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HAZARDS EVALUATION

WASTE FRACTIONIZATION - B PLANT (PROJECT CAC-144)

B-PLANT MODIFICATIONS FOR FPCE-WASTE MANAGEMENT INTEGRATED FACILITIES  
(PROJECT CAC-181) AND PROMETHIUM PURIFICATION - B PLANT (PROJECT AAE-207)

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HAZARDS EVALUATION  
WASTE FRACTIONIZATION - B PLANT (PROJECT CAC-144)  
B-PLANT MODIFICATIONS FOR FPCE-WASTE MANAGEMENT INTEGRATED FACILITIES  
(PROJECT CAC-181) AND PROMETHIUM PURIFICATION - B PLANT (PROJECT AAE-207)

I. INTRODUCTION

This report presents the findings of a hazards review team for the Phase III portion of the Waste Fractionization B Plant. The specific emphasis of this report is on Phase III installation of facilities for fractionization of waste, plus integration of equipment and facilities from Phases I and II into an operating complex. Previous hazards evaluations have been made; first, of the separations area<sup>(1)</sup> in general, and second, with Phases I and II of the reactivation of the B-Plant complex.<sup>(2,3)</sup> Phase I, Fission Products Storage, was concerned with the restoration of services to B Plant and accumulation of a fission product inventory.<sup>(4,5,6)</sup> Phase II was concerned with the installation of facilities in B Plant to demonstrate a process system for packaging the long-lived fission products as a small volume concentrated waste and included additional ventilation capability.<sup>(7)</sup> The purpose of Phase III is to provide waste fractionization facilities in the B Plant for processing high level wastes from Purex and Tank Farms into fractions that can in the future be immobilized and contained more safely.<sup>(8)</sup> The B-Plant modifications for FPCE-Waste Management Integration Facilities, CAC-181, were cancelled or transferred to Project CAC-144, because the FPCE Plant project was terminated. This report covers the modified facilities as presented in Revision 4 of Project CAC-181,<sup>(9)</sup> and Revision 2 of Project CAC-144<sup>(10)</sup> and Project AAE-207, Promethium Purification.<sup>(11)</sup> Project AAE-207 provides equipment to permit the processing and purification of promethium which had been accumulated during Phase I operation in B Plant.

The waste management program has included the recovery of radioactive strontium, cesium, promethium, and cerium in portions of this and other facilities. The history of some of the work done at B Plant, the Purex Building, and the Semiworks is included to indicate that the recovery processes have been tested or used for some time and the experience gained provides a firm foundation upon which the plant processes are designed.

II. SUMMARY AND CONCLUSIONS

This report presents the findings of the hazards review team, assembled to review facilities and operating plans and to identify and evaluate potential hazards for Phase III of the Hanford Waste Management Program. Phase III of the Waste Management Program comprises the installation in B Plant of equipment for: (1) the separation of long-lived radionuclides, strontium-90 and cesium-137, from currently generated and aged high-level wastes (termed "waste fractionization"), and, (2) the storage of concentrated solutions

of strontium-90 and cesium-137, pending the choice of a process to immobilize these nuclides for long-term storage. Completion of Phase III also marks the integration of several waste management projects into an operating complex with B Plant becoming a major chemical processing facility.

In making this analysis, the hazards review team has evaluated the following:

- All engineering flow diagrams and certain of the detail design drawings.
- The technical specifications and process flowsheets.
- Physical installation of the facilities, and the operating plans.
- A large volume of reference material providing history, reported operating experience, and technical information pertinent to the hazards analysis.

A variety of potentially hazardous incidents were postulated. These incidents, their possible and probable effects, and means of preventing their occurrence are discussed in this report.

The conclusions of the hazards review team are as follows:

- The facility will contain unprecedented, large quantities and high concentrations of the long-lived radionuclides strontium-90 and cesium-137; up to 85 megacuries of strontium-90 and 25 megacuries of cesium-137 can be stored in present facilities. Individual tanks will contain up to 35 megacuries of strontium-90 or cesium-137 at concentrations up to 10,000 curies per gallon. The basic design philosophy employed in B Plant is consistent with that of other Hanford chemical processing facilities and is expected to result in ample protection of personnel and the public from the hazards associated with radiochemical processing.
- It is theoretically possible to release up to two (2) megacuries of strontium-90 to the steam condensate crib from the failure of the heat exchange tube bundle in the strontium concentrator. There are engineering and operating safeguards to prevent a release of this magnitude and the postulated probable effect from such an occurrence is the release of 2 kilocuries of strontium-90 to the steam condensate crib. Engineering studies are currently being made of the feasibility of modifying the steam condensate system to provide additional safeguards which would further reduce the effects of such an incident.
- The concentrations of radionuclides in the liquid effluents from the plant which are discharged to the environs will not meet the standards of maximum permissible concentrations (MPC) at the point

and in the ground water beneath the disposal site!  
see 0510 II, E. 3.(c).

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discharge as stated in AEC Manual Chapter 0524. However, the disposal methods for these effluents are considered adequate to control the concentrations of radionuclides well below MPC levels at the boundary of the Hanford Reservation.

- The most severe effects of the various postulated incidents would result from a fire in the 291-B Building which houses the absolute filters in the exhaust ventilation system. It has been estimated that such an incident could possibly release about 1,300 curies, including 200 curies of strontium-90. The maximum probable quantity of strontium-90 released from such an incident is estimated to be about 40 curies, including 60 millicuries of strontium-90. As a result of this incident, it is estimated that: (1) the exposure to personnel on the Hanford Reservation would not exceed the yearly limits for occupational employees; and (2) the exposure to anyone off the Hanford Reservation would not exceed the limits for non-occupational personnel.
- There are some B Plant processes which present potentially hazardous conditions beyond those currently experienced, e.g. head-end treatment of current Purex waste resulting in a heat-generating solids layer in TK-11-2 during one portion of the cycle and cesium ion exchange carried out in a pressurized canyon vessel, T-18-2. There are engineering and operating safeguards provided to make such incidents highly unlikely events.
- Criticality safety is not a problem in the operation of B Plant. Mass limits for plutonium have been established for process vessels and procedures will be implemented to insure that plutonium does not accumulate in excess of these safe limits.

### III. PROCESSING EXPERIENCE

In 1955 a program was started at Hanford to develop the methods for economic processing of fission products. The first order for fission products was placed in the summer of 1960 and called for one megacuries of strontium-90 to be used as a heat source for generating electrical energy. In April, 1961, the first shipment of 0.02 megacuries of strontium-90 was made to the Oak Ridge National Laboratories. At the same time, an order for 0.1 megacuries of cesium-137 for teletherapy uses was filled. Since the beginning of this program, a total of 19.0 megacuries of fission products has been recovered, separated, packaged, and delivered either to the AEC for safe off-site shipment or to the Battelle-Northwest Laboratories. The following table shows the quantities of the various products:

Strontium-90	7.9 megacuries
Cesium-137	4.5 megacuries
Promethium-147	4.1 megacuries
Cerium-144	2.5 megacuries

Total 19.0 megacuries

In addition to these shipments, 28.3 megacuries of fission products have been recovered from the wastes and stored in vessels in B Plant, 244-CR Vault, and Semiworks in anticipation of the Waste Fractionization Plant (B Plant) startup. Not only did this program produce useful radioactive isotopes for other government-sponsored programs, it also provided experience, refinement, and demonstration for the B-Plant Phase III flowsheets, which has significantly raised the confidence level for Phase III operations.

#### A. Purex Head-End

The Purex Head-End equipment, which was no longer required for the separation processing of plutonium and uranium, was used for recovering the strontium and rare-earth fractions in the Purex acid waste stream by a lead-carrier, sulfate-precipitation process. A liquid-solid centrifugation accomplished the phase separation of precipitated lead, strontium, cerium, and rare earths from the bulk of the residual fission products and salts that remained in solution. The collected precipitate was metathesized to convert it to the carbonate form, which was then dissolved in dilute nitric acid. The aqueous nitric acid fission product stream was pumped to the 244-CR Storage Vault by underground pipeline. This Purex Head-End process was used until February, 1967. More than 40 megacuries of fission products were recovered in the six years of its operation. Very similar processes using centrifugation and precipitation equipment provided in B Plant in Cells 11, 12, 31, and 32 will be continued as a part of the Phase III operation.

#### B. B Plant (Phase I)

In 1963, the Phase I - Waste Management Program Project CGC-897 was completed, equipping eight cells of the idle B-Plant facility with process equipment to permit the segregation and storage of unrefined solutions of strontium-90 and rare earths (including promethium-147). The Phase I cells in B Plant were used for this purpose on an intermittent basis for three years. In June, 1966, processing was discontinued to make way for Phase III construction.

In the Phase I operation, fission product concentrate from Purex Head-End was pumped to B Plant by underground pipeline where the strontium was separated from the lead, cerium, and rare earths by an acid-side, oxalate-precipitation process. A centrifuge was used to separate the phases. The lead, cerium, and rare-earth fractions were dissolved in nitric acid and stored. The strontium fraction was thermally concentrated and stored. During the Phase I processing campaign, about 12 megacuries of strontium-90 and more than 18 megacuries of two-year aged promethium-147 in the rare-earth fraction were separated.

Portions of the strontium and rare earths produced in Phase I were pumped by underground pipeline to the Semiworks for purification of the strontium-90 fraction and separation of the rare-earth fraction

into cerium-144 and a rare-earth fraction including promethium-147. Quantities listed in the following table, which were not required to fill shipping requirements, have been stored for Phase III processing. The inventories of promethium-147 in tanks 8-2, 9-1, and 12-1 are at two-year's equivalent age which is attained in October, 1968.

TABLE  
CURRENT INVENTORY OF STRONTIUM AND PROMETHIUM

(September 1, 1967)

<u>Location</u>	<u>Vessel</u>	<u>Product</u>	<u>Megacuries</u>		
			<u>Sr-90</u>	<u>Pm-147</u>	<u>Ce-144</u>
Semiworks	E-4	Purified Concentrate	0.9		
244-CR Vault	002	Separated Crude	3.6		
B Plant	6-1	Promethium, Rare-Earth Crude Fraction		5.1	10.9
	6-2	Promethium, Rare-Earth Crude Fraction		5.2	9.6
	7-1	Promethium, Rare-Earth Crude Fraction		2.4	3.0
	8-1	Promethium, Rare-Earth Crude Fraction		2.8	3.5
	8-2	Head-End Crude Product	0.6	1.5	7.9
	B Plant	9-1	Head-End Crude Product	0.9	2.3
12-1		Head-End Crude Product	0.8	2.2	11.5
Total			6.8	21.5	58.3

### C. Semiworks

The Semiworks, initially built to pilot the Redox process and later converted to pilot the Purex process, was reactivated. Equipment was provided for processing and for loadout of fission products, and the Semiworks has been used in the dual capacity of production and process demonstration up to the present time. The initial processing of strontium-90 used the Purex Head-End product which was pumped to the Semiworks from the 244-CR Vault. The rare-earth fractions were not recovered and were returned to the underground waste storage tanks. Later, when Phase I came into operation, rare earths were prepared in the Semiworks for shipment.

All 18 megacuries of fission products produced for delivery were processed through the Semiworks facility with the exception of a very small amount of strontium-90 that was processed and loaded out in the Hanford Laboratories at the 300 Area in the early part of 1961. The final strontium purification was performed with a solvent extraction process in two pulse columns similar to those in the Purex Plant but employing the organic extractant, di(2-ethylhexyl)phosphoric acid, commonly abbreviated D2EHPA. The transfer mechanism between a cation

and D2EHPA involves a liquid-liquid ion exchange process. Strontium, calcium, and a small amount of cerium were extracted into the solvent in the first extraction column; thus, effectively separating strontium from the majority of the metal-ion and fission - product contaminants. The strontium was partitioned from calcium and the remainder of the cerium in a stripping column. The cerium and calcium were then stripped from the solvent in a batch contactor. Periodically, the purified strontium-90 product was loaded out as a carbonate precipitate into a filter cask for shipment. For the rare-earth feed, a similar solvent extraction process using the same equipment in the Semiworks was used to prepare the promethium-147 and cerium-144 products for shipment.

The solvent extraction process and equipment being provided for Phase III operation in B Plant are similar to those which have been used for production in the Semiworks the past several years, and Semiworks processing experience will benefit the Phase III operations. In addition to production, the Semiworks was used for testing various solvent extraction process flowsheets planned for use in B Plant. The Phase III equipment is larger in capacity, capable of higher throughput rates, arranged in series to perform several functions on a continuing basis, designed to produce a better quality product, and flexible for processing a variety of waste streams.

#### D. Cesium Loadout

A program to supply the required quantities of cesium-137 was initiated in late 1960. A small loadout station was built near the 241-C tank farm. It contains a space for positioning a trailer holding a shielded transfer tank (STT cask). The loadout process is relatively simple. Aged supernate solution is filtered, cooled, and passed through the cask which contains an alumino-silicate, ion-exchange bed. Cesium is selectively adsorbed by this medium and cesium-depleted supernate is returned to waste storage. When the STT cask ion exchange bed is loaded to capacity, the bed is water-washed and the cask is prepared to be shipped off-site. Since 1961, 72 STT casks containing a total of 4.5 megacuries of cesium-137 have been loaded in this facility and shipped to ORNL. With B-Plant start-up, removal of cesium from the waste stream will be performed in B Plant using a similar ion exchange bed. The aged alkaline supernate from the Purex and possibly Redox tank farms will be transferred to B Plant. The supernate will be pumped through the zeolite bed where cesium will be removed by adsorption on the zeolite. The cesium will be removed from the bed with an eluent solution, thermally concentrated, and held in storage for future processing, encapsulation, or shipment.

The Waste Fractionization Plant (Phase III) processes have been demonstrated in plant or pilot-plant scale systems. These processes are now being integrated for continuous operation in a scaled-up, more efficient and flexible plant.

IV. THE OPERATING PLANTA. Site and Facilities

Since Hanford has long been a center for nuclear facilities, a great deal of investigation and analysis of the environment at the site has been performed. This information will be summarized as it affects the B Plant. There are several site characteristics that make the location of B Plant highly desirable. These include a flat terrain, a low population and agriculture density in the surrounding area, semiarid and moderate climate with a well-understood meteorology, a low water table, a low soil permeability, no potable water supplies or streams within miles, and no major local earth shocks noted in the past 50 years.

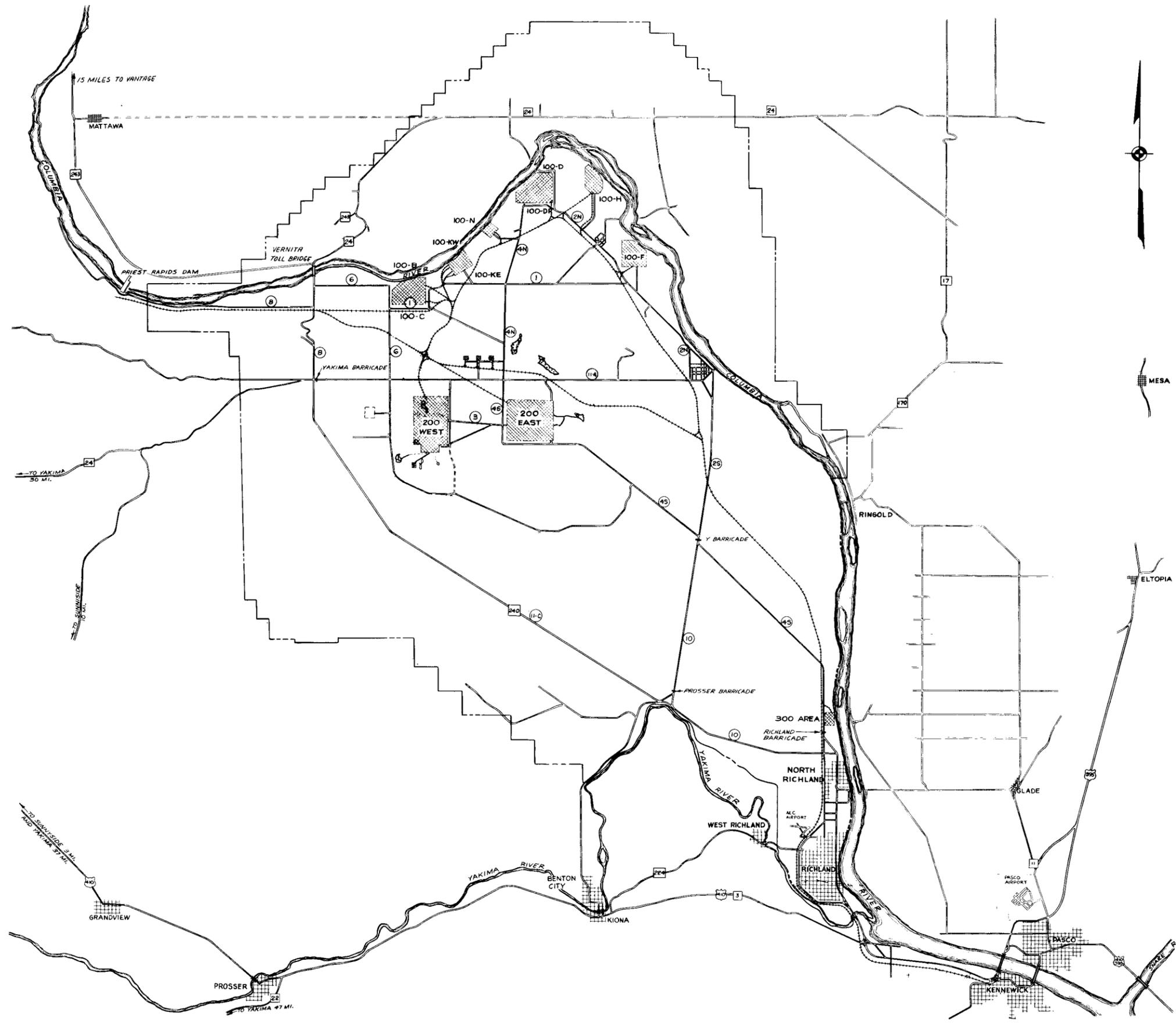
B Plant is located in the central part of Hanford in an area known as the 200-East Chemical Separations Area, See Figure IV-A-1. The Columbia River, at its closest point, is about eight miles from the site. The location of B Plant in the approximate center of the Hanford Reservation, remote from populated and agricultural areas, minimizes, even in the event of an accidental release, the effects on the general public.

The Hanford site has no residential population, and the maximum weekday work force is approximately 7,000 persons. The daily work force in the 200-East Area itself totals 700 persons, including those employed in B Plant. The surrounding area is sparsely populated, with four small cities within 50 miles from the site. Should the population double in the next 20-25 years, as anticipated, the population density would still be described as sparse.

The Hanford region meteorology features frequent strong inversions occurring at night breaking during the day to provide unstable conditions. (12) Near B Plant, the prevailing winds are from the northwest with strong drainage and cross winds causing distorted flow patterns. The probability of occurrence of stable conditions is about the same for all wind speeds. The power house and administration area southeast of B Plant is downwind approximately 32 percent of the time. The 202-A Building is downwind about 20 percent of the time with the Hot Semiworks, located east of B Plant downwind seven percent of the time.

A thorough study of all meteorological data (wind directions, wind speeds, precipitation rates, dispersion parameters, etc.) indicates that there are no unusual meteorological features of the site which would escalate the consequences of an accidental radioactive release.

The pattern of earth deposits underlying the B-Plant site is remarkably consistent with those minor variations that do occur not being of a nature to seriously affect the safety of the facility. In the 200-East Area, the ground surface is approximately 680 feet above sea level, and the underlying gravels are about 280 feet thick. The water table is found near the base of the gravel and flows toward the Columbia River. Perme-



- LEGEND**
- ROADS
  - RAILROADS
  - ② HANFORD WORKS ROAD NUMBER
  - Ⓢ U.S. HIGHWAY NUMBER
  - Ⓠ STATE HIGHWAY NUMBER
  - GRAVEL ROADS



Figure IV-A-1  
MAP OF HANFORD AREA  
MAP HANFORD PROJECT  
SCALE 1" = 6 MILES

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ability of the Pliocene gravels is low, with test values being as low as 0.005 feet per minute. The low permeability increases the time for radioactive decay before any radioactivity-bearing waste fluid reaches the water table. The low permeability in conjunction with the high specific retention of six to ten percent for liquids and the extreme thickness (100 feet plus) of undersaturated soil allows large quantities of fluid waste to be retained in the subterranean soil in the event of an accidental discharge.

Rainfall at Hanford is approximately 6 to 7 inches per year. Precipitation is taken into the ground by percolation, and subsequently released to the atmosphere by evaporation; there is no net release to the water table from rainfall.

The water table at the 200-East Area is over 200 feet below the surface. The ground water flow, monitored by wells, is toward the Columbia River. Drinking water for B Plant is obtained from the existing sanitary systems serving the 200-East Area. This water is drawn from the Columbia River at a production reactor site and is filtered and chlorinated at the 200 East power house.

The Hanford Area is classified as Zone II in the Uniform Building Code. In the half century 1915-1965, seven major shocks have been recorded in the Tri-City Area with epicenters 100 miles or more away. No major shocks have been recorded since 1965. The character of the basalt flows, together with well-developed joint systems, permits stresses to be continuously relieved. The possibility of a shock originating under B Plant is considered to be remote. A maximum shock of MM5.5 on the Richter scale may be possible. (13)

The B Plant, Building 221-B, is a multi-storied, predominantly reinforced concrete structure, approximately 810 feet in length. The building has two major portions: the process portion which contains the radioactive ("hot") equipment and the regulated zones, and the service portion which houses personnel and equipment necessary for remote operation of the process portion. As shown in Figure IV-A-2, the building consists of operating, pipe, and electrical galleries, process cells, pipe trench, air vent tunnel, and cell drain line, all of which extend the full length of the building. The building contains 40 cells in the canyon, most of which are being reequipped with new and used equipment for the fractionization operations. The cell equipment and their functions are illustrated in Figure IV-A-3. The bulk of the equipment, consisting of processing tanks and extraction columns, was taken from deactivated processing plants. Existing centrifuge sections have been renovated for head-end current and stored acidified waste treatment. The new extraction battery consists of four, pulsed, nozzle-plate columns, 15 to 20 inches in diameter and 14.5 to 18.5 feet high. The ion exchange column is cylindrical, 15 feet high and 6 feet in diameter, and contains a bed of approximately 2000 gallons of zeolite for supernate processing. Tanks are provided for the accumulation of cesium-137 and strontium-90.

