen generation per tank, scfh	4
Air purge rate, scfm	5

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