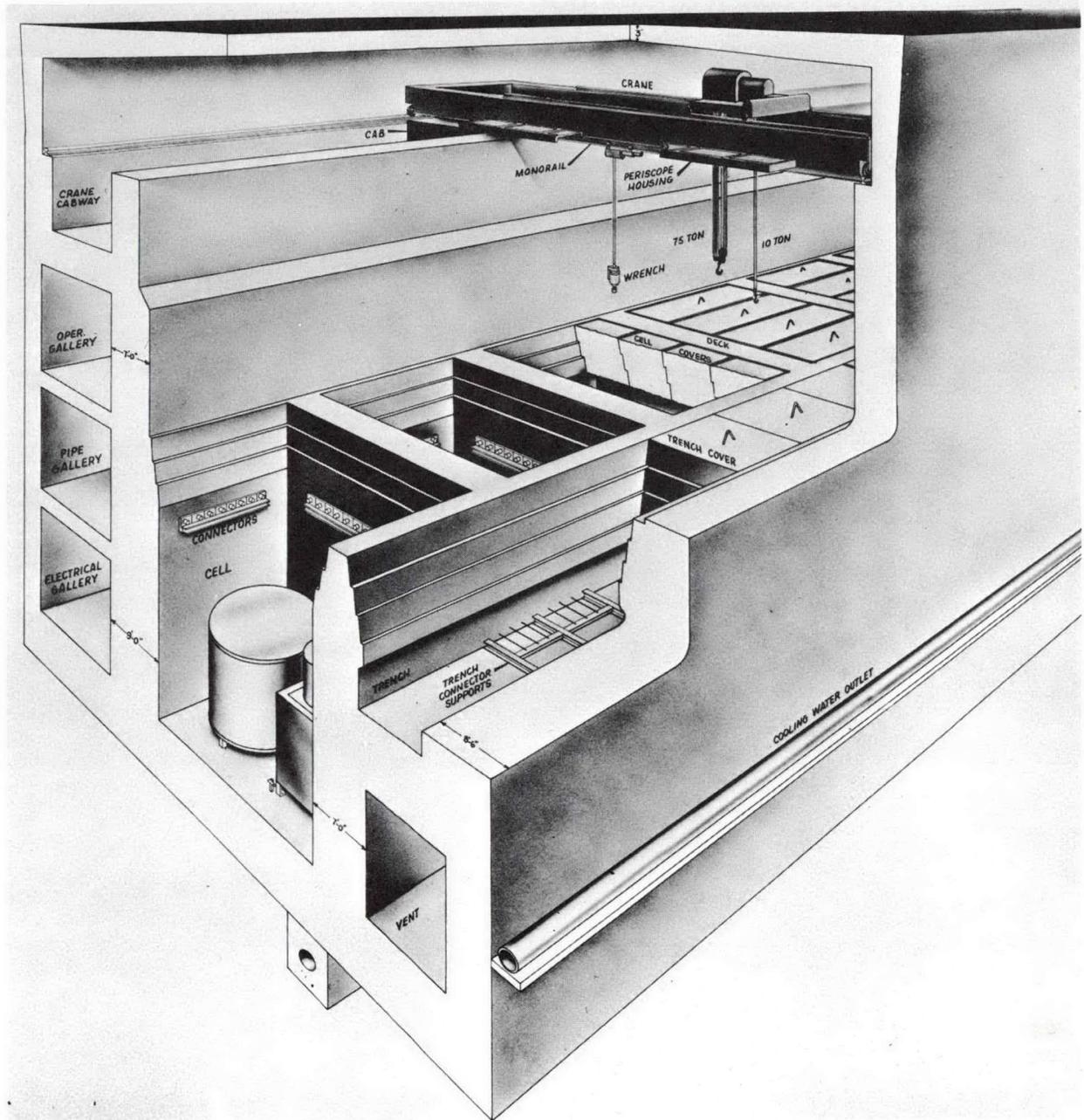


Figure IV-A-2

Sectional View  
221-B Building

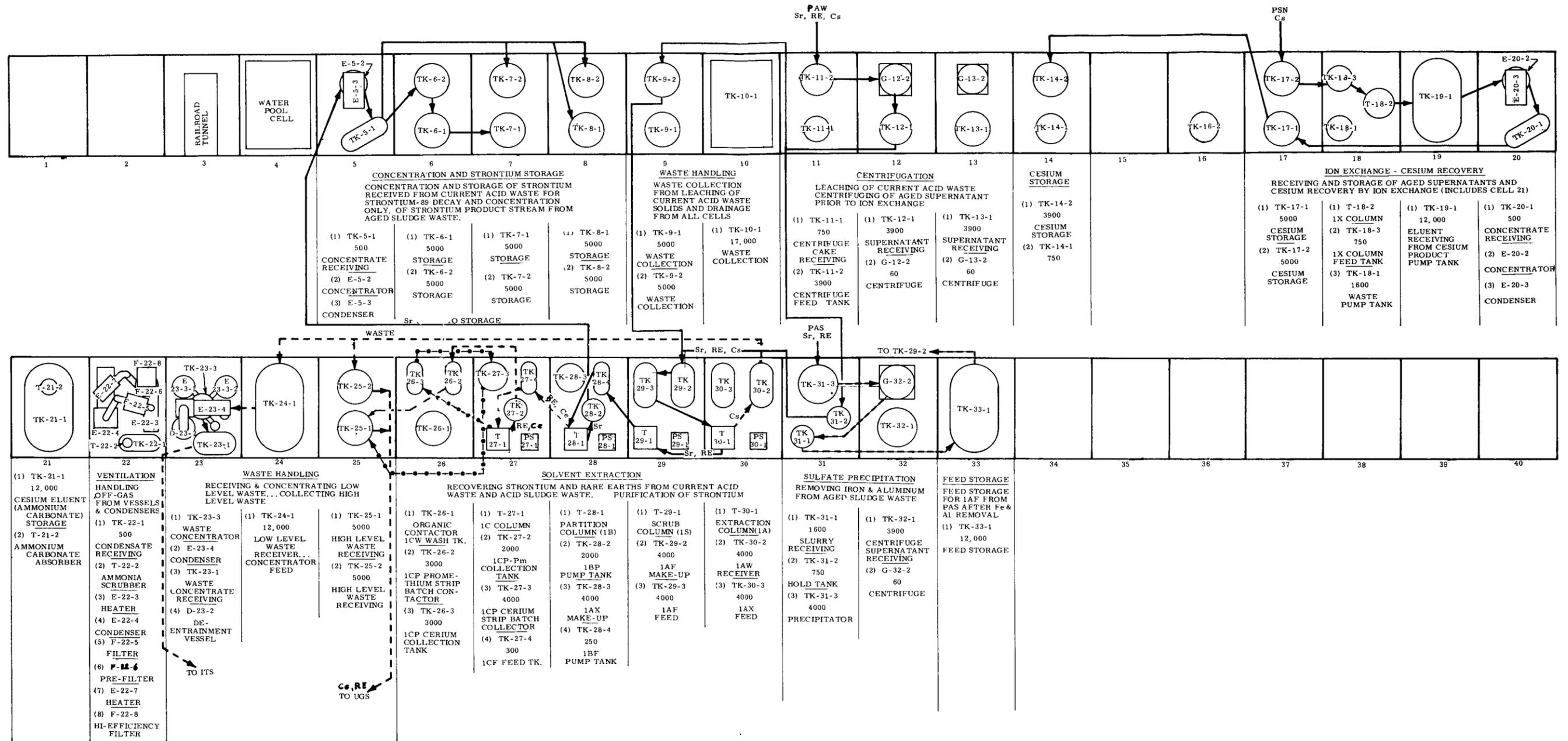


Figure IV-A-3

Cell Equipment and Functions

Fractionation of Strontium and Cesium from Current and Stored Boiling Wastes

see Section IV.B.4. All process equipment is stainless steel to resist corrosion. The equipment in the cells and service supplying the equipment are controlled at instrument panel boards which are spread throughout the length of the operating gallery. Maintenance of the equipment in the cells is done remotely using the crane to move the cell blocks and to make changes on the equipment.

The dispatcher coordinates operation, and provides constant monitoring of the alarms which are located in his office so that personnel can take prompt action to remedy the cause of the alarm. These vital functions include radiation alarms from the stack, the operating and pipe gallery, the canyon air monitor, the 15-inch and 24-inch cooling water sewers, and the steam and process condensate. Other alarms are related to the differential air pressure between the canyon and operating gallery, filter sumps, and exhaust system trouble. Control switches and indicating lights show and control the status of the air supply fans, and the locks on the access doors to the canyon and crane. Laboratory analytical results are also received in this office. Thus, ventilation, personnel access, and alarm systems are centrally controlled for the building.

Chemicals and services required for operations in cells are supplied via the pipe gallery, through the cell walls. Transfers of radioactive solutions into and between cells are made via the hot pipe trench. Any leakage from the vessels drains to the floor and thence to the deep cell (Cell 10) via the cell drain line. Access to the equipment in the cells is obtained by removing the cell cover blocks using the crane; maintenance work and alterations on the equipment also are done remotely using the crane.

Regulated work zones consist of areas where limited contact of personnel with radiation and radioactive contamination is allowed under carefully prescribed and monitored conditions. The following are regulated work zones: the canyon, the canyon crane gallery, the railroad tunnel, and the SWP lobby at the east end of the operating gallery.

The service portion of the building includes the operating, pipe, and electrical galleries. Other service areas are located adjacent to the 221-B Building in the 271-B Building. Further detailed information may be obtained from the Technical Manual. (14)

The layout of the B-Plant area is given in Figure IV-A-4. The functions of B Plant and the various facilities in its vicinity are as follows:

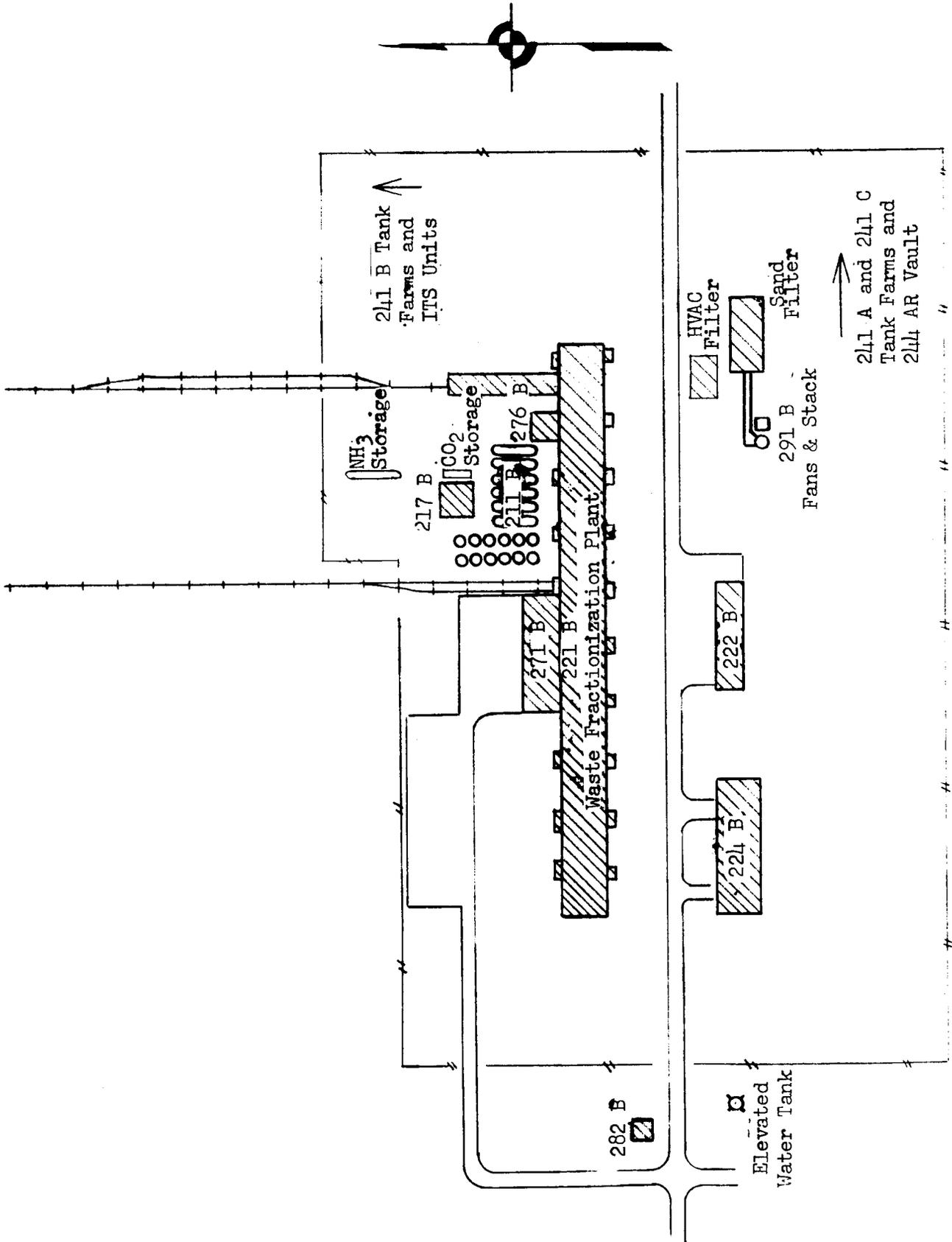


Figure IV-A-4

Layout of B-Plant Area

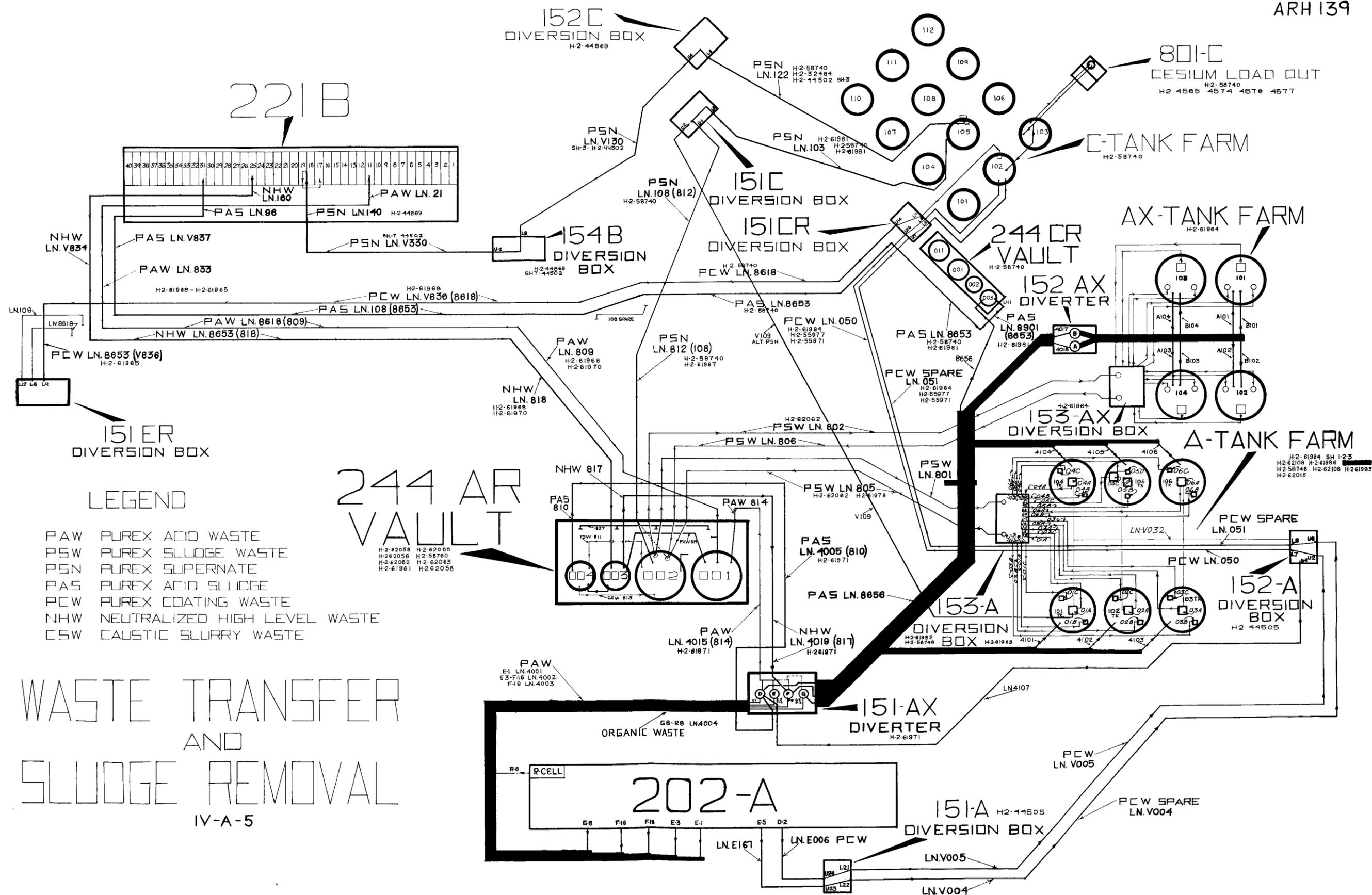
<u>Facility</u>	<u>Function</u>
221-B Waste Fractionization Plant	Processing of waste to isolate selected fission and other products.
224-B Future Products Recovery	Not used as part of initial waste management program. May be used later for the recovery of products not included in the original design.
271-B Service Building	Contains shops, supervisors room, offices and aqueous make-up facilities for the Waste Fractionization Plant.
222-B Office Building	Offices for support facilities, such as Financial.
291-B Fans, Stack, and Filters	Ventilation for the process building.
282-B Raw Water Well	Emergency raw water supply.
211-B Chemical Tank Farm	Bulk storage area for liquid chemicals used in the Waste Fractionization Plant.
271-B Water Demineralizer	Demineralized process water supply.
276-B Organic Make-up and Storage	Storage and make-up facility for organic solvent materials.
216-B-12 Process Condensate Crib	Crib for the disposal of condensates from process concentrators.
216-B-55 Steam Condensate Crib	Crib for disposal of condensates from heating coils in process tanks.
216-B-56 Organic Crib	Crib for disposal of organic solvents under emergency conditions.

The B-Plant is part of the waste management facilities which include in-tank solidification (ITS) units, underground waste storage tanks, a facility for the removal and handling of stored waste (244-AR Vault), a storage vault (244-CR), cribs, swamps, and associated facilities (see Figure IV-A-5 for waste transfer lines).

## B. Functional Description

### 1. Function

The function of the B Plant is to separate the long-lived fission products, strontium-90 and cesium-137, and intermediate-lived cerium-144 from the bulk salts in high-heating Purex and Redox wastes. The isolated strontium and cesium fractions are to be stored temporarily as solutions in B-Plant tanks until facilities are available to encapsulate the strontium and cesium in high integrity



221B

152C DIVERSION BOX  
H2-44869

801-C CESIUM LOAD OUT  
H2-58740  
H2-4585 4574 4576 4577

151C DIVERSION BOX

C-TANK FARM  
H2-58740

154B DIVERSION BOX  
H2-44869 SH7-44502

151CR DIVERSION BOX  
H2-58740

AX-TANK FARM  
H2-61964

244 CR VAULT  
H2-58740

152 AX DIVERTER

151 ER DIVERSION BOX

244 AR VAULT  
H2-62058 H2-62059  
H2-62056 H2-58760  
H2-62062 H2-62063  
H2-61961 H2-62056

153-AX DIVERSION BOX

A-TANK FARM  
H2-61964 SH 1-23  
H2-62108 H2-61986  
H2-58746 H2-62108 H2-61985  
H2-62016

LEGEND

- PAW PUREX ACID WASTE
- PSW PUREX SLUDGE WASTE
- PSN PUREX SUPERNATE
- PAS PUREX ACID SLUDGE
- PCW PUREX COATING WASTE
- NHW NEUTRALIZED HIGH LEVEL WASTE
- CSW CAUSTIC SLURRY WASTE

# WASTE TRANSFER AND SLUDGE REMOVAL

IV-A-5

202-A

151-A DIVERSION BOX  
H2-44505

ORGANIC WASTE  
E8-R6 LN4004

151-AX DIVERTER  
H2-61971

152-A DIVERSION BOX  
H2-44505

PCW LN.V005

PCW SPARE LN.V004

151-A DIVERSION BOX

LN.V005

LN.V004

LN.E161

LN.E006 PCW

PAW E1 LN.4001  
E3-F16 LN.4002  
F18 LN.4005

ORGANIC WASTE

E8-R6 LN4004

R6

RCELL

E8 F16 F18 E3 E1 E5 D-2

LN.V005

LN.V004

L21

L22

U23

U24

U25

U26

U27

U28

U29

U30

U31

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containers for long-term storage. The cerium is to be transferred to underground boiling waste storage tanks and kept segregated for decay. This fission product removal operation should permit the solidification of the bulk salt wastes within five to seven years from their generation. If the cerium were not segregated, an additional three years of aging would be required before the wastes could be immobilized.

The two sources of waste are Purex acid wastes as generated (PAW and ZAW), and stored Purex and Redox wastes. Following appropriate head-end treatments, solvent extraction will be employed to separate strontium and cerium from the currently generated and stored wastes. The cerium from current acid waste will be neutralized and sent to boiling waste storage. Initially, the cesium in PAW and ZAW will remain with the alkaline radiochemical salt waste for temporary storage for decay of short-lived radioisotopes and for separation of the waste into sludge and supernate fractions. After three to five years aging, cesium will be recovered from the supernates by ion-exchange processing in B Plant.

## 2. Process Description

Separations processing for fractionization of the wastes comprises three mainline processes: a) Feed preparation and solvent extraction of current acid wastes; b) Feed preparation and solvent extraction of stored sludge wastes; and c) Ion exchange of stored supernates and sluicing solutions which are required to remove the sludges from the tanks. These processes are briefly described below. The reader should refer to reference 15 for a detailed flowsheet description including stream compositions.

An inventory of strontium, promethium and other rare earths, which has accumulated in B Plant during the Phase I processing operation, will be processed through the solvent extraction equipment at plant startup. The process to be used is briefly described later in this section.

### a. Feed Preparation and Solvent Extraction of Current Acid Wastes

A simplified flowscheme depicting this process is shown in Figure IV-B-1. Current acid wastes (PAW) or (ZAW) containing the bulk of the fission products and from one to eight percent solids are denitrated in Purex and routed to B Plant via the 244-AR Vault. The solids in these wastes contain about 55 percent of the strontium and 70 percent of the rare earths. The solids, consisting mostly of silicates, phosphates, and sulfates, are treated by a carbonate-hydroxide metathesis solution to convert the sulfates to carbonate-hydroxide solids. These



solids are then dissolved with nitric acid to recover the fission products. A centrifuge (G-12-2) is used to perform the solid-liquid separations. The solubilized fission products are combined with the original acid waste supernate after it has been treated to form feed for the solvent extraction battery by the addition of a metal-ion complexant, a pH buffer, and a pH adjustment solution.

The feed is contacted in the first two (1A and 1S) columns with the solvent, consisting of di(2-ethylhexyl) phosphoric acid (D2EHPA) extractant and tributylphosphate (TBP) modifier in a normal paraffin hydrocarbon (NPH) diluent. The strontium, cerium, and other rare earths are extracted from the aqueous phase into the solvent. The aqueous fraction containing cesium from the 1A column is routed to the A or AX underground waste storage tank farms for temporary storage to allow decay of short-lived activity. The strontium is stripped in a third (1B) column with dilute nitric acid and concentrated in E-5-2 for storage in Cells 6-8. The cerium and other rare earths are stripped in a fourth (1C) column with nitric acid which is combined with the organic wash wastes and sent to underground waste storage tanks. The solvent from the 1C column is batch-washed to remove residual fission products and metallic impurities. The washed solvent is recycled to the 1A column following a small acid addition.

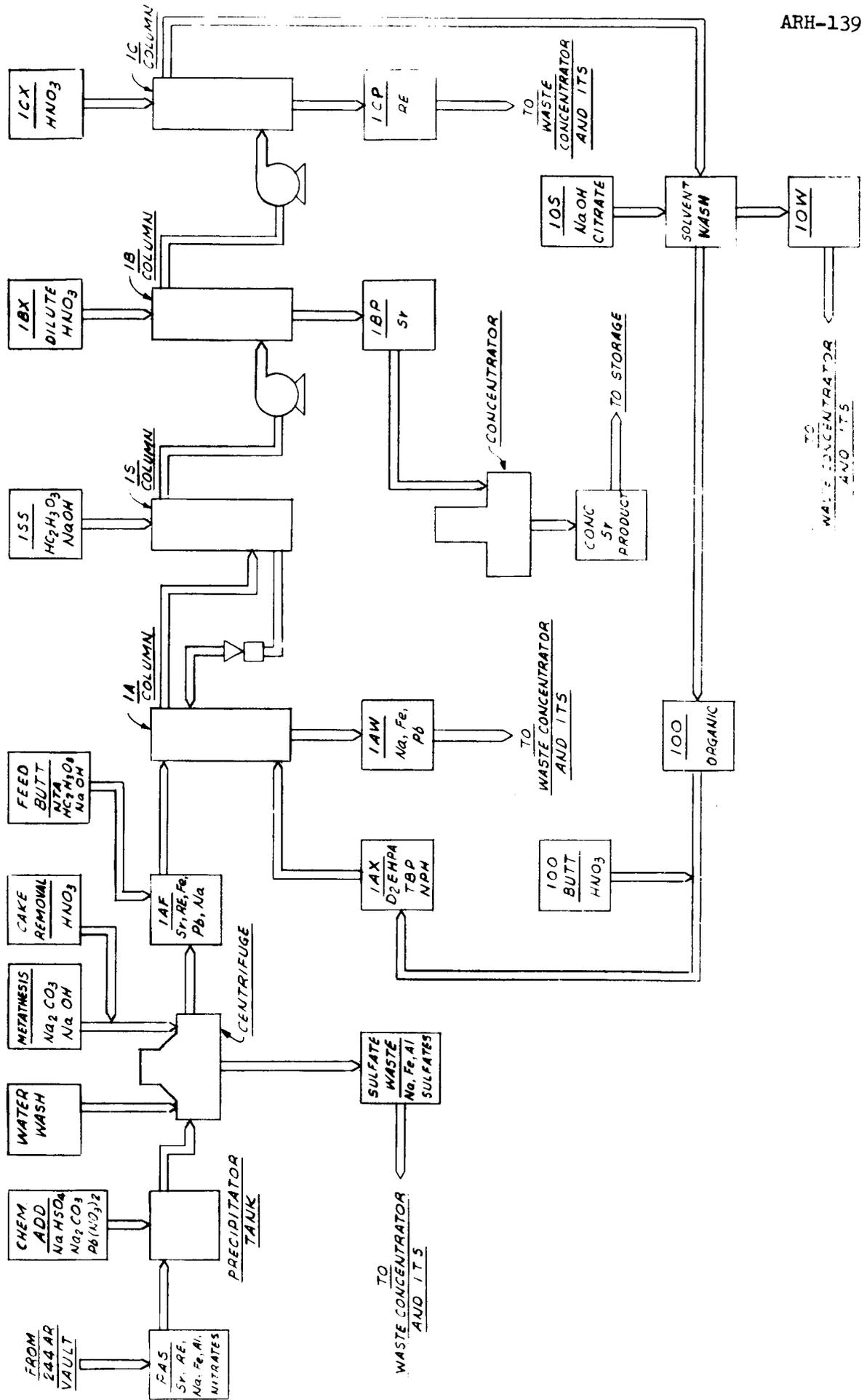
Overall recoveries are expected to be about 95 percent for strontium and rare earths. The principal difference in the Purex acid waste and Zirflex acid waste processing is the use of tri-sodium hydroxyethylenediaminetriacetate ( $\text{Na}_3\text{HEDTA}$ ) as complexant for Zirflex acid waste processing instead of tri-sodium nitrilotriacetate ( $\text{Na}_3\text{NTA}$ ). The solvent extraction process is to be operated under a slight vacuum and at temperatures less than  $65^\circ\text{C}$ .

b. Feed Preparation and Solvent Extraction of Stored Sludge Wastes

A simplified flowsketch illustrating the Purex Acidified Sludge (PAS) processing scheme is presented in Figure IV-B-2. The stored, high level, Purex and Redox sludges in the A, AX, and SX tank farms, following removal of the supernate, are sluiced with water or supernate discharged through nozzles at high velocity and pumped to the AR or SR Vault. The sluicing water, containing some Cs-137, is decanted to storage to await treatment for cesium removal. The sludge, containing the bulk of the fission products, is dissolved in nitric acid and accumulated for B-Plant treatment.

In B Plant, the rare earths and strontium are precipitated as sulfates using lead sulfate as a carrier to separate them

Figure IV-B-2  
SIMPLIFIED FLOWSHEET FOR STRONTIUM RECOVERY FROM PUREX ACID SLUDGES



from iron and aluminum. A sodium hydroxide-sodium carbonate metathesis is then performed to convert the sulfates to hydroxides and carbonates and to eliminate the bulk of the lead. The resulting cake is dissolved with nitric acid and accumulated for solvent extraction treatment. A centrifuge (G-32-2) is used in this operation for separating liquid and solid phases. The accumulated feed is then treated similarly to the current acid wastes in the extraction battery except that, in this case, the 1A column wastes, the rare-earth fraction and the solvent wash wastes are concentrated in B Plant (Cell 23) and transferred to immobilization processing (in-tank solidification). An overall removal of 90 percent is expected for strontium.

The current acid wastes and the strontium-rare earth fraction from the stored wastes will be campaign processed through the solvent extraction battery. The equipment will be flushed between campaigns as required.

c. Ion Exchange of Stored Cesium Supernates and Sluicing Solutions

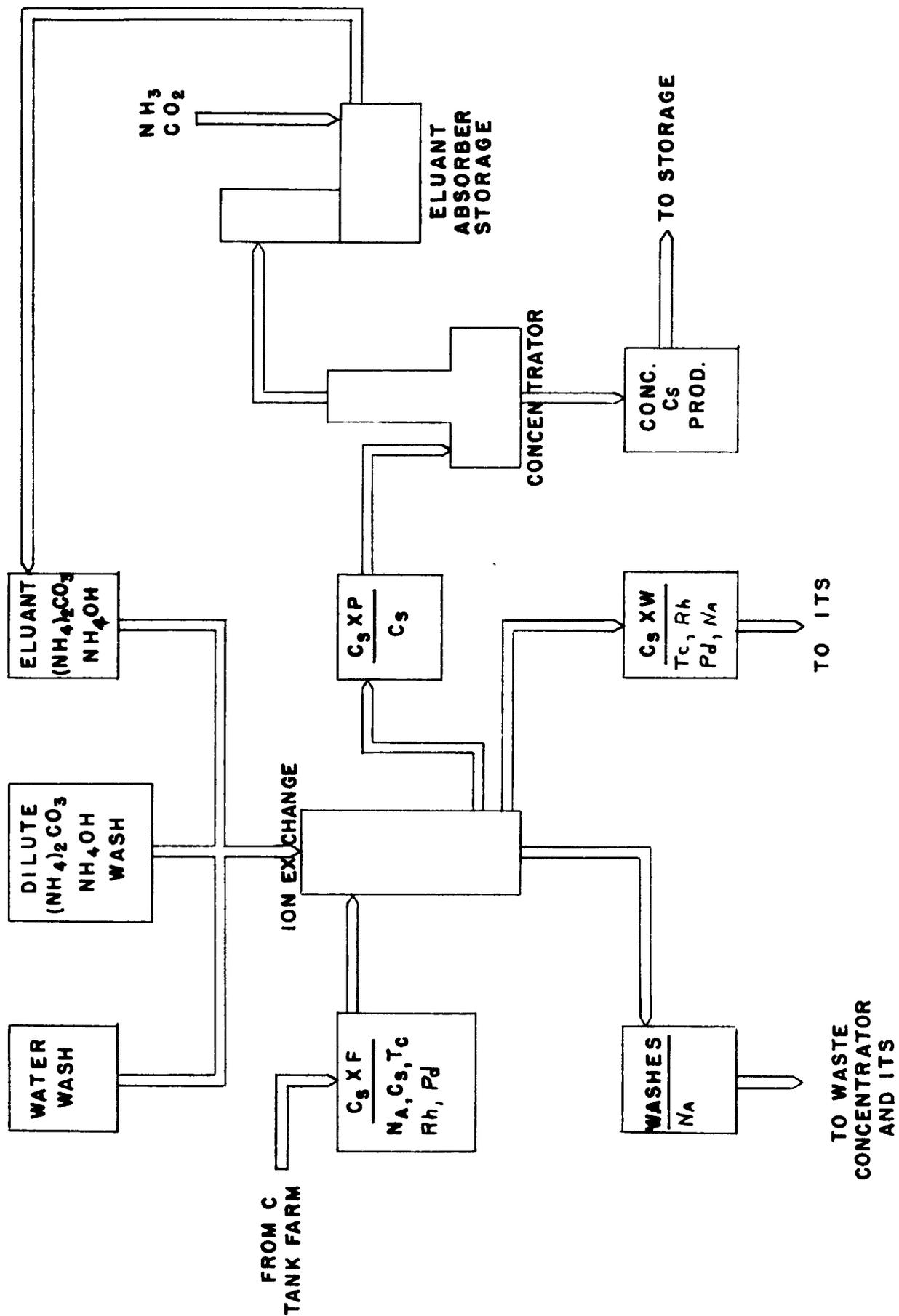
The simplified flowscheme for this process is shown in Figure IV-B-3. High-level tank farm supernates and sluicing water containing cesium-137 are passed through an ion-exchange bed in the B-Plant column (Cell 18) where the cesium and a small fraction of the sodium are loaded on a synthetic zeolite (alumino-silicate). About 97 percent of the loaded sodium and 0.5 percent of the loaded cesium are removed from the column with a dilute ammonium carbonate-ammonium hydroxide scrub. The cesium is then eluted with a concentrated mixture of ammonium carbonate and ammonium hydroxide. The eluted cesium is concentrated in the E-20-2 concentrator and stored in tanks in Cells 14 and 17. The eluant is condensed and recycled to the process. The waste from the loading cycle is routed directly to in-tank solidification. Column washes and scrubs are immobilized following B Plant concentration.

The ion-exchange column will be operated under positive pressure up to an estimated 15 psig and at temperatures up to 55°C. During operation, all pipe connections between the pressurized column and the operating and pipe galleries will be blanked to assure that radioactive solutions are not discharged into these galleries.

d. Solvent Extraction of B Plant Fission Product Inventory

There are two types of fission product solutions currently stored in B Plant. One contains rare earths and strontium-90. Both solutions contain only the rare earths (including promethium-147) and the other contains a high lead concentration introduced in the Purex Head-End fission product recovery process. The

Figure IV-B-3  
SIMPLIFIED FLOWSHEET FOR CESIUM RECOVERY FROM PUREX SUPERNATANTS



solvent extraction process which is to be used to purify these fission products is similar to that shown in Figure IV-B-1 with a few exceptions:

1.  $\text{Na}_3\text{HEDTA}$  is used for complexing metal ions in place of  $\text{Na}_3\text{NTA}$ .
2. Cerium is separated from the promethium and other rare earths in the 1C column using a silver-catalyzed persulfate-oxidation process. Cerium is oxidized to  $\text{Ce}^{+4}$  and remains in the solvent; whereas, other rare earths are not oxidized and are stripped from the solvent under the strong acid conditions. Cerium is later stripped from the solvent on a batch basis with nitric acid containing a reducing agent (sodium nitrite). The aqueous phase from the 1C column containing the promethium and rare earths is treated further by batch D2EHFA extraction to separate the sulfate from the rare earths.
3. The rare-earth fractions from both feed types are concentrated in E-5-2 and stored in Cell 8 rather than being sent to waste storage.
4. The strontium contained in one of the feeds is removed from the solvent in the 1B column and is concentrated in E-5-2 prior to storage in Cell 7. The rare-earth fraction in this feed is stored in tanks in Cells 32 and 33 temporarily until it can be campaigned through the E-5-2 concentrator.

### 3. Chemicals

The chemicals to be used in B Plant for waste fractionization are shown in Table IV-B-1. Additional chemicals required for purifying the rare-earth inventory are also included. The solvent extraction system is also equipped to purify strontium using a modified D2EHFA flowsheet which has been used for several years in the Semiworks.<sup>(16)</sup> Additional chemicals which are required for strontium purification are also included in the table.

TABLE IV-B-1

B PLANT CHEMICALS

	<u>Inorganic</u>	<u>Organic</u>
Waste Fractionization	Nitric Acid <sup>(1)</sup>	Hydroxyacetic Acid
	Sodium Hydroxide	$\text{Na}_3\text{NTA}$ <sup>(2)</sup>
	Sodium Bisulfate	$\text{Na}_3\text{HEDTA}$ <sup>(3)</sup>
	Lead Nitrate <sup>(1)</sup>	Tartaric Acid
	Sodium Carbonate	Oxalic Acid
	Ammonia	TBP <sup>(5)</sup>

TABLE IV-B-1 (Continued)

	Carbon Dioxide Sodium Nitrate(1) Linde AW-500	D2EHPA <sup>(6)</sup> NPH <sup>(7)</sup>
Promethium Purification	Silver Nitrate <sup>(1)</sup> Potassium Persulfate(1) Sodium Nitrite <sup>(1)</sup> Sulfamic Acid	Sugar
Strontium Purification	Hydrogen Peroxide <sup>(1)</sup>	Acetic Acid Na <sub>5</sub> DTPA <sup>(4)</sup> Citric Acid

- (1) These chemicals are generally classified as oxidants.
- (2) Na<sub>3</sub>NTA = Tri-sodium nitrilotriacetate.
- (3) Na<sub>3</sub>HEDTA = Tri-sodium N-hydroxyethylethylenediaminetriacetate.
- (4) Na<sub>5</sub>DTPA = Penta-sodium diethylenetriaminepentaacetate.
- (5) TBP = Tributylphosphate.
- (6) D2EHPA = di-(2-ethylhexyl) phosphoric acid.
- (7) NPH = Normal paraffin hydrocarbon.

#### 4. Inventory of Radionuclides

The inventory of radionuclides in B Plant may be broken down into two categories: processing inventory and storage inventory. Since current acid waste and aged stored waste will be campaigned through the solvent extraction equipment, the inventory of short-lived radionuclides in process will vary significantly. The estimated maximum inventories of selected fission products in B Plant in existing equipment are shown in Table IV-B-2.

TABLE IV-B-2MAXIMUM FISSION PRODUCT INVENTORIES

	Megacuries		
	<u>Sr-90</u>	<u>Cs-137</u>	<u>Ce-144</u>
<u>In Process(PAW)</u>	3.4	1.2	20
<u>In Storage</u>			
TK-6-1,6-2,7-1	85	-	-
7-2,8-1,8-2			
TK-14-2,17-1	-	25	-

Since these radionuclides release a large amount of decay heat, cooling coils are required in the tanks to maintain solution temperature control. The quantities of fission products which may be placed in the vessels will be limited so that the contents of the vessels can be maintained below jetting temperatures without agitation and with the emergency water supply.

The mode of operation adopted for storage of the product concentrates is always to have an empty tank (or tank freeboard) available to permit emergency transfer of the defective tank's contents within a period of several hours.

Other backup safety systems or operating procedures can be used on emergency to supplement the empty tank philosophy:

1. The coil discharge jumper can be disconnected from the defective tank and cooling water discharge allowed to run to the deep cell (Cell 10) via the cell drain system.
2. Blank connectors can be removed and process solution allowed to overflow to the cell drain system by addition of water to dilute the contents of the defective tank. The deep cell has a cooling coil which has adequate area for emergency duty. In normal plant operations, the deep cell should be maintained essentially empty to be available for emergency service.
3. The solution could be allowed to boil in a defective tank. Water would be added to the tank to maintain the contents in a mobile form.

Figure IV-B-4 shows the present transfer routes examined for emptying the B-Plant canyon vessels to underground tank storage should the need arise. It is estimated that 1.5 days would be required to empty the building vessels assuming that jets and pumps function properly.

### C. Pertinent Safety Features

#### 1. Confinement of Radionuclides

As in all Hanford radiochemical processing plants, the facility design and the operating procedure for B Plant provide for the maintenance of multiple containment barriers between radioactive material and: (1) the environment, and (2) the operating personnel.

##### a. Air-Borne

##### (1) Normal Provisions

For normal confinement of air-borne radionuclides, the

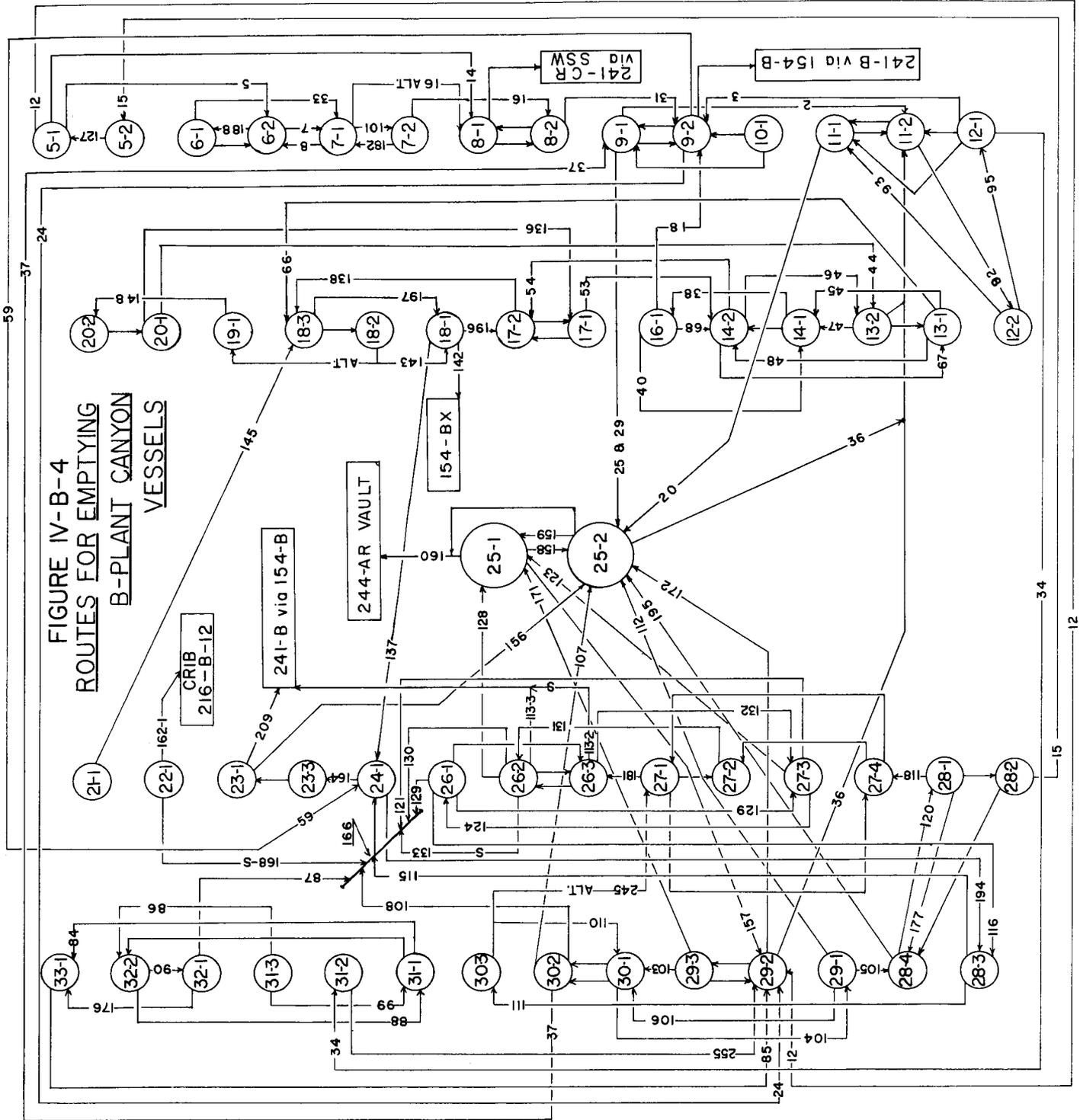


FIGURE IV-B-4  
ROUTES FOR EMPTYING  
B-PLANT CANYON  
VESSELS

plant areas can be categorized into four zones, namely:

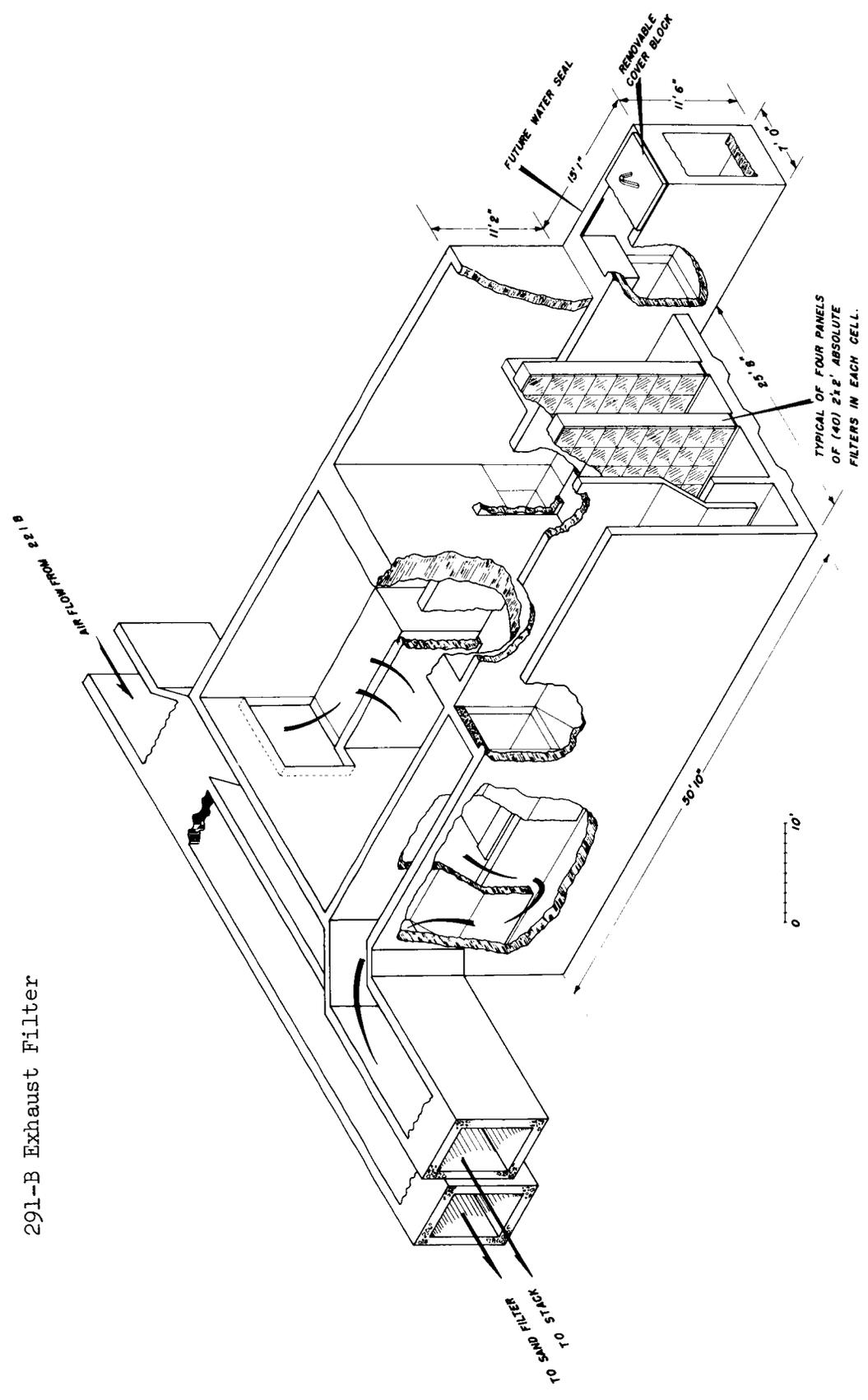
- Zone 1 - The areas which are not intended to be contaminated at any time, such as offices.
- Zone 2 - The areas that are normally not contaminated but which might become contaminated through abnormal processing conditions, such as the operating and pipe galleries and the canyon deck.
- Zone 3 - The process cells in which the process equipment is located. These cells are known to be heavily contaminated but contain only minor amounts of process material.
- Zone 4 - The process equipment itself; that is, vessels and auxiliaries intended to contain the radioactive materials. This equipment, through normal failures and/or maintenance operations, can release process materials into Zone 3.

Pressure differentials are maintained between the different zones so that the flow of ventilation air is from the less radioactive to the more radioactive areas. The ventilation systems are also designed to maintain the necessary direction and velocity of air flow during intentional breaches of zone barriers such as entry into Zone 2 or 3 for operational or maintenance purposes.

The B-Plant Zone 3 air is normally exhausted at the rate of 75,000 cfm through filter system consisting of two stages (banks) of high-efficiency filters in series, each having a particle removal efficiency of 99.97 percent for 0.3 micron diameter particles, resulting in a decontamination factor (D.F.) of  $10^7$ . (See Figure IV-C-1.)

The gaseous effluents from Zone 4 (process vessels) are exhausted through one of the two vessel vent systems. (See Figure IV-C-2.) Vessel Vent System No. 1 consists of a header which vents process vessels that do not contain significant quantities of ammonium compounds. The header exhausts through a steam heater (E-22-3), two high-efficiency filters in series (F-22-5), a 1200 cfm steam jet (J-22-4) and discharges to the air tunnel. Thus, gases passing through the No. 1 Vessel Vent System are subjected to four stages of high efficiency filtration before release to the atmosphere, theoretically resulting in a D.F. of  $10^{14}$  for 0.3 micron

Figure IV-C-1  
291-B Exhaust Filter





diameter particles. Vessel Vent System No. 2 services tanks which contain ammonium compounds. The system exhausts through a scrubber (T-22-2) for removal of ammonia vapors, a steam heater (E-22-7), a glass wool prefilter (F-22-6), two high-efficiency filters in series (F-22-8), two back-up, high-efficiency filters in series (F-22-9), a 125 cfm steam jet, and discharges to the atmosphere via the 24-inch sewer. All gases exhausted through the No. 2 Vessel Vent System are therefore subjected to the glass-wool prefilter and four stages of high-efficiency filters prior to atmospheric release, theoretically resulting a D.F. of  $10^{16}$  for 0.3 micron diameter particles.

(2) Emergency Provisions

The ventilation system is provided with the following features to assure containment of air-borne radionuclides under emergency conditions:

- A steam turbine powered exhaust fan, rated at 37,500 cfm, is provided to maintain building ventilation in the event of a power failure.
- The original, B-Plant graded-sand, exhaust filter is available for emergency service. It can be used in parallel with the high efficiency filters or it can be used independently at a flow rate of 37,500 cfm.
- Any gases escaping from a ruptured process vessel must pass through the building exhaust filter system, thereby assuring two stages of high-efficiency filtration prior to atmospheric release.
- Sampling and monitoring equipment is located on the stack and at various points throughout the plant to immediately alert personnel to abnormal levels of air-borne radionuclides.

b. Liquids and Solids

(1) Normal Provisions

The control of liquid and solid radioactive materials is fundamental to the operation of the plant. Spent solid materials which have become contaminated with radioactive material are buried. Large pieces of plant equipment which fail beyond repair are placed in wooden or concrete boxes for burial in trenches.

Liquids may be characterized as low, intermediate, and high activity-level streams. Low activity-level liquids are

those which do not create a hazard when discharged to the ground surface. Intermediate activity-level liquids are those with radionuclide concentrations up to several times their maximum permissible concentrations in water  $MPC_W^{(17)}$ , and which have a potential for contaminating the environs. High activity-level liquids are those with radionuclide concentrations many times their  $MCP_W$  and have a high potential for contaminating their environs.

The cooling water discharged from the coils and tube bundles is a low activity-level liquid. Normally, the radionuclide concentrations are below their  $MPC_W$ , however, the streams may become contaminated if process solutions were to enter a failure in a coil or tube bundle. Raw water supplies the cooling medium for process vessels. It is regulated in the operating gallery, goes through a coil or tube bundle in a vessel, drains through subheaders outside the building, and joins discharged cooling water from other vessels and goes into a common 15-inch line. An alternate 24-inch header also is available. The water is sampled and is monitored by a scintillation crystal to check for the presence of radionuclides. The water in the 15-inch line runs through a diversion box to a 24-inch line, to a retention basin and thence to an open pond (see Figure IV-C-3). The cooling water is automatically diverted by an electric motor-driven valve to a specific retention trench when the monitor detects radionuclides in the cooling water. The valve is fully open before the water at the monitor has moved more than half-way the distance to the diversion box. The monitor also sounds an alarm at the dispatcher's office, thus alerting operating personnel of the discharge of radionuclides. The process condensate and steam condensate from process heat exchangers are intermediate activity-level liquids and are routed to cribs 216-B-12 and 216-B-55, respectively (see Figure IV-C-4). Each stream passes through a tank where it is sampled, and monitored for radioactivity by a Geiger-Muller tube. An audible alarm sounds at the dispatcher's office when radionuclides are detected in the condensate thus alerting operating personnel to take prompt action to reduce release of further radionuclides by selectively shutting down suspect process one at a time and trying to identify the source of radionuclides.

The safety of this technique relies primarily on the ion exchange properties of the soil and on the fission-product decay in transit to the Columbia River. Test wells have been drilled in the area to monitor the movement of radionuclides. The appearance of long-lived radioactive materials such as cesium-137, strontium-90, cobalt-60, or plutonium beneath the crib in concentrations exceeding the limits specified under



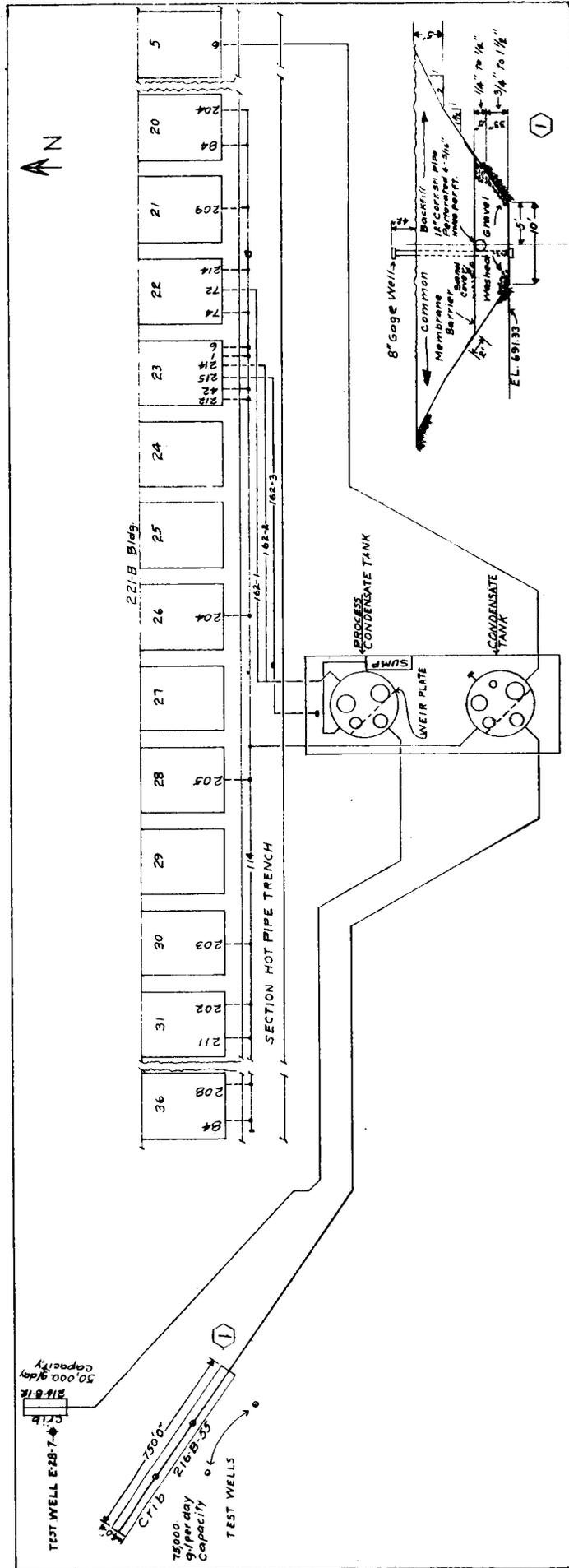


Figure IV-C-4  
Process and Steam Condensate Lines

Annex 1, Table 2, Column 2, in AEC Manual, Chapter 0524 is cause for diversion to a new crib.

The high activity-level streams are confined within the building in stainless steel vessels and associated piping. Should a leak occur in a vessel, the liquid would drain into the cell and leave the cell via the cell drain system to the tank in the deep cell (Cell 10). This liquid may be removed by jetting to the waste system. Strontium and cesium product solutions are stored in stainless steel tanks and cooled by heat exchanger through the tube bundles or coils.

The high activity, level wastes are of two types: One generates heat at a sufficiently low rate to allow concentration to a salt cake within an underground waste storage tank. The second type generates enough heat from radionuclide decay to self-boil and is allowed to self-concentrate in underground waste storage tanks under controlled conditions. The study of these operations is not within the scope of this hazards evaluation so no further comment will be made other than to indicate that the feed to the plant and the effluent to the storage tanks are transferred underground in stainless steel lines.

## (2) Emergency Provisions

The release of radionuclides to the cooling water effluent header and thence to the retention trench will be followed by filling the trench to prevent possible spread of contamination. The amount of radionuclides released from the plant will be minimized by prompt action of the personnel to determine the source and to isolate the radionuclide bearing effluent. The total amount of water diverted to the retention trench will be minimized by diverting to the 24-inch cooling water header the effluent in those subheaders which contain no radionuclides. Both the radiation monitor and the electric motor which operates the diverting valve are on the emergency power circuit.

Similarly, plant personnel can promptly take action to reduce the quantity of radioactivity introduced into the process condensate and/or the steam condensate when the monitor alarm is tripped. If the monitor alarms, the presence of radioactivity will be verified by supplemental radiation monitoring and, if confirmed, suspect equipment which contributes radioactive condensate to the system will be identified by selectively shutting down such equipment. The source of activity would be isolated prior to restarting the system. The radiation monitors are on the emergency power supply circuits.

2. Radiation Protection

a. Radiation Protection Program

Heavy shielding is required for the protection of plant personnel because of the large quantities of highly radioactive materials contained in process vessels and associated piping. The process cells in the canyon are separated from the pipe and electrical galleries by nine feet of concrete, and the canyon area above deck is separated from the operating gallery by seven feet of concrete. Plant personnel working inside the canyon are protected from cell and hot pipe trench radiation by concrete cover blocks, most are six-feet thick. All underground lines carrying radioactive streams are covered with a minimum of seven feet of earth.

Operating equipment in the pipe and operating galleries is designed to prevent radioactive materials from backing up into these areas. Equipment capable of detecting radioactive air contamination or high radiation levels is spaced strategically throughout. Any significant quantity of radioactive materials that inadvertently backed up into these areas because of equipment failure or operating error would be detected and an alarm sounded promptly. In addition, equipment is available for continuously measuring unusual levels of radiation or air contamination in the canyon proper.

Radiation procedures are prepared for all work involving radioactive materials by personnel knowledgeable in radiation control. Routine programs are in place which require that all facilities are surveyed periodically for radioactive surface or air contamination and unusual radiation levels. The work force consists of management knowledgeable and experienced in work with large quantities of radioactive materials; and operations, radiation monitoring, and maintenance personnel are well trained and highly skilled in this type of work. Protective clothing and respiratory equipment are available for all work in radiation zones. All personnel entering these zones are trained in the use of this equipment.

Programs are in existence for measuring and recording external radiation and internal body deposition of radionuclides. All employees are required to wear film badges at all times. The badges are changed periodically and the estimated exposure as determined from the film badge is entered in the employee's permanent record. A whole body counter is available for measuring internal deposition from gamma emitters and a bioassay system is in place for determining deposition from other radioactive materials.

Emergency procedures have been prepared for facilities associated with Phase III. These procedures are periodically reviewed by all personnel and emergency drills are conducted on a routine basis.

A major function of operations personnel is to routinely collect process solution samples for analysis. Personnel exposure during sampling on the canyon deck is minimized by the use of a portable shielded Gilmont sampler as required. Lead shielding (4-inches) protects the operator from the solution while it is in the pipette above deck. The sample bottle containing the diluted sample normally has only 0.025 to 0.1 ml of undiluted solution; this is easily shielded with a steel cover for protection of the operator. Larger samples, up to 1.0 ml, may also be taken. These will be transported in shielded "doorstop" sample carriers.

Well-equipped facilities are provided in the plant for personnel decontamination. The medical staff of Hanford Environmental Health Foundation, Inc. who are highly skilled and have excellent facilities for personnel decontamination and treatment of contaminated injuries, are promptly available.

b. Radiation Levels

Radiation exposure above background will normally be incurred only in the canyon proper. During routine operation, most radiation exposure will occur during the sampling operation. This exposure will be maintained well within applicable standards. The possibility exists for extremely abnormal radiation levels, up to 1000's of R/hr. during remote canyon cell maintenance and subsequent decontamination. Administrative controls will be exercised to exclude personnel entry into these areas while work is being performed under these circumstances. Strong efforts will be directed toward assuring that radiation exposure will be maintained within applicable standards.

Radiation exposure in routine work areas, other than the canyon, will normally be below 1 mR/hr. However, the possibility exists for radiation levels on the order of 1000's of R/hr to occur in areas such as the pipe and operating galleries through equipment malfunction or operating error. Radiation warning equipment and associated procedures are in place that will allow evacuation of these areas in emergencies without applicable radiation exposure limits being exceeded.

3. Criticality Safety

The possibility of a criticality incident in B Plant is remote. The concentration of plutonium in the current acid waste and acidified sludge waste feeds is expected to be less than 0.1 gram per liter.

Inasmuch as the minimum critical density of plutonium systems is approximately 7 grams per liter, no criticality hazard exists for any amount of the feeds. Feed samples will be analyzed for plutonium to ensure that high concentrations of plutonium are not obtained. The Purex Plant operation also samples the current acid waste streams and reworks the waste for plutonium recovery if the concentration is too high.

Since the B Plant equipment is not geometrically favorable, mass limits have been established for each vessel.<sup>(18)</sup> The vessels will be sampled and flushed periodically to assure that these limits will not be exceeded.

Americium-241 will be present in the B-Plant feed streams. The minimum critical mass for americium-241 is greater than 20 kilograms. This is greater than the total quantity of americium which will be processed annually in the plant. Since americium remains with the rare-earth fraction and will not be accumulated in the plant, there will be no americium-241 criticality hazard.

#### 4. Industrial Safety

##### a. Plant Safety

These facilities will be manned with experienced well-trained personnel, many of whom have been with the project throughout the construction phase. Work will be performed using procedures which were designed to assure the safety of personnel and facilities. The process will be operated on a 24-hour basis and fire checks, which cover essential parts of the plant, will be conducted on every shift. Emergency plans for the facility are in place. There are back-up facilities that are almost immediately available to B Plant in case of emergency. These include excellent medical personnel and equipment, a professional fire department whose personnel are skilled in fire fighting and rescue, and are thoroughly acquainted with the plant, and evacuation buses and system capable of evacuating the entire personnel complement of the plant.

##### 221-B Building

The 221-B canyon building is constructed of reinforced concrete walls, floors, and roof. The canyon proper contains 40 cells, a railroad transfer cut, a work area, a hot pipe trench and air tunnel and a 75-ton bridge crane. It is adjoined by an electrical gallery below grade, a pipe gallery and an operating gallery above grade.

The number and location of fire extinguishers throughout the facility are adequate for the hazards. Flange and valve guards are installed on all corrosive and irritating chemical lines in all areas which may be occupied by personnel. The number and location of safety showers are adequate. Three canyon cells

each contains a 4000 gallon storage tank for B-Plant solvent. Electric equipment is also installed in each of these cells. Therefore, each cell is equipped with 12 wash down spray nozzles, 6 near the top and 6 approximately 9 feet from the top, to combat fire.

#### 271-B Building

The 271-B services building is constructed of hollow concrete block panel walls, reinforced concrete frame, and reinforced concrete floor and roof. The basement contains a compressor room, ventilation equipment, instrument shop, and general maintenance shop including a welding booth. The first floor is occupied by offices, change rooms, a stores receiving dock, and a chemical storage area with compressed gas cylinders outside on the dock. The upper floors contain a lunchroom, offices, and storage and chemical make-up areas.

The primary fire hazard is one of ordinary combustible material (Class A). The building is equipped with an adequate number of the proper type of fire extinguishers. There are adequate safety showers at proper locations.

#### 211-B Area

The 211-B chemical tank farm contains 24 large tanks of liquids for use in the B-Plant process. The anhydrous ammonia and normal paraffin hydrocarbons are flammable. The other chemicals present no fire hazard. The nitric acid and sodium hydroxide are corrosive. The other chemicals are not hazardous.

The bulk of the paraffin hydrocarbons is stored in underground tanks. The tank vents are equipped with approved flash arrestors. Exposure from fire in adjacent facilities is negligible. The anhydrous ammonia is stored above ground in a 12,200 gallon tank. The tank piping and equipment are built to the specifications of the USASI safety codes. The tank and safety devices have also been inspected by a third party inspector. Safety showers and fire extinguishers are located strategically throughout the tank farm.

The 276-B solvent storage building is constructed of hollow concrete block with a built-up asphalt roof on a poured concrete silo. There are two exits (hollow metal doors) with approved panic hardware and an explosion vent. A spot dry chemical automatic extinguishing system is installed over the normal paraffin make-up tank.

b. Chemical Safety

The use of chemical reagents and process systems in the B Plant involves potential hazards of both a physiological nature affecting the health of operating personnel and potentially disruptive reactions such as fires and explosions affecting building and equipment operation. The design and operational controls planned for using hazardous chemicals in the B Plant are based on past operating practices and experience that have proven safe and effective.

The toxicity, flammability, and reaction control of chemical reagents used in the B Plant are discussed in standard manuals such as: Sax(19). Operating practices and standards successfully employed at Hanford in the use of these chemicals will be used in the B Plant. Thus, use of materials, handling equipment, protective apparel, safety information and devices, pre-job safety planning, standard operating procedures, spot sampling, and work safety reviews will all contribute to control of chemically-hazardous materials. In addition, the chemical make-up system is designed with appropriate materials of construction, instrumentation and valving, service piping, drains, and vessel vent systems, showers and flush facilities. Valves, headers, and flush systems are designed to avoid inline mixing of acids and bases, oxidants and reductants, and other unstable mixtures. Oxidants are stored separately and are kept away from organic chemicals.

Air purges are directed to radiochemical solution storage vessels to keep the concentrations of hydrogen from radiolytic decomposition below the explosive mixture. Steam pressure to the concentrators is limited to assure temperature control and thus reduce the potential of solvent nitration and potential explosion hazards. Administrative controls prevent concurrent evaporation of cesium ion-exchange eluant and solutions of silver compounds in the low-level waste concentrator, to reduce the risk of potentially violent chemical reactions.

5. Operating Control

Operating control is maintained by adherence to the operating procedures and regulations which are based on the technical specifications and standards and on compliance with operation and policy guide instruction.

a. Technical Specifications

The technical specifications define currently known limits within which the plant and associated facilities may be operated. The specifications define product quality and set forth process and operating limits required to protect the environs, operating

facilities, and personnel.<sup>(18)</sup> The specifications are written by and issued by Waste Management Process Engineering, approved by the Manager, Research and Engineering, and accepted by the Manager, and other Section Managers as indicated on the approval sheets. Changes in specifications may be made only with the written approval of the Managers listed above.

b. Technical Standards

Standards set forth process and operating limits designed for efficient operation of the B Plant and associated facilities. While deviation from these standards would not cause immediate potential danger to the environs, equipment, or personnel, continued operation outside these limits could have serious consequences. Standards are approved by the Manager, Waste Management Process Engineering and accepted by the Manager, Fission Product Processing and other Subsection Managers as indicated on the approval sheets. Changes in standards may be made only with the written approval of the Managers listed above.

c. Operating Procedures

The Operating Procedures are designed to provide operating personnel with operating instructions which will assure safe and efficient operation within the limits set by the specifications and standards. The procedures are written by Process Control Engineers or Production Supervisors, issued by Process Control and approved by the Manager, Waste Management Processing. Changes in the procedures require written approval by the same personnel.

d. Facilities Change Notices

A system of approval and recording of actual plant and equipment changes on appropriate design reference media is well established and is known as the Facilities Change Notice (FCN) system.<sup>(20)</sup>

A FCN must be prepared to describe any physical change to plant or equipment as described in Operating Instruction 5.8.1.3 and is to be forwarded to Facilities Engineering. The individual authorizing the change is responsible: for determining justification and need, for costs, for obtaining necessary approvals when required prior to start of work, and for forwarding the FCN to Facilities Engineering. Research and Engineering personnel are responsible for reviewing and approving changes on request when an element of chemical or criticality safety is potentially involved. Facilities Engineering personnel are responsible for

reviewing and approving for engineering soundness and for prior approval when such changes affect process related equipment or structural modifications. They also update related reference media.

## V. POSTULATED INCIDENTS

Despite the precautions taken to prevent hazardous conditions, a breakdown of administrative procedures, an equipment failure, an operator error, and/or inadequate technology could permit the release of radionuclides to the environs. The probability is very low that such potential failures would result in a major release of radioactivity. Reviews were made of the B-Plant equipment and processes to identify the potential incidents that might involve large inventories of radionuclides and the vulnerability of containment barriers. Where two or more potential incidents were similar with respect to inventories, potential energy release, and containment barriers, the one which probably embraces the worst hazard was chosen for evaluation. The selected incidents are discussed below. In each case, the possible and probable effects are based on a best engineering estimate of the performance of structures and equipment and of the characteristics of the process materials under the conditions postulated.

### A. Hydrogen Accumulation and Explosion

High concentrations of radioisotopes are present in the B-Plant process solutions so that radiolysis of water to yield hydrogen is an important consideration. Air purges are used to maintain the hydrogen concentrations in the vessel atmosphere below the lower flammability limit of four percent in dry air.

The following assumptions are made in the analysis of the potential effects of a postulated hydrogen explosion:

- The explosion occurs in a strontium product storage tank containing 4000 gallons of 8000 Ci/gal strontium-90 solution.
- The hydrogen is completely oxidized to water according to the reaction  $2\text{H}_2 (\text{g}) + \text{O}_2 (\text{g}) \rightarrow 2\text{H}_2\text{O} (\text{g})$ .

The heat of combustion for this reaction is given as 51,550 Btu/lb of  $\text{H}_2$ . (21)

- The concentrations of hydrogen and oxygen in the tank atmosphere prior to the explosion are in stoichiometric proportions, i.e. 29.6%  $\text{H}_2$ , 14.8%  $\text{O}_2$ , and 55.6%  $\text{N}_2$ .
- Water vapor is assumed to be absent.
- The maximum quantity of solutions in the tank which can be converted in the explosion to an aerosol having particle sizes less than 100 microns in diameter is 0.1 percent of the tank's inventory (4 gals).

This assumption is based upon liquid atomization studies in which under ideal conditions a maximum of 0.1 percent of the total liquid passed through the atomizer was converted to particles less than 100 microns diameter.<sup>(1)</sup> The explosion should not produce a percentage of fine particles which is higher than from an atomizer especially designed to produce small particles.

- All particles less than 100 microns in diameter remain suspended and are removed only by the high efficiency filters.
- Each high efficiency filter is rated at 99.97 percent removal efficiency for particles equal to 0.3 microns in diameter. Two filters in series should provide an overall decontamination factor of up to  $10^7$  for removal of the particles generated in the explosion.

1. Potential Cause

A flammable mixture of hydrogen in air could be obtained in the tank if the purge air to the tank were stopped for a sufficiently long period of time. The time required to obtain a four percent hydrogen concentration, following total loss of purge air and weight factor instrument air, is estimated to be 0.6 hours, assuming a 1.2 cc/watt hour hydrogen generation rate.<sup>(22)</sup> An additional 3.6 hours would be required to obtain a 29.6 percent hydrogen concentration. The mixture is to be ignited by a spark from an electrical source (agitator, pump, etc.) of by maintenance work on the tank.

2. Energy Considerations

- The maximum pressure in the tank is estimated to occur in 0.01 second following detonation.
- Experimental results for similar conditions indicate a maximum pressure of about 60 psia would be obtained in the tank following an explosion.<sup>(23)</sup> Calculations, assuming adiabatic combustion, yield a maximum pressure somewhat higher than this. Since detonation would likely occur, a shock wave resulting from the detonation could produce a transient maximum pressure of greater than 1000 psia.
- The pressure resulting from the detonation would probably burst the tank. The fragments of projectiles from this tank could penetrate and rupture the neighboring strontium storage tank.
- The cell pressure, assuming attainment of equilibrium, would increase to about 40 psi. This pressure would not be sufficient to lift the cell cover blocks (6.1 psi required). Gases generated during the detonation would escape into the air tunnel resulting in a maximum pressure rise in the air tunnel of less than 0.3 psi, assuming all the reaction gases are released

instantaneously into the air tunnel. This pressure would not be sufficient to damage the high efficiency filters in the 291-B Building.

### 3. Path of Potential Radionuclide Release

- Normal - From vessel to vessel vent header, to vessel vent filters, to air tunnel, to 291-B filters, to stack, to atmosphere.
- Probable - From vessel to cell, to air tunnel, to 291-B filters, to stack, to atmosphere. Some process solution could also back-up into the operating or pipe gallery piping systems..
- Possible - From vessel to cell, to canyon, to air tunnel to 291-B filters, to stack, to atmosphere.

### 4. Probable Effects

The total quantity of radioactivity released to the atmosphere would be less than 0.1 curie of Sr-90. The strontium solution from the ruptured storage tanks would flow from the cell into the sump, through the 24" cell drain pipe (see Section V.K.) to the waste collection cell (Cell 10). The waste tank in cell 10 has a volumetric capacity of 10,000 gallons and a cooling capacity of 2,000,000 Btu/hr which is adequate to cool the contents of the two ruptured tanks.

Since the coil in the tank in which the detonation occurred would probably be ruptured, the cooling water to the coil would have to be shut off within 10 minutes to avoid overflowing the waste collection tank. The canyon would probably be highly contaminated and remote decontamination might be necessary before human occupation of the area would be permitted.

Radioactivity which was forced into the pipe and operating gallery piping systems could create radiation levels of several R/hr. The high radiation level would be detected with immediate notification of personnel by alarms. Procedures will be implemented to restore the area to normal radiation levels. (See Section V.1.3)

### 5. Preventatives

- Air purges through special dip legs, weight factor dip legs, and normal in-tank leakage are used to prevent the attainment of flammable mixtures.
- In the event of an electrical outage, process air is supplied by compressors operated on an emergency electrical supply. (See Section V.H.)

- Agitators, pumps, etc. will be shut off in the event of a loss of air purge to the tank containing radioisotopes. Maintenance work is not permitted on this tank during the air outage.
- If normal and emergency electrical power supplies are lost, portable gasoline driven compressors are available for use.
- If desired, the off-gas can be sampled to determine the potential for exceeding the lower explosive limit for hydrogen in air.

#### B. Ammonia Accumulation and Explosion

Anhydrous ammonia is used in the ion exchange process for recovery of cesium-137 from stored tank-farm supernates. Ammonia has lower and upper flammability limits in air of 15 and 28 percent by volume, respectively. The range of flammability decreases upon addition of water vapor to the system until at 20.8% NH<sub>3</sub> and 10.8% water the limits coincide.<sup>(24)</sup> If a mixture of ammonia and air stands over an aqueous solution of ammonia, solution temperatures greater than 44° will produce sufficient water vapor to render all mixtures of ammonia and air non-flammable.

The following assumptions are made in the analysis of a postulated ammonia explosion:

- The ammonia explosion occurs in the cesium eluent receiver tank (TK-19-1) which contains 12,000 gallons of 40 Ci/gal cesium-137.
- Ammonia is oxidized completely to nitrogen and water according to the reaction:



The heat of combustion is 8010 Btu/lb of NH<sub>3</sub> (calculated from heats of formation in reference 21).

- The concentrations of ammonia and oxygen are in stoichiometric proportions, i.e. 22% NH<sub>3</sub>, 16.5% O<sub>2</sub>, and 61.5% N<sub>2</sub>. Water vapor is assumed to be absent. Maximum explosion pressures are obtained when the components are in stoichiometric proportion.
- The maximum amount of solution in the tank which is converted in the explosion to an aerosol having particle sizes less than 100 microns in diameter is 0.1 volume percent of the tank's inventory (see Section V.A.).
- Particles less than 100 microns do not settle and are carried with the air stream.
- The initial pressure in the tank is 14.7 psia and the initial temperature is 80°F.

### 1. Potential Cause

A flammable concentration of ammonia could accumulate in the atmosphere of TK-19-1 if high concentrations of ammonium carbonate and ammonium hydroxide in the solution and a temperature less than about 44°C were attained. Ignition could occur from a spark of sufficient energy as generated by an electric source or from maintenance work.

### 2. Energy Considerations

- The maximum pressure in the tank is estimated to occur in about 0.1 second.
- The calculated maximum pressure in the tank assuming adiabatic combustion is about 140 psia. However, experimental results obtained in a 5-inch diameter by 25-inch long pipe, indicates a maximum pressure of about 90 psia for the assumed conditions. (24)

Either of these pressures would probably burst the tank resulting in an equilibrium cell pressure of up to 25 psia. The cell pressure would be sufficiently high to raise the cell cover blocks (6.1 psi required) and vent the cell to the canyon. The resulting pressure rise in the canyon would be less than an inch of water. The normal pressure in the canyon is a negative 0.2 to 0.3 inch of water.

- The maximum pressure increase attained in the air tunnel is estimated to be less than 0.3 psi. This pressure rise would not damage the 291-B Building filters.
- Detonation would probably not occur so that shock pressures which normally accompany a detonation would be avoided. It has been reported that mixtures of ammonia in air cannot be detonated. (25,26)

### 3. Path of Potential Radionuclide Release

As a result of the explosion up to 480 curies of cesium-137 contained in 0.1 percent of the tank's solution could escape the tank and enter the cell atmosphere as particles having diameters less than 100 microns. The cesium would leave the cell via two paths: into the canyon as a result of lifting the cell cover blocks, and into the air tunnel through the twelve 14-inch diameter ducts connecting the cell and air tunnel. The particles discharged to the canyon would later enter the air tunnel and be carried out to the 291-B filter building. Particles remaining in the air following high efficiency filtration would be discharged to the atmosphere through the 200-foot stack. Minor contamination could escape the slightly-pressurized canyon via openings around the railroad-tunnel doors, etc.

#### 4. Probable Effects

It is estimated that less than one millicurie of cesium-137 would be released to the atmosphere following an ammonia explosion. Since the tank would likely rupture, its contents would be spilled into the cell where it would overflow into the 24" cell drain pipe and discharge into the tank in the deep cell (Cell 10). The capacity of the 10-1 tank is 10,000 gallons so that the 12,000 gallons from TK-19-1 would overflow TK-10-1. If TK-19-1 coil were also breached during the explosion, it is likely that additional water would be added to the cell drain system until the cooling water was stopped. TK-10-1 is capable of dissipating up to 2,000,000 Btu/hr which is more than sufficient to keep the contents cool. Because of the geometry of the 10-1 tank in the bottom of Cell 10, solution which overflowed the tank could be kept cool and could later be removed from the cell prior to emptying the 10-1 tank.

Areas in the canyon would be highly contaminated and would require decontamination to reestablish normal radiation exposure conditions.

#### 5. Preventives

- All solutions containing ammonia are maintained at greater than 44°C where there is sufficient water vapor present at equilibrium to prevent attainment of flammable mixtures.
- The concentration of ammonia in solutions is normally maintained below that required to attain flammable mixtures at equilibrium at any temperature.
- Purge air is introduced near the bottom of the tank for hydrogen concentration control (see Section V.A) and would provide a source of water-saturated air to maintain sufficient water vapor for the prevention of flammable mixtures.
- Ignition sources having sufficient energy to initiate an ammonia explosion are not likely to be present. A value of 680 millijoules has been reported, which is at least 10 times that used in automobile test engines, and more than 1000 times that required to initiate hydrocarbon or hydrogen explosions.
- No maintenance work is permitted on vessels or appurtenances containing flammable concentrations of ammonia.

#### C. Red Oil Explosion

"Red Oil" is believed to be a complex mixture of nitrated decomposition products of tributylphosphate (TBP) and heavy-metal nitrate compounds such as uranium, plutonium, or thorium nitrates.<sup>(27,28)</sup> The material has a color varying from yellow-orange to red and a specific gravity range

of 1.1 to 1.6. It is produced by a chemical reaction between nitric acid, degraded TBP, and heavy-metal nitrates. It normally exists as a dense third phase in aqueous and organic systems. Red oil can explode upon heating in the presence of concentrated nitric acid and/or heavy-metal nitrate systems at temperatures above 135°C depending upon the applied pressure and chemical composition of the mixture. Violent reactions of TBP decomposition products and nitric acid (no heavy metals present) have been produced in the laboratory at high initial pressures of from 1 to 30 atmospheres and temperatures greater than 150°C. (28)

The B-Plant solvent consists of 0.2M TBP and 0.3M D2EHPA in an NPH (normal paraffin hydrocarbon) diluent. Small amounts of extracted uranium (up to 0.002M) will be present in the solvent at some points in the process and nitric acid solutions which have contacted this solvent will be thermally concentrated.

Recent laboratory studies to attempt to produce red oil in boiling systems containing B-Plant solvent and high concentrations of strontium nitrates, rare-earth nitrates and nitric acid have been unsuccessful. Addition of a small amount of uranyl nitrate (0.1M) and radiolytically degraded solvent (to  $> 1 \times 10^6$ R) did not yield red oil after more than 72 hours of boiling under constant reflux. (29) These conditions were considerably more stringent than those normally expected in B Plant. Nevertheless, it may be possible under adverse plant conditions to form nitrated solvent (red oil) so that a red oil explosion is postulated to occur in the E-5-2 concentrator. (Figure V-C-1) Another concentrator in B Plant (E-23-3) could potentially contain solvent; however, it is operated under alkaline conditions where red oil explosions should not take place. Nevertheless, this concentrator will be operated with the safeguards described in the succeeding sections.

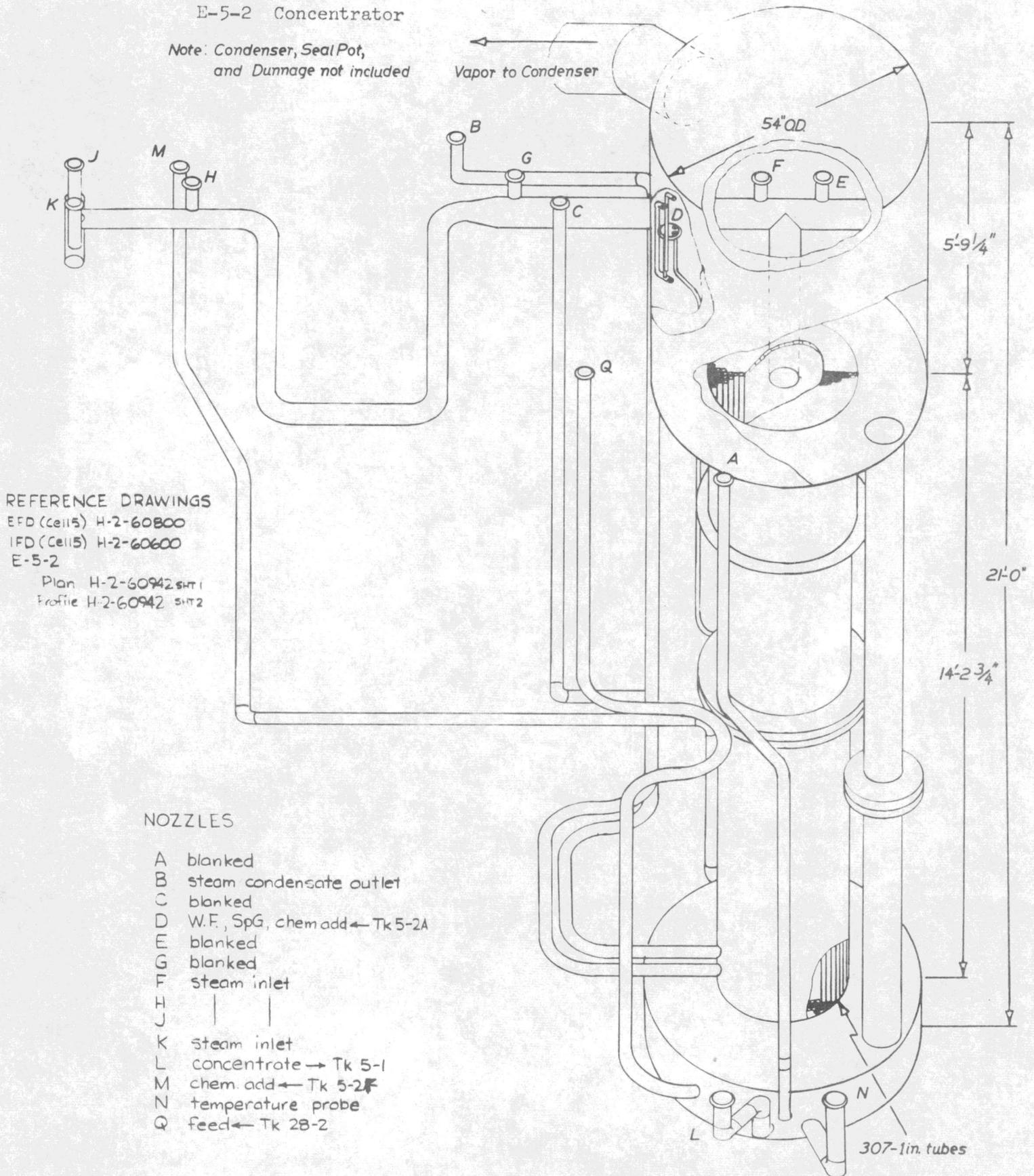
The following assumptions are made in postulating a red oil explosion:

- Five gallons of red oil are accumulated in the E-5-2 concentrator which also contains 300 gallons of 8000 Ci/gallon strontium-90 solution.
- The explosion results in the release of 300,000 BTU and about 4500 ft<sup>3</sup> of reaction gases. (1)
- A maximum of 0.1 percent of the solution in the concentrator is converted in the explosion to an aerosol having particle sizes less than 100 microns diameter (see Section V.A.). It is assumed that these particles do not settle out in the ventilation system piping and duct work.

Figure V-C-1

E-5-2 Concentrator

Note: Condenser, Seal Pot,  
and Dunnage not included



REFERENCE DRAWINGS  
 EFD (Cell 5) H-2-60800  
 IFD (Cell 5) H-2-60600  
 E-5-2  
 Plan H-2-60942 SH11  
 Profile H-2-60942 SH12

NOZZLES

- A blanked
- B steam condensate outlet
- C blanked
- D W.F, SpG, chem add ← Tk 5-2A
- E blanked
- G blanked
- H steam inlet
- I |
- J |
- K steam inlet
- L concentrate → Tk 5-1
- M chem. add ← Tk 5-2F
- N temperature probe
- Q feed ← Tk 28-2

- The initial pressure in the concentrator is 14.7 psia and the initial temperature is 150°C.
- Two CWS high-efficiency filters in series should provide a minimum DF of  $10^7$  for particles greater than 0.3 microns.

1. Potential Cause

Solvent could enter the E-5-2 concentrator from the concentrator feed tank (TK-28-2). After a sufficiently long residence time in the concentrator, the solvent could become nitrated forming red oil and upon failure of the steam pressure limiting valve, temperatures in excess of 150°C could be attained, resulting in a "red oil" explosion.

2. Energy Considerations

- The maximum pressure in the concentrator would be reached in about 0.01 second.
- A maximum pressure in the concentrator of greater than 2000 psia could be attained. The concentrator would burst releasing the contents to the cell. The pressure in the cell would approach 30 psia which would be sufficient to lift the cell cover blocks. The maximum pressure in the canyon following attainment of equilibrium would be less than 2 inches of water. The railroad-tunnel doors should withstand this pressure increase.
- The water in the concentrator seal pot would be blown out of the pot into the air tunnel. Some solution in the concentrator would also be discharged directly into the air tunnel through the 8" diameter pipe connecting the seal pot with the concentrator overhead line. This solution would drain out of the tunnel into the 10-1 tank.
- Assuming sonic gas flow is attained in the line to the seal pot, and through the ventilation ducts connecting the cell to the air tunnel for the estimated 0.25 seconds required to vent the cell to the canyon, the maximum pressure rise in the air tunnel is estimated to be less than 0.3 psi. (If the cell were instantaneously vented entirely to the air tunnel without credit for canyon venting, the maximum pressure rise in the air tunnel is estimated to be less than 0.8 psi.) The high efficiency filters in the 291-B Building should not be damaged by this sudden pressure increase.

### 3. Path of Potential Radionuclide Release

Up to 0.1 per-cent of the strontium in the concentrator would be converted to particles having diameters smaller than 100 microns. These particles would enter the air tunnel via three means: 1) through the lines connecting the seal pot to the concentrator overhead line and air tunnel, 2) through the ventilation ducts in the cell, and 3) into the canyon and subsequently to the air tunnel through other cells. It is assumed that the particles would be swept through the tunnel to the 291-B filter building. Particles not removed by the filters would be discharged to the atmosphere via the 200-foot stack. Minor contamination could escape the slightly-pressurized canyon via openings around the railroad-tunnel doors, etc.

### 4. Probable Effects

The quantity of strontium-90 released to the atmosphere is estimated to be less than 1 millicurie. Strontium solution from the concentrator and an adjacent vessel, TK-5-1, (which would also likely be ruptured) would flow from the vessels through the 24" cell drain line to the 10-1 tank. The concentrator tube bundle would probably rupture permitting steam to escape directly into the cell.

When the steam pressure fell to below 10 psig, the steam would be automatically cut off and air from a 20 psig source would be admitted to the cell. Insufficient steam would be admitted to damage the 291-B filters from overloading with moisture. Air would continue to flow at about 100 CFM until the supply was manually valved off in the operating gallery. The air flow rate would be sufficiently low to permit handling by the ventilation system.

### 5. Preventives

The explosive red oil reaction requires the presence of a significant amount of TBP and a temperature in excess of 150°C in the absence of heavy metals. None of these conditions normally exist in the E-5-2 concentrator. The following safeguards are provided to prevent a red oil reaction:

- The automatic interface control system on the 1B column limits the amount of solvent which leaves the column via the aqueous effluent line to the concentrator pump tank (TK-28-2).
- The presence of appreciable organic in the E-5-2 concentrator is rapidly indicated by abnormal concentrator operation (foaming, loss of vessel vacuum, reduced boil-off

rate). As little as 1 gallon of solvent should be detectable in the concentrator.

- Continuous boil-off overhead of small organic inputs to the concentrator prevents an organic phase accumulation in the concentrator.
- The temperature in the E-5-2 concentrator is kept below that at which the red oil reaction can be initiated by limiting the steam pressure to less than 39 psig and the steam temperature to less than 141OC.
- The quantity of solvent introduced to the concentrator is minimized. Feed which is pumped from TK-28-2 to the concentrator is decanted from the bottom of the unagitated tank. The residence time in the tank is sufficiently long that any solvent which entered the tank from the 1B column would collect as a light phase on top of the aqueous phase, assuming that emulsions are not formed. Instrumentation is installed in the tank to detect the presence of an organic layer. Accumulated solvent may be removed from the tank as required.
- The Purex Plant concentrators have been operated using similar preventive measures for many years without a red oil explosion.<sup>(30)</sup>

## V. POSTULATED INCIDENTS

### D. Fire

The most serious radiological hazard resulting from a fire at B-Plant is considered to be a fire in the 291-B Building which houses the high-efficiency exhaust filters for the plant exhaust ventilation system.

#### 1. Potential Cause

The postulated cause of a fire in the 291-B Building is a fire in the canyon of the 221-B Building which spreads through the building exhaust ventilation ducts finally reaching the exhaust filter building, 291-B. A collection of flammable materials such as lint and nitrate salts is postulated to exist in the building air tunnel, exhaust duct and filter building to spread the fire and support combustion once the fire reaches the filter building.

As part of this hazards analysis, it has been determined that the heat emitted from the quantities of radionuclides which is expected to collect on the filters is not sufficient to ignite

flammable materials which might be present nor is the heat sufficient to adversely affect the integrity of the filters.

## 2. Path of Potential Radionuclide Release

A fire in the 291-B Building such that both stages (banks) of filters are destroyed would release radionuclides from the filters to the exhaust air stream, to the stack, and to the atmosphere.

## 3. Postulated Effects

### a. Possible

It is possible, if sufficient fuel were available and if the fire burned for a long enough period of time, that both filter banks could be depleted of all the radioactive materials which they contained. The estimated quantities of radionuclides which could be released to the atmosphere from such a fire are shown in Table V-D-1.

TABLE V-D-1  
POSTULATED EFFECTS OF FIRE IN 291-B FILTER BUILDING<sup>a</sup>

<u>Nuclide</u>	<u>Possible Release (Curies)</u>	<u>Probable Release (Millicuries)</u>
Ru-103	60	6,000 <sup>b</sup>
Ru, Rh-106	300	30,000
Zr, Nb-95	10	3
Ce-144	600	200
Cs-137	100	30
Sr-90	200	60
Sr-89	10	3

<sup>a</sup>The estimated quantities of nuclides released are based on a dust loading of five pounds per unit for the first filter bank and nuclide concentrations in the dust which were determined as follows:

- Dust concentrations are based on an analysis of salts collected in the primary Purex exhaust filter over a period of nine years.<sup>(33)</sup>
- Corrections were made for differences and inert dust sources and radionuclide process inventory of B Plant versus the Purex Plant.

<sup>b</sup> It is assumed that all ruthenium is converted to the volatile tetroxide and vaporized by the heat from the fire; the vaporized ruthenium is assumed to pass through the second filter bank with a D.F. of 10 resulting from the reduction of part of the tetroxide to the particulate dioxide.

Tables V-D-2 and B-D-3 contain criteria developed by the Phase I hazards review team<sup>(2)</sup> for release of radioactive strontiums 89 and 90.

TABLE V-D-2

INHALATION EXPOSURE CRITERIA

<u>Exposure Zone</u>	<u>uCi Sr-90 Deposited<sup>a</sup></u>	<u>Effect<sup>b</sup></u>
A	180	Lethalities after prompt and extended hospitalization.
B	30 - 180	Deleterious effects may involve late hospitalization.
C	2 - 30	Slight increase in incidence of leukemia and bone sarcoma.

<sup>a</sup> Deposited in bone.

<sup>b</sup> Includes the effect of the attendant Sr-89 deposited.

TABLE V-D-3

PROPERTY CONTAMINATION CRITERIA

<u>Exposure Zone</u>	<u>uCi Sr-90/m<sup>2</sup></u>	<u>Probable Corrective Action</u>
D	20	Extended evacuation. Destructive methods of decontamination.
E	3	Confiscation of current and future crops. Less rigorous decontamination required.
F	0.15	Loss of milk and some current crops. Treatment of soil. Minor decontamination.

None of these criteria is likely to be exceeded off the Hanford Reservation for a 200 Ci release of strontium-90. A very small area estimated at 1 - 2 square miles could fall in exposure zone F on the reservation if poor atmospheric conditions existed. However, no corrective action would be anticipated.

One microcurie per square meter ( $1 \text{ uCi/m}^2$ ) or greater has been defined as the nuisance contamination level for fission products other than strontium-90 where countermeasures would be required for public land.<sup>(34)</sup> It is not anticipated that this level would be approached off the Hanford Reservation.

Table V-D-4 shows estimates of the maximum exposure to critical organs by personnel within the path of the radioactive plume if it were assumed that poor atmospheric conditions existed at the time of release (wind speed 1 m/s, unstable or neutral condition). The maximum exposure estimates are within the yearly limits of occupational employees, 15 rem for lung, 30 rem for bone, and 5 rem for whole body. It is not anticipated that anyone off the Hanford Reservation would exceed the yearly limits for non-occupational people, 1.5 rem for lung, 3.0 rem for bone, and 0.5 rem for whole body.

TABLE V-D-4

MAXIMUM EXPOSURE TO PERSONNEL FROM PASSING RADIOACTIVE PLUME

Radionuclide	Quantity Inhaled	Dose <sup>a</sup> in One Year		
		Lungs	Bones	Whole Body
Sr-89	.12 uCi	20 mrem	49 mrem	1 mrem
Sr-90	2.4 uCi	2,400 mrem	4,300 mrem <sup>b</sup>	220 mrem
Zr-95	.06 uCi	15 mrem	5 mrem	1 mrem
Nb-95	.06 uCi	5 mrem	1 mrem	0 mrem
Ru-103	.72 uCi	55 mrem	2 mrem	1 mrem
Ru-106	3.6 uCi	4,000 mrem	75 mrem	15 mrem
Cs-137	1.2 uCi	10 mrem	60 mrem	40 mrem
Ce-144	<u>7.2 uCi</u>	<u>7,000 mrem</u>	<u>5,600 mrem</u>	<u>350 mrem</u>
TOTAL	15.4 uCi	13,500 mrem	10,100 mrem	630 mrem

<sup>a</sup>Calculations based on standard man criteria shown in ICRP Report No. 2.

<sup>b</sup>The life time dose is estimated at 107 rem.

b. Probable

The probable effect of a fire in the 291-B building is considered to be the release of a portion of radionuclides from the first filter bank through the second bank, assuming that the integrity of the second filter bank is not affected by the fire. The estimated quantities of radionuclides which could be released are shown in Table V-D-1. This analysis is based on the following considerations:

- It is improbable that sufficient combustibles will be present to consume the first filter bank and subsequently consume the second filter bank.
- It is unlikely that any significant quantity of combustibles will collect between the first and second filter banks.
- The average size of radionuclide-bearing particles released during consumption of the first filter bank is assumed to be 0.3 micron, resulting in a removal efficiency of 99.97 percent for the second bank.
- It is probable that installed water sprays which are directed at the first filter bank can be used to extinguish the fire, thus preserving the integrity of the second filter bank.

NOTE: Although the second bank of filters may be ruined by the water sprays, its removal efficiency should not be affected. Exhaust ventilation flow should be cut back and diverted to the sand filter as quickly as possible if a fire is detected in the 291-B absolute filter building. Diversion of the flow by draining and filling appropriate water seals would require approximately four hours. Reported data indicate that absolute filter media will hold up for an hour or longer when subjected to saturated air containing entrained water droplets as from a spray nozzle.<sup>(31)</sup> This period of time should be more than sufficient to extinguish the fire using the spray system.

#### 4. Preventives

Procedures will be put in effect to periodically evaluate the inventory of radionuclides on the filters and reevaluate the hazards in light of the actual plant conditions. These evaluations will prevent an undetected buildup in radionuclide filter-loading which could be greater than the loading used for estimating the effects of the incident.

Preventives employed at B Plant to avoid a fire in the 291-B building are as follows:

- The 291-B filter installation employs absolute filters of the fire resistant type; i.e., filter frames, gaskets, etc. are selected such that they will not support combustion. See Hanford Specification HWS-7511-S for wood-frame high-efficiency particulate air-filters; Type "C" Fire Resistant (nonmetallic, moisture-resistant) filters are used exclusively.
- A fire detection system is provided for the canyon cells as described in Reference 32. This system will warn personnel of a cell fire. Cell sprays will be activated to extinguish the fire.
- The quantity of combustible materials is minimal in the canyon of the 221-B building with the exception of certain process chemicals as discussed in Section IV.C.4.
- Sources of ignition are minimized in the canyon. All electrical installations are compatible with the applicable codes for the types of materials being handled.
- A separate process vent system is provided to dispose of off-gases which could potentially contain ammonia and thus result in deposition of flammable ammonium compounds on the 291-B off-gas filters.
- Good housekeeping practices are to be followed in the canyon area to prevent a collection of combustible debris in the ventilation exhaust ducts and filter building.

#### E. Ruthenium Release Via Chemical Oxidation

Large quantities of radioactive ruthenium will be present in current acid waste feeds to B Plant. Ruthenium can be chemically oxidized by strong oxidants, such as sodium or potassium persulfate, to ruthenium tetroxide. Ruthenium tetroxide can be volatilized from an acid solution and carried into the ventilation system which, if not

contained, could result in a ruthenium release to the atmosphere. Oxidation and volatilization of ruthenium with persulfate occurs readily at ambient temperatures, but the reaction is more vigorous with a greater percent volatilized at elevated temperatures. While persulfate will not normally contact solutions containing significant quantities of ruthenium, the presence of persulfate in the building is the basis for postulating a ruthenium release. This is the only chemical used in the plant which is a sufficiently strong oxidant to initiate large sudden releases of ruthenium.

#### 1. Potential Cause

A solution of sodium carbonate-sodium hydroxide is routinely made up in the aqueous makeup area for introduction into TK-11-2. This tank does on frequent occasions contain up to  $10^6$  curies of Ru-103 and  $3 \times 10^5$  curies of Ru-106. Sodium persulfate could be added, through operator error, to the aqueous makeup tank in place of sodium carbonate. The two chemicals do not differ appreciably in appearance. Upon admittance of the persulfate solution to TK-11-2 containing the ruthenium in an acid solution, large quantities of ruthenium could be oxidized and volatilized, if the sodium hydroxide were omitted from the aqueous makeup.

#### 2. Energy Considerations

The chemical oxidation reaction is not violent even at elevated temperatures. If it is assumed that sufficient persulfate is added to completely oxidize ruthenium, that the temperature of the solution is between 30-40°C, and that the tank agitator is running, up to 30 percent of the ruthenium could be volatilized within a few minutes. The rate of ruthenium volatilization would not be sufficient to pressurize the tank.

#### 3. Path of Potential Radionuclide Release

The volatilized ruthenium would pass through the vessel vent system (as described in Section IV.C.1), into the air tunnel to the 291-B Building and out the 200-foot stack to the atmosphere.

#### 4. Probable Effects

The quantity of ruthenium which would ultimately be released to the atmosphere depends upon several factors. These are:

- Residence time within the vessel vent and building ventilation systems.
- Reducing potential and water vapor content of the atmosphere throughout the ventilation systems.

- Temperatures of the air flowing through the vessel vent systems.
- Material of construction of the vessel vent system piping and its reducing potential.
- Number of condensers, filters, or absorbers in the off-gas system.

The residence time of the ruthenium throughout the ventilation system is between two and three minutes at normal flows. This is sufficiently long that much of the ruthenium would be expected to be reduced and plate out on the walls of stainless steel pipes and ductwork in the ventilation systems. The presence of small concentrations of reducing agents in the atmosphere, such as nitrites and solvent vapors and a relatively low temperature (30-40°C) throughout most of the vent systems should aid in the reduction of ruthenium tetroxide to the non-gaseous state. A condenser, a primary filter, and four CWS absolute filters all in series provide the principal containment of the ruthenium. Experience in handling ruthenium has indicated a ruthenium DF of approximately  $10^3$  across a condenser(35) The Redox sand filter has routinely demonstrated ruthenium DF's of from  $10^2$  to  $10^3$ .(36) CWS-type absolute filters should yield a DF up to 10.

Considering these factors, the expected minimum overall DF for ruthenium is about  $10^6$ . Thus, less than one curie of Ru-103 plus Ru-106 would be expected to be discharged to the atmosphere.

##### 5. Preventives

The following safeguards are employed to prevent an accidental ruthenium release:

- Sodium or potassium persulfate is used in the process for separating cerium from other rare earths. This chemical is not used for waste management processing. Therefore, when processing current acid waste solutions or acid sludge waste solutions, solid persulfate or its solutions will not be kept within the building.
- Drums of sodium or potassium persulfate will be plainly and adequately labeled to deter accidental use of persulfate in place of sodium carbonate.
- Sodium carbonate is added to TK-11-2 after the majority (>95%) of the ruthenium has been removed from the tank.

- Sodium hydroxide is also present in the aqueous makeup. Ruthenium cannot be readily volatilized from a caustic solution.
- The aqueous makeup is sampled and analyzed to confirm that the sodium carbonate and sodium hydroxide concentrations are within flowsheet standards. Substitution of persulfate for carbonate should be detected by the laboratory.

Thus, five independent errors are required to put persulfate into TK-11-2 at a time when the tank contains large quantities of ruthenium:

1. Persulfate is left in the building.
2. Persulfate is substituted for sodium carbonate.
3. The sodium hydroxide is omitted from the makeup.
4. The analytical laboratory fails to identify the out-of-standards solution.
5. Persulfate (supposing it be carbonate) is added at the improper time in the processing sequence.

F. VESSEL COIL FAILURE

Vessel coils or heat transfer tubes are used in either heating and cooling service or for the single purpose of cooling. Since cooling service only is at least an order of magnitude less severe on the integrity of the heat transfer surface, the potential and severity of failure are minimized. Because the difference in probability of failure in the two types of service, i.e., alternating or cooling only, the effluent systems are segregated and disposal is by distinctly different systems. The heat transfer systems in the two services are analyzed independently in light of these differences.

1. Coil Failure - Heating and Cooling Service

a. Potential Cause

Coils and tube bundles required in alternate heating and cooling service may fail as a result of thermal shock, corrosion, mechanical vibration, or mechanical damage from the installation, operation, or removal of an agitator.

b. Path of Nuclide Release

Radionuclides entering a failed heat transfer coil or tube bundle would go with the effluent to crib 216-B-55 via a

condensate tank located outside building (see Figure IV-C-4).

c. Postulated Effects

(1) Possible

The amount of radionuclides released would depend on several factors including the concentration in solution, size of the hole in the heat transfer system, the driving force across the hole, and the time during which the liquid flowed through the hole. The following assumptions were made in postulating a discharge of radionuclides through a heat transfer surface used in alternate heating and cooling service:

- The pressure of the steam chest in the strontium concentrator has been reduced at a time when one-eighth inch diameter hole exists in a tube bundle.
- A leak in a tube bundle is postulated to occur in a strontium concentrator (E-5-2) containing 8,000 Ci/gal of strontium-90. The failure was postulated to occur in this vessel since it will contain the highest strontium concentration.
- The automatic air purge fails to function when the pressure drops in the tube bundle.
- The failure of the tube bundle has not been detected through evidence of faulty operation such as pressurization of a concentrator, reduction of specific gravity, change in volume, or loss in concentrator capacity.
- The reduced pressure continues for ten minutes before air, cooling water or steam pressurizes the coil or tube bundle, thus allowing 5 gallons of solution containing 40,000 Ci of strontium to be discharged to the crib. If the steam chest were completely filled with solution (270 gal.),  $2 \times 10^6$  Ci could be discharged.
- Minimum leakage occurs during continued service with pressure on the tube bundle.

The alarm at the dispatcher's office would prompt an immediate response to send personnel to monitor and verify the existence of a discharge of radionuclides. An analysis of recent equipment operations would be made while holding

all associated equipment in the current operational status. Equipment having the suspected failure would be isolated and the existence of a leak would be verified by a hydrostatic test or other conclusive evidence. The solution in the vessel containing the failed tube would be cooled and transferred to another tank. The crib would be abandoned as its radionuclide storage limit is reached<sup>(19)</sup> and an alternate crib would be placed into service.

(2) Probable

The probable effect of a radionuclide release from a failure in a heat transfer surface which is used for alternate heating and cooling is based on the following assumptions:

- The vessel, solution, and failure mechanism are similar to that above except that the low pressure situation continues for one-half minute, thus discharging one-fourth gallon of solution containing 2,000 Ci of strontium when pressure is restored.
- Corrective action would be taken, as stated above to minimize the discharge of radionuclides to the crib.

d. Preventives

Several measures have been taken to assure minimum risk of a failure in a heat transfer surface. These include:

- Quality assurance inspection of the equipment was made before use including hydrostatic testing of the coils or tubes.
- Operability tests are performed on the concentrators before they are placed into service.
- Frequent hydrostatic tests are made to evaluate the condition of the heat transfer system by blanking the discharge line during shutdowns.
- Instrumentation is installed to measure liquid level and specific gravity. Unexpected changes may be investigated to determine their cause.
- An automatic air pressure blow is established on the concentrator tubes if the pressure falls below a given value, thus maintaining a positive pressure on the tubes.

2. Coil Failure - Cooling Service Only

a. Potential Cause

Coils and tube bundles required in cooling service only may fail as a result of corrosion, mechanical vibration, or installation, removal or malfunction of operating agitator.

b. Path of Potential Radionuclide Release

The cooling water from storage vessel coils and others which require cooling service only leaves the building via sub-headers to a 15-inch cooling water effluent header past a radiation monitor through a diversion box to a 24-inch line to a retention basin and thence to an open pond (See Figure IV-C-3). When radionuclides trip the monitor alarm, a motor actuated valve at the diversion box opens automatically in adequate time to assure that all of the radionuclides are diverted to a specific retention trench. An alarm is located at the dispatcher's office together with manual control of the valve signal. If the valve should fail to function, the radionuclides would continue to a retention basin. From this point, it is possible to route the radionuclides to another specific retention trench.

c. Postulated Effects

(1) Possible

The amount of radionuclides released would depend on several factors including the concentration in solution, the size of the hole in the heat-transfer surface, the driving force across the hole, and the time during which the liquid flowed through the hole. The following assumptions were made in postulating a discharge of radionuclides through a tube used in cooling service:

- The internal pressure in a tube bundle of a strontium storage tank is reduced (flow is stopped) at a time when a one-eighth inch diameter hole exists in the tube.
- A leak is postulated to occur in a strontium storage tank containing 8,000 Ci/gal of strontium-90. This vessel was chosen since it is one that contains the highest concentration of strontium in storage.
- The failure has not been detected by a change in volume in the vessel.

- One-eighth of the tube bundle is filled with solution containing about 70,000 Ci of strontium-90 which is discharged to the cooling-water header when flow is restored. (If all of the tube bundle had been filled, about  $0.5 \times 10^6$  Ci of strontium-90 would be discharged to the header.)
- Minimum leakage occurs when flow is maintained through the tube bundle.

The alarm at the dispatcher's office would prompt immediate dispatching of personnel to monitor the cooling water and verify that a discharge of radionuclides had occurred. Further monitoring of the subheaders would take place until the source of radionuclides is found. Water discharged from the other subheaders would be valved to the 24-inch header to minimize the volume of water discharged to the specific retention trench. The individual headers would indicate in which cell the faulty equipment was situated. The solution in the faulty vessel would be transferred to other vessels and the equipment repaired or replaced. The retention trench has sufficient capacity to hold the equivalent of three hours of full flow of cooling water without percolation of the water into the ground. The isolation of the subheader carrying the radionuclides should take less than one hour and probably less than one-half hour. The retention trench would be promptly covered when the flow of radionuclides has stopped to prevent the spread of airborne contamination of strontium. If the diversion valve failed to function, each of the two retention basins has a 500,000 gallon capacity which would hold the full cooling-water flow for over three and one-half hours, making a total of over seven hours holding capacity. Under reduced water flow conditions, much more time would elapse before these were filled.

(2) Probable

The probable effect of a radionuclide release from a failed heat transfer surface used in cooling service only is based on the following assumptions:

- The vessel, solution, and failure mechanism are similar to that listed above except that the low pressure (no flow) situation continues

until one-eighth of one gallon containing 2,000 Ci of strontium-90 leaks and is discharged upon resumption of flow of water.

- Corrective action would be taken, as stated above to minimize discharge of radionuclides to the retention trench.

d. Preventives

Several measures have been taken to assure minimum risk of a coil failure. These include:

- Quality assurance inspection of the equipment was made before use including hydrostatic testing of the coils.
- Steam and air services are not connected to heat transfer surfaces on vessels requiring cooling service only. With positive pressure of cooling water on the heat transfer surfaces in the event of tube failure, gross leakage should be into the vessel.
- Operability tests are conducted before the equipment is placed into service, again evaluating the condition of the heat transfer surfaces.

G. Loss of Cooling Water

Many of the process solutions encountered in B Plant require continuous cooling for the removal of heat generated by decay of the radionuclides. This continuous cooling requirement is met by tank coils which exchange heat to a cooling water stream. The cooling water flowrates to individual tanks are adjusted manually to maintain desired temperatures of the tank contents. The cooling water is supplied to B Plant at about 100 psig from the 200-East Area raw water system through a single, buried, 12-inch line; the normal flowrate is expected to be about 2200 gpm.

1. Potential Cause

Loss of cooling water supply to the B-Plant tank coils could be caused by a break in the 12-inch line supplying B Plant from the 200-East Area raw water supply system. Such an event could result from deterioration of the line or some violent physical shock, e.g. earthquake.

2. Path or Potential Radionuclide Release

Upon extended loss of cooling, process solutions containing sufficient quantities of radioisotopes would heat up to the boiling point and evaporate to dryness. The resulting dry salts

would continue to heat up and in some cases it has been postulated that local temperatures could be reached which would cause vaporization of certain radioactive compounds. These vapors would be condensed and solidified by the lower temperature surroundings, thereby forming an aerosol dispersed in the canyon air with the potential for atmospheric release. The path of potential release for this incident is therefore from tank, to process vent system, to canyon atmosphere (air tunnel), to 291-B filters, to stack, to atmosphere.

3. Postulated Effects

a. Possible

A severe effect upon loss of cooling water is considered to be the atmospheric release of all dispersed radionuclides through only two stages of absolute filters. This assumes that either the two-stage process vent filter is bypassed or that the two-stage building ventilation filter is breached. This is considered to be a conservative assumption since there is no postulated event caused by a loss of cooling water which would affect the integrity of the filter system.

The quantities of radionuclides released to the atmosphere from this postulated incident are as follows:

<u>Radionuclide</u>	<u>Quantity Released</u> (Curies)
Cs-137	3
Sr-90	7
Ru-103 and 106	3

These values are based on the following conservative assumptions:

- Where vaporization temperatures are reached, it is assumed that all of the radioactive material present is vaporized.
- Upon vaporization and subsequent condensation and solidification, it is assumed that all of the radioactive material is dispersed as an aerosol and that all of the particles reach the filter.
- The average particle size is assumed to be 0.3 micron, resulting in a  $1.1 \times 10^7$  decontamination factor for two stages of filtration.

b. Probable

The probable effect of loss of cooling water supply to B Plant would not involve the release of radioactivity to the environment. This determination is based on the following conclusions:

- There are two back-up, cooling-water supply sources for emergency use if the raw water supply fails (see discussion of preventives). The probability of losing the primary and both back-up systems simultaneously is considered remote.
- The length of time between loss of water and realization of temperatures which could result in environmental release is a minimum of one and one-half days, sufficient to permit restoration of water supply or implementation of emergency procedures.
- Even with the total loss of all cooling water supply, the incident can be kept under control as discussed in the following paragraphs:

For the "hottest" solutions handled in the plant, containing up to 250 Btu/hr-gal., 2.3 hours would be required to reach the boiling point and 33 additional hours would be required to evaporate to dryness, neglecting all heat losses from process vessels. Thus a minimum of one and one-half days are available to restore the cooling water supply. In the event that the supply can't be restored, water could be trucked to B Plant and added to process vessels to prevent the contents from boiling to dryness. Make-up water would be required at the rate of about 15,000 gallons per day based on projected plant inventories. This quantity of water could be trucked from the river with available equipment under emergency conditions. The condition of boiling in many of the process tanks results in pressurization of the process vent system and vessels. At the projected tank inventory levels, pressures could approach the 5 psig design pressure of the tanks. The tanks would not be ruptured, and although a serious plant contamination problem could occur, release to the environment would not take place.

4. Preventives

The features which exist to prevent loss of cooling water supply to B Plant are summarized with regard to the primary and back-up systems.

a. Primary Supply - Raw Water

- All pumping stations maintain steam driven pumps on standby to meet emergency requirements in the event of power failure.
- Parallel supply lines, each capable of emergency requirements, exist between the river and the 200-East raw water supply system.
- A 3.0 million gallon reservoir in 200-East Area provides surge and storage capacity.
- The 221-B Building raw water header is supplied at both ends; either end of the "loop" is capable of handling emergency requirements.

b. Back-up Supply Systems

- Two emergency wells, provided at B Plant, supply water to diesel driven pumps which are connected to the building raw water supply system. This system is designed to provide in excess of 500 gpm of emergency water at 100 psig<sup>(37,38)</sup>, and is activated automatically upon loss of pressure in the raw water system. One well is completed, the other is presently under construction.
- The 221-B Building raw water header can be supplied from the sanitary water header via hose bibs and backflow preventers. The buried, 8-inch sanitary water supply line provides a parallel route from the 200-East Area supply station to B-Plant.

H. Loss of Electrical Power

The Bonneville Power Administration supplies power to the Midway Station and hence to the 251 Sub-station over 230 KV dual lines. From the 251 Sub-station, power is delivered over two 13.8 KV lines, C8-L7 and C8-L8, to the C8-S3 (252-E) Sub-station (see Figure V-H-1).

Electrical power is furnished to B-Plant over two normal 2400-volt feeders, E8-L53 and E8-L58, from Sub-station C8-S3 (252-East). These lines feed 2400-volt switch gear in the 221-B electrical gallery. From the switch gear, power is fed to 2400 V/440 V and 2400 V/110 V transformers for equipment and lighting service.

Since January, 1960, there have been 10 outages lasting for a total of one hour and 46 minutes on the main feed system to 252-E Sub-station. The longest duration was for 44 minutes occurring May 25, 1961.

NOTES:  
1. THE LETTERS AR WITHIN A BREAKER INDICATE AUTO-RECLOSEURE.  
2. THE LETTERS AT WITHIN A BREAKER INDICATE AUTO-TRANSFER.

SUPERSEDES  
DWG NO. H-2-358  
5-17-57  
7-24-59

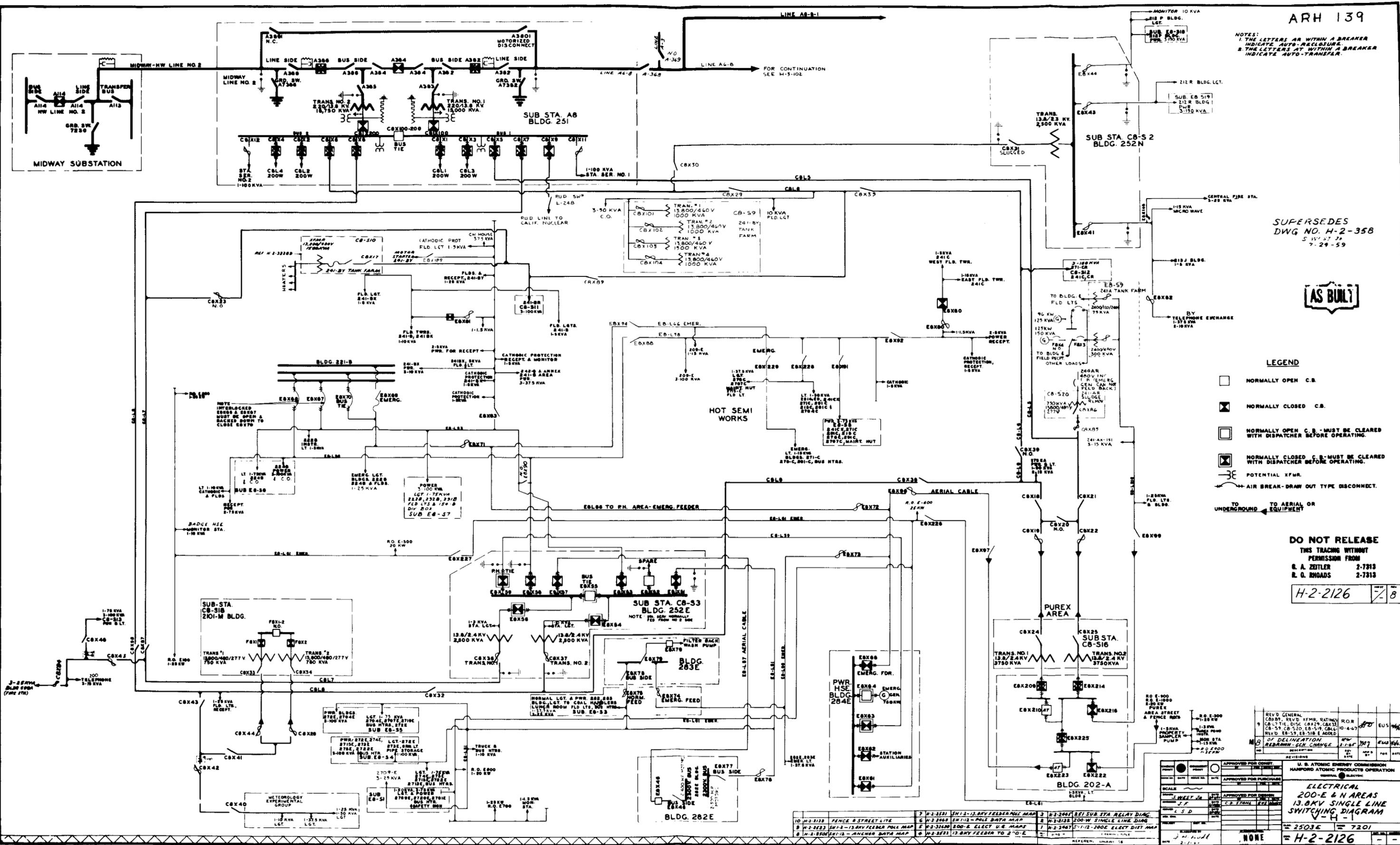
AS BUILT

LEGEND

- NORMALLY OPEN C.B.
- NORMALLY CLOSED C.B.
- NORMALLY OPEN C.B. - MUST BE CLEARED WITH DISPATCHER BEFORE OPERATING.
- NORMALLY CLOSED C.B. - MUST BE CLEARED WITH DISPATCHER BEFORE OPERATING.
- POTENTIAL XFMR.
- AIR BREAK-DRAW OUT TYPE DISCONNECT.
- TO UNDERGROUND
- TO AERIAL OR EQUIPMENT

DO NOT RELEASE  
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R. A. ZETTLER 2-7313  
R. G. RHODS 2-7313

H-2-2126 8



NO.	DATE	DESCRIPTION	BY	CHKD.	APP'D.
10	11-2-57	FENCE & STREET LIGHT	J.P.		
9	11-2-57	13.8KV FEEDER POLE MAP	J.P.		
8	11-2-57	200-E ELECT. U.S. MAP	J.P.		
7	11-2-57	200-E ELECT. U.S. MAP	J.P.		
6	11-2-57	13.8KV FEEDER TO 200-E	J.P.		
5	11-2-57	13.8KV FEEDER POLE MAP	J.P.		
4	11-2-57	13.8KV FEEDER POLE MAP	J.P.		
3	11-2-57	13.8KV FEEDER POLE MAP	J.P.		
2	11-2-57	13.8KV FEEDER POLE MAP	J.P.		
1	11-2-57	13.8KV FEEDER POLE MAP	J.P.		

REV'D GENERAL  
C8-S-16, REV'D R.F.M.R. RATING R.O.R. 10-4-57  
C8-S-16, DISC. C8-S-16, C8-S-16, 10-4-57  
C8-S-16, C8-S-16, C8-S-16, 10-4-57  
REV'D C8-S-16, C8-S-16, 10-4-57  
OF DELINEATION  
REVISIONS

U. S. ATOMIC ENERGY COMMISSION  
HANFORD ATOMIC PRODUCTS OPERATION  
ELECTRICAL

ELECTRICAL  
200-E & N AREAS  
13.8KV SINGLE LINE  
SWITCHING DIAGRAM  
H-2-2126

1. Potential Cause

The most likely causes of failure of the Bonneville supply are climatic orientated such as high winds or lightning.

2. Path of Radionuclide Release

The containment of radionuclides in the primary B-Plant process system is achieved by a single-pass ventilation exhaust system, which draws outside air into the canyon, through the cells, into the wind tunnel, through filters and exhausts to a 200-foot stack. Containment is supplemented by a vessel off-gas exhaust system which draws the off-gas through an air cleaning and filtering process, prior to combining with the main exhaust air.

3. Postulated Effects

a. Possible

A loss of power coupled with a simultaneous complete loss of steam would cause a breakdown in the basic containment provision for the B-Plant process system.

In the event of a power failure, one boiler will supply the steam loads for operating the fire and sanitary and raw water pumps, the emergency power steam turbine-generator, the steam turbine exhaust fans, and the process off-gas exhaust jets.

Since 1944, the steam generation and distribution lines in 200 East Area (see Figure V-H-2) have been in use without loss of service, accumulating a total of 210,000 hours. This record exceeds the reliability established for the electrical power.

Even in the unlikely event that both electric power and steam supply are lost, there is no reasonable mechanism whereby radionuclide would be released to the environs.

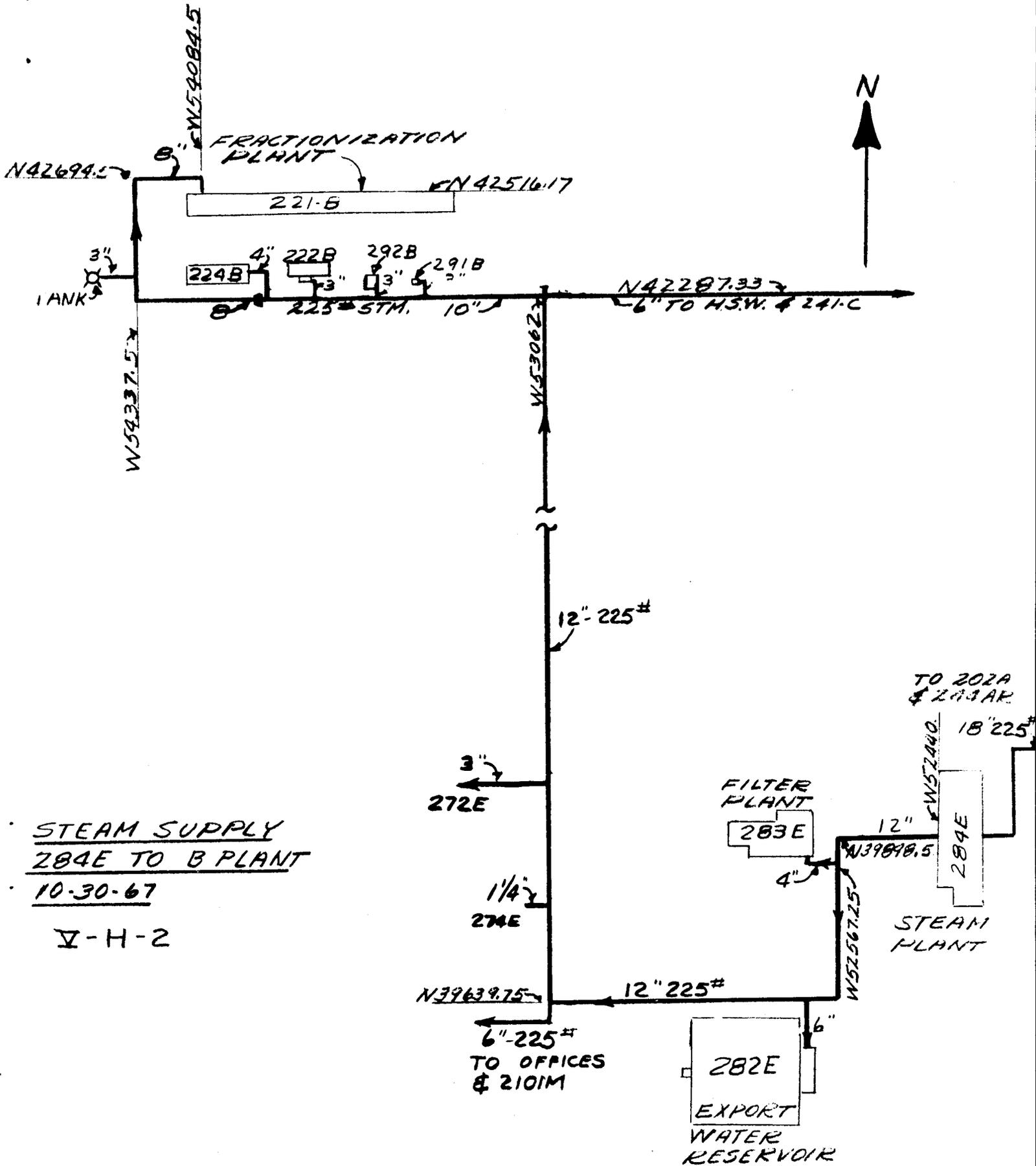
b. Probable

An electrical outage from an interruption in the 2400 volt normal power will result in the activation of emergency equipment and the initiation of emergency procedures.

It is not expected that an electrical outage would result in a release of radionuclides to the environs.

4. Preventives

The emergency backup systems which are installed to maintain



STEAM SUPPLY  
284E TO B PLANT  
 10-30-67  
 V-H-2

operating continuity of the critical systems are discussed in the following:

a. Steam Turbine Generator Emergency Power

An emergency 2400-volt line, E8-L66, to B-Plant is fed from a breaker at 284 E Powerhouse. In the event of a failure, originating on the 13.8 KV lines C8-L7 and C8-L8, or on the 2.4 KV lines, E8-L53 and E8-L58, an emergency steam turbine-generator automatically takes over the load on the emergency line. Allocation of the various operating equipment in B Plant which are authorized to be on this line is shown on the following table:

TABLE V-H-1

B-PLANT  
EQUIPMENT ON EMERGENCY GENERATOR

- 1) Canyon door latches.
- 2) Instruments serviced by regulated voltage circuit.
- 3) Instrument air compressor, 60 HP, and one Pennsylvania process air compressor, 60HP.
- 4) Evacuation bus heater.

b. Load Switching

In the addition to the emergency 2400 volt power, if any one of the dual, normal 2400 volt lines fails, its load can be switched at the 440 volt level to the other line. In this event, at the present time, noncritical equipment must be left out of service, since either one of the feeder breakers at 252-E sub-station is not sized to carry both loads. However, a new line is being installed in 200-E Area that will relieve this restriction and will enable either one of the incoming B-Plant feeders, E8-L53 or E8-L58, to carry the total B-Plant load.

c. 291-B Ventilation Exhaust Fans

The ventilation system for 221-B, a one-pass airflow, is exhausted to a 200-foot stack by two centrifugal fans, each rated at 37,500 scfm. The fans are driven by 100 HP electric motors. A steam-driven fan, which also has a capacity of 37,500 scfm, serves as an emergency standby in the event of an electrical outage or mechanical failure of the electric fans. A vacuum control on the inlet duct automatically starts the steam turbine should the vacuum within the duct drop below 0.2 inches of water.

d. Process Vessel Agitators

All process vessels, except centrifuges, columns, and towers, are supplied with cooling coils and agitators for the dissipation of fission product heat release. However, the fission product inventories are maintained so that the temperatures without agitation of the solution will be sufficiently low to be transferred by steam jetting.

e. Centrifuge Cake Removal

The fission product heat release from the precipitate cake in the centrifuge bowl is dissipated without a cooling coil. The temperature of the cake is kept under control by limiting batch size and the residence time in the centrifuge bowl. The cake dissolution is performed by pumping solutions through bowl sprays, pumping from feed tanks, and rotating the centrifuge slowly. Without power, the cake removal procedure will be interrupted. During power outages, the bowl will be kept full of water, through gravity additions of water from gallery head tanks to maintain control of the residual cake in the centrifuge bowl.

f. Ion Exchange Column Feed Pump

The ion exchange column T-18-2 is the only vessel which operates normally under pressure. In this condition, the chemical addition lines from the cold side to the column are removed and the nozzles blanked, using the 75-ton crane, and there is no physical means by which solution can be forced back to the operating gallery. Cooling of the T-18-2 column is controlled by limiting the fission product inventory and by the normal process flow through the column. In the event of a power failure, the process flow would be interrupted and the 75-ton crane would not be available for installing a chemical addition line from the operating gallery. In two days, the contents of T-18-2 could boil and in about three weeks could evaporate to dryness. The probability of an electrical outage lasting for this period of time is essentially zero. Solution may be added to the T-18-2 exchange column from the 18-3 feed tank by means of the 5 HP pump. If it becomes necessary, this pump could be powered by the electrical emergency system. Water can be added to the 18-3 tank by gravity flow.

g. Cooling Water Diversion to Specific Retention Trench

The motor-operated valve that diverts cooling water effluent from the retention ponds to a specific retention trench, upon a signal from a radiation detection instrument, is tied into the emergency 2400-volt power source through the regulated voltage circuit.

#### h. Instrumentation Circuits

Radiation detection and health monitoring systems, with their associated alarms, annunciators, and recorders are connected into the emergency 2400 volt power line E8-L66. This line, in the event of a power failure, is supplied by the emergency steam-turbine generator 284-E powerhouse. The detection and monitoring systems include the following:

- Cooling water effluent.
- Steam condensate effluent.
- Process condensate effluent.
- Stack monitor (ventilation exhaust).
- Canyon air monitors.
- Operating gallery monitors.
- Cell leak detectors.

#### i. Storage Battery Power

A separate, emergency lighting circuit, sparsely spaced throughout the building, is powered by a 100 ampere-hour battery system and will be energized automatically in the event of a failure of the 2400-volt normal power sources. The lighting draws a 2-ampere load and the batteries will endure for about two days.

The door lock controls for the various canyon deck entrances also are tied into a battery system.

### I. Loss of Instrument or Process Compressed Air

Maximum compressed air requirements for B Plant are 350 scfm of process air and 150 scfm of instrument air, both at ambient temperatures. The process air is supplied by two single-stage Pennsylvania compressors having a combined capacity of about 600 scfm at 90 psig. A third, four-stage, Worthington compressor has a backup capacity of approximately 700 scfm at 90 psig. The instrument air will be supplied by a new oil-free compressor having a capacity of about 300 scfm at 90 psig. The compressors are arranged to start automatically and to hold respective discharge header pressures within a 5 psig differential over a wide range of demand. Process and instrument air are used in the plant for the following process control items:

- Aqueous solutions containing fission products generate hydrogen gas by radiolysis. About 1-2 scfm of dilution air is provided per tank to maintain the vessel atmosphere below the lower explosive limit. About 80 scfm is taken from the process air system for this purpose.
- Process air is required for purging gang valves and supply lines to the steam jets after usage to prevent their filling with process solution, as a result of the vacuum caused by condensed steam.

- Process air is used as a backup for the steam supply to the exhaust jets on the two vessel-vent systems. These systems are provided to contain and decontaminate the process off-gases prior to discharging to the environs.
- Process air is used for operating the process solution recirculation jets located in the sampling ports of the canyon deck.
- Instrument air is used for air purging the weight-factor and specific-gravity dip tubes, for controlling diaphragm-operated valves, and for a variety of pneumatic instrumentation including pressure indicators, recorders, recorder-controllers, transducers, transmitters, etc.

1. Potential Cause

Loss of air supply could be caused by a complete electrical power failure, a mechanical breakdown of the compressors, or a break in the main distribution line from the compressed air distribution tanks.

2. Path of Radionuclide Release

In the event of a complete loss of process and instrument air supply, radionuclides could move from canyon process equipment to the non-regulated operating and pipe galleris, e.g., suck-back in steam lines to jets. The abnormal radiation levels would be detected by radiation monitoring surveillance equipment which alarms to alert personnel.

3. Postulated Effects

a. Possible -

It is possible that an estimated volume of one gallon of the highest radiation content solutions could be transferred by a suck-back into the pipe or operating galleries associated piping and equipment with a resultant radiation dose rate of several thousand R/hr near the source. Personnel would be evacuated from adjacent areas and emergency procedures would be evoked. Personnel access would be permissible if required for operations essential to plant safety. No radionuclides would be released to the environs. Work would be initiated to restore the area to normal radiation levels. Procedures would involve flushing, shielding, and removal of affected systems.

b. Probable

The loss of process and instrument air would probably result in the movement of trace quantities in the order of a few milliliters of solution from canyon process system to the non-regulated

operating and pipe gallery piping and associated equipment. Effective radiation levels for the most highly radioactive solutions would result in dose rate in R/hr levels.

The change in dose rates would be detected and personnel would be alerted by surveillance radiation detection instrumentation. Procedures would be initiated to restore the areas to normal background radiation dose rates. Plant personnel would not be in jeopardy and there would not be a release of radionuclides to the environs.

#### 4. Preventives

The features which are provided to prevent loss of instrument and process air are summarized as follows:

- The new 60 HP instrument air compressor and the two 60 HP Pennsylvania air compressors are fed by the emergency 2400-volt Power line, E8-L66, that in event of an electrical outage, is supplied from the 200-E emergency steam turbine-generator.
- The four compressors that will be in service have a reserve capacity of about 600 scfm at 90 psig. This provides a back-up in the event of mechanical failure and also allows ample preventive maintenance.
- The two large air receiver tanks for both the process and instrument air are intertied, with isolating valves, to serve as back-up for the other pair in an emergency.
- Provisions are made for hooking the compressed air systems to a diesel-driven portable air compressor to give additional back-up.

The following plant operating procedures and system design features are established to assure safety:

- In the event of a complete air outage, all processing operations would be suspended. All electrical motors located in the process cells where hydrogen generation by radiolysis is possible would be shut down to minimize electrical sparking. The motors would not be restarted until after the air purging had been resumed and continued sufficiently to reduce the hydrogen in the vessel atmosphere below flammable concentrations. All remote maintenance operations in the process cells would be discontinued.
- Since the primary ventilation exhaust fan damper settings at 291-B are controlled by pneumatic instrumentation, arrangements are provided for a fail-safe condition in the event of loss of instrument air supply. The electric motor driven exhaust fans are stopped, and their dampers closed. The emergency steam turbine exhaust fan automatically is started.

- All diaphragm-operated valves have been engineered and arranged to be left in a fail-safe position in the event of loss of instrument air supply.

J. Temperature Excursion, TK-11-2

Prior to solvent extraction processing, current acid wastes (PAW and ZAW), are treated to remove solids and to recover the strontium and rare earths associated with the solids. The treatment process carried out in TK-11-2 contains three steps during which solids, containing large quantities of radionuclides, are allowed to settle and the liquid is decanted from the tank. The heat generated by the solids layer could produce local temperatures high enough to vaporize radionuclides such as cesium and ruthenium, if proper controls were not maintained.

1. Potential Cause

The postulated cause of a temperature excursion in TK-11-2 is failure of engineering safeguards coupled with improper operation, i.e. permitting the solids layer in TK-11-2 to dry out by failure to promptly add liquid solution for initiation of the next process step.

2. Path of Potential Release

The path for the potential release of vaporized radionuclides from a TK-11-2 temperature excursion is from tank, to the No. 1 Vessel Vent System, to the air tunnel, to the exhaust filter system and stack.

3. Postulated Effects

It may be possible that local temperatures could be achieved which would vaporize radionuclides in the solids layer of TK-11-2. A maximum solids temperature of 1200°F has been calculated for a temperature excursion in a solids layer resulting from a "flowsheet" batch. In this event, the quantities of radionuclides released to the atmosphere are estimated to be as follows:

Ru-103 and 106	<u>1</u> curie
Cs-137	<u>1</u> microcurie

This analysis is based on the following assumptions:

- A D.F. of  $10^{14}$  is realized for cesium compounds. This assumes all the cesium vapors are condensed and solidified producing an aerosol which is then subjected to four stages of absolute filtration prior to discharge from the stack.
- A D.F. of  $10^6$  is realized for ruthenium compounds. (See Section V.E)

The probable effects of this incident are difficult to distinguish from the possible effects mentioned above for the following reasons:

- The actual geometry assumed by the settled solids would be expected to approach a uniformly thick layer; however, if the majority of the solids settle rapidly, a much larger concentration of solids might occur at the perimeter of the tank bottom due to the vortexing action resulting for a short time when the agitator is turned off.
- Meaningful predictions of the heat transfer from the solids layer to that portion of the cooling coil which is in contact with the solids are not possible because of the many unknowns.

#### 4. Preventives

The following preventives will reduce the chances of occurrence for this incident.

- A small liquid heel will be left in TK-11-2 by the physical elevation of the jet suction used for decanting. This quantity of liquid ensures that the solids will not reach dryness for a finite period of time (about an hour for a "flowsheet" batch size).
- Temperature instrumentation and a water add system is provided to automatically "dump" about 70 gallons of water into TK-11-2 if excessive temperatures are encountered.
- Careful administration of technical specifications and operating procedures will result in an awareness of this potential hazard and will outline the necessary practices for safe control of the process.

#### K. Cell Drain Line Failure

Process solutions which leak from wall nozzles, leaking tanks, etc. into the cells flow through 6-inch diameter vertical drains located at the low points of the cells into a 24-inch I.D. vitrified clay pipe which traverses the length of the canyon below the cells. Leaks in the hot pipe trench also flow into the cell drain line. The line is embedded in concrete between 3 and 9 feet below the cell floors. There is a minimum of 12 inches of concrete thickness on either side of the pipe and below the pipe. Soil surrounds the concrete encasement. The pipe drains with a 1% slope from either end of the canyon into the deep cell (Cell 10) collection tank. Solution from this tank is jetted to the waste handling system and, if required, is reprocessed for fission product recovery.

It is probable that solutions containing high concentrations of fission products will enter the cell drain system, and if the cell drain pipe were

breached, the solution could leak out of the pipe, through voids in the concrete and into the soil.

1. Potential Cause

The cell drain line could lose integrity by physical deterioration or uneven settling of the 221-B Building, causing caulked joints to open or cracks to form. Solution containing high radionuclide concentrations could enter the cell drain system from leaking vessels or transfer systems including the hot pipe trench.

2. Energy Considerations

Assuming that 150 gallons of 8000 Ci/gal strontium-90 solution entered the drain system, about 28,000 Btu/hr of decay heat would be generated by this solution.

3. Path of Potential Radionuclide Release

Normal: From cell floor to cell drain line to deep cell tank, TK-10-1.

Probable: From cell floor to cell drain line to deep cell tank, TK-10-1. with a small portion (1 gallon) leaking out of small cracks in the pipe, through the concrete encasement into the soil.

Possible: From cell floor to cell drain line, through a large break in the line and the concrete encasement into the soil.

4. Postulated Effects

a. Possible

The strontium solution leaving the pipe would be absorbed by the soil below the concrete drain line encasement. Temperatures in the soil and concrete could approach 2000°F after two or three weeks.<sup>(39)</sup> This temperature would be adequate to cause dehydration of the concrete resulting in further deterioration of the concrete encasement and line integrity. Subsequent transfer of low-heating solutions through the line could cause the "hot spot" to move further away from the line. Once a "hot spot" has been formed, if organic solvent leaked into the area, an organic explosion could occur.

b. Probable

A small leak (1 gallon) through a crack in the cell drain line would probably not yield sufficiently high temperatures to cause further line deterioration. However, succeeding transfers of fission product solution through the line could cause an accumulation of significant quantities of fission products in the concrete or soil. After a sufficient quantity of solution has leaked in to the soil,

elevated temperatures could be obtained resulting in line deterioration, thereby, increasing the potential for larger leaks during subsequent passages of solution through the line. It is unlikely that radionuclides would be released to the atmosphere.

### 5. Preventives

The following safeguards are provided for minimizing the amount of solution which enters the cell drain system and for assuring line integrity:

- All new transfer routes are leak checked prior to solution transfer through them.
- Leak-detectors are installed in the openings of the cell floor drains. The leak detectors are capable of detecting less than 1 cup of solution which enters the drain. An alarm will alert personnel so that corrective action can be initiated to stop the flow of liquid to the drain.
- A quarterly leak check of the cell drain line is made to determine the line integrity. A volumetric measurement technique is used.
- The volume of solution in process vessels is maintained less than 80 percent of capacity to limit the potential for overflowing a vessel. High and low level alarms are installed on all tanks to warn of high or low volume levels.
- Solution transfer volumes are verified from shipper and receiver tank volume measurement systems. Volumes shipped and received are recorded and the records are maintained on a continuing basis.
- Each time cell cover blocks are removed a visual inspection is made to look for evidence of leaks. Leaks found are eliminated.

### L. Concentrator Pressurization

Three concentrator vessels are provided in B-Plant for the evaporation of process solutions. Data for these units is presented in the following table:

TABLE V-L-1

	<u>B PLANT CONCENTRATORS</u>		
	Strontium Product Concentrator E-5-2	Low-Level Waste Concentrator E-23-2	Cesium Product Concentrator E-20-2
Heat Duty (Btu/hr)	$4.5 \times 10^6$	$9.0 \times 10^6$	$30 \times 10^6$
Working Vol. (gal)	250	250	3,000
Concentration of Radionuclides (Curies per gal)	8000 Sr-90	7800 Cs-137	See Reference 15

These vessels will contain boiling process solutions with high concentrations of radionuclides and are subject to pressurizations which could expel process solutions from the vessels.

1. Potential Cause

The most likely cause of concentrator pressurization is improper operations; e.g., failure to have cooling water valved to the condenser before starting boil-off. Pressurization resulting from an explosion or violent chemical reaction is treated as a separate incident in Section V.C.

2. Path of Potential Radionuclide Release

The concentrator vessels are protected by seal pots which vent the vessels to the air tunnel if the pressure or vacuum in the vessel exceeds the static head of the water seal. Once the seal is "blown", vapor containing entrained radioactive process solution is routed to the air tunnel via the seal pot discharge line until the cause of the pressurization is corrected and the seal reestablished. From the air tunnel, the entrained radionuclides would be exhausted through the exhaust ventilation filter to the stack and released to the atmosphere.

3. Postulated Effects

The quantities of radionuclides released from the strontium and cesium concentrators, E-5-2 and E-20-2, respectively, have been estimated for a pressurization incident which "blows" the concentrator seal pot. The quantity of material which might be released by pressurization of the low level waste concentrator is much smaller and is therefore neglected. As stated earlier, the incident considered here is a pressurization which is relieved by the seal pot and the integrity of the vessel is therefore not affected.

The possible and probable effects shown in Table V-L-2 were determined based on the following conservative assumptions:

- The concentration of radionuclides in the seal pot liquid is taken as the normal distillate concentration, although the only liquid routinely added to the seal pot is water.
- All liquid in the seal pot is assumed to be expelled to the air tunnel with 0.1 weight percent as droplets smaller than 100 microns.
- All radionuclides in the vapor stream are assumed to be entrained liquid droplets of less than 100 microns diameter. It has been shown, using Stoke's Law, that liquid droplets larger than 100 microns will settle out before reaching the ventilation filter.

a. Possible Effect

It is considered possible that pressurization occurs simultaneously

with a process upset (severe foaming, for example) such that a D.F. of only 100 is obtained between overheads and bottoms. For this case, the estimated quantities of radionuclides released to the atmosphere are shown in Table V-L-2.

b. Probable Effects

The probable effect as shown in Table V-L-2 is the result of pressurization at a time when a normal minimum D.F. of  $10^3$  is obtained between overheads and bottoms.

4. Preventives

The following provisions are made to prevent pressurization of the concentrators:

- Concentrator pressure and seal-pot liquid level are measured and prominently displayed.
- Operating procedures instruct personnel in the correct methods of concentrator operation.
- A gas eductor (jet) is provided to overcome the pressure drop in the vapor ducts and condenser and to maintain a slight vacuum on the concentrator vessel; the motive fluid supply is switched automatically from air to steam if the air supply fails.

TABLE V-L-2

POSTULATED EFFECTS OF CONCENTRATOR PRESSURIZATION

	<u>Strontium Product Concentrator (E-5-2)</u>	<u>Cesium Product Concentrator (E-20-2)</u>
Seal Pot Volume (gal)	50	50
Seal Pot Concentration (curies/gal)	1.7 Sr-90	7.0 Cs-137
Bottoms Concentration (curies/gal)	1,700 Sr-90	7,000 Cs-137
Boil-off rate (lbs/hr)	4,500	9,000
<u>POSSIBLE EFFECT:</u>		
Initial Release (curies)		
to air tunnel	320 Sr-90	350 Cs-137
to atmosphere	$30 \times 10^{-6}$ Sr-90	$3.2 \times 10^{-5}$ Cs-137
Subsequent Release Rate (curies/min)		
to air tunnel	600 Sr-90	1,300 Cs-137
to atmosphere	$6 \times 10^{-5}$ Sr-90	$1.1 \times 10^{-4}$ Cs-137
<u>PROBABLE EFFECT:</u>		
Initial Release (curies)		
to air tunnel	320 Sr-90	350 Cs-137
to atmosphere	$30 \times 10^{-6}$ Sr-90	$3.2 \times 10^{-5}$ Cs-137
Subsequent Release Rate (curies/min)		
to air tunnel	60 Sr-90	130 Cs-137
to atmosphere	$6 \times 10^{-6}$ Sr-90	$1.1 \times 10^{-5}$ Cs-137

VI. PROTECTION SYSTEM RELIABILITY

A. Emergency Power System

1. General

The emergency generator is a steam driven 750 KVA unit automatically starting on a power failure. This unit for the past 20 years has been reliable, has never failed to start and come on the line during a power failure.

It has on occasion during the weekly test runs failed to function properly. At no time has this faulty operation failed to the extent that attending personnel could not have corrected if an emergency demand had been on the system.

An emergency generator failed in the 200-West Area August 5, 1958<sup>(10)</sup> During a start-up test run, the unit accelerated to destruction. The investigating committee determined that the destruction was caused by failure of the governor and the overspeed trip.

There are five other similar separate installations on the project with no similar failure experience, these units have all been in service for over 20 years.

Experience indicates that a complete emergency generator failure is unlikely.

2. Load on the System

The present connected load on the emergency generator is 725 KVA. The B-Plant portion of this load is 75 KVA. The generator will assume the load under normal sequencing of load accumulation during an emergency start-up.

3. System Condition

The physical condition of the emergency electrical distribution system is good. To assess the reliability of the circuit to B Plant, it is necessary to divide into sections:

- The section from the powerhouse to 4th Street, continuing down 4th Street west to the intersection of 4th Street and Baltimore is all multiple systems consisting of normal feed, emergency, telephone, and fence lights. A failure of one system may interfere with the others.
- The section from the intersection of 4th and Baltimore to the intersection of 7th and Baltimore is a single-line system subject only to the normal traffic hazards of the adjacent roadway and weather elements.

- The section from the intersection of 7th and Baltimore past 222-B and into 221-B is considered the most vulnerable. It is a multiple system structure heavily loaded and in a high-density traffic area. The impact from a traffic accident could knock down a single pole and interrupt the normal feed, the emergency back up, and the phone system services. A gross failure of any of the feed systems would cause extensive damage to the others and result in an outage duration of a couple of hours to a full day depending on the time of the incident and the nature of the damage.

B. Water Systems

The primary (raw water) and the secondary (sanitary) water are mentioned in Section V.G. A review of the system, sectionalizing valves and other pertinent features leaves the following areas for hazard consideration:

1. The 282 Pump Room

A rupture of any of the pump cases, pipe sections, or valve bodies could flood the floor and, if extensive, be beyond the capability of the drain system. This condition, a rare possibility, would stop all water pumping and leave only the reserve sanitary system until such time as a crossover from the export system could be tied into the raw water system.

The 42" drainage line from the 282-E, 283-E, 284-E power facilities could become plugged and cause water to back up into pump rooms of the 282-E, 283-E Buildings, thereby stopping all electrically-driven pumps and eventually the turbine-driven pump if the water were allowed to back up.

Such an incident occurred and damaged several pump motors, action has been taken to clean out the discharge ditch which caused this back up. Periodic surveillance has been established to prevent accumulation and build up of silt in the 42" line to assure reliability of this system.

2. Water Pipe Lines

The raw and sanitary water lines to B Plant are in good condition; and there are no areas of significant vulnerability to damage.

3. Emergency Well Water

The self-starting engine-driven deep-well pump located west of B Plant will supply B Plant with emergency water. A reverse flow of water through the original line is prevented by a check valve. The recently installed raw water line to the east end of the building does not have a check valve and would require a prompt manual valve closing to direct the well water to B Plant.

C. Turbine Driven Exhaust Fan

The turbine-driven exhaust fan and the associated ventilation exhaust system is considered a protection system.

The unit is a Clarage Fan, 37,500 cfm @ 1,900 RPM with a lift of 15.4" H<sub>2</sub>O. The turbine is a Terry 113 HP quick start unit capable of starting in 0°F temperature after standing still for 10 days.

This type of turbine and its connection to the system have been found very reliable, and this unit is considered adequate back up for the normal system.

D. Instrument Air System

The instrument air system is considered a protection system and is described in Section V.I.4. A review of the instrument air system, and its coupling with the process air system, indicates high reliability. In addition, operability is further assured by emergency electrical power, as further back-up gasoline driven portable equipment can meet emergency air requirements.

VII. PROTECTION AGAINST MULTI-DISABILITY OF PROTECTION SYSTEMS

The potential for a multi-disability of B Plant protection systems is considered remote with the exception of the potential for a pole line structure failure from lightning or other impact. Protection from such elements can best be accomplished by separate structures spaced for safety. The review revealed no other significant multi-disability hazards of concern.

Listed below in Table VII-1 are the unscheduled outages since January 1, 1960 (with duration and cause) on the normal 2400 volt feeders E8-L53 and E8-L58 to the 221-B Building.

TABLE VII-1  
UNSCHEDULED ELECTRICAL OUTAGES

<u>Date</u>	<u>Time</u>	<u>Duration</u>	<u>Feeder</u>	<u>Cause</u>
7-12-60	10:13 A	4 Minutes	E8-L53	Switching Error
3-21-61	12:07 P	Momentary	E8-L58	Operating Error
5-25-61	6:49 P	44 Minutes	E8-L53, E8-L58	Lightning
8-22-61	3:54 P	1 Minute	E8-L53	Lightning
8-22-61	3:54 P	17 Minutes	E8-L58	Lightning
2-08-65	1:03 P	8 Minutes	E8-L53	High Winds
2-08-65	2:01 P	6 Minutes	E8-L58	High Winds
1-01-66	11:04 P	20 Minutes	E8-L58	High Winds
6-10-66	11:27 A	Momentary	E8-L53, E8-L-58	Lightning
8-17-67	8:41 A	6 Minutes	E8-L53, E8-L58	Overload

Letter, R. C. Ingersoll to R. B. Bixler, "Unscheduled Outages - Lines E8-L53 and E8-L58 to 221-B Building", 9-28-67.

The average time to restore the power after these outages is 10.6 minutes. Electrical distribution personnel can always be called from their homes to the job within 45 minutes, if switching loads or other work is required.

The "mean time between failures" for the period January, 1960, to the present is 6,789 hours.

$$\left( \frac{8760 \text{ hrs/year} \times 7.75 \text{ years}}{10} \right)$$

The probability that the system will be available to perform its function is estimated to be 99.997%.

$$\left( \frac{8760 \text{ hrs/year} \times 7.75 \text{ years} - 1 \frac{46}{60} \text{ hours}}{8760 \times 7.75} \right)$$

- B. The steam is generated at 284-E powerhouse in two Riley and three Erie City boilers, each rated at 80,000 lb/hr., and enters the distribution lines at 225 lbs. and 450°F. The entire system meets ASME codes and is maintained and inspected accordingly. In 1964, the distribution lines from the boilers were inspected and overhauled. The lines were installed on new, solid metal supports, close to the ground, and with gradients readjusted to eliminate condensate pockets. All steam traps and valves are replaced with new and reconditioned parts. Overhead clearances of lines at roadways are 20 feet or more (see Figure V-H-2).
- C. A water reservoir containing 3,000,000 gallons provides sufficient reserve for steam generation and cooling water demands during the time lag in reducing flows to minimum, emergency, standby conditions. After this, a minimum water supply of 3,000 gpm from the 100 Area export system is assured by 100 Area steam pumps, in the event of an electrical outage which also affected the 100 areas.

#### VIII. ENVIRONMENTAL INCIDENTS

##### A. Earthquake

The possibility of damaging earthquakes occurring in the Hanford region is remote. Eastern Washington is in Zone 2 of the U.S. Coast and Geodetic Surveys Seismic Probability Map, verging to Zone 1. This indicates that the site is in an area of moderate seismicity (moderate damage) verging to minor seismicity (minor damage).<sup>(13)</sup> While quakes of MM-VI intensity and higher can be postulated, given sufficient time, the limited history of seismic occurrences indicates to various experts that the maximum probable intensity should be no greater than MM-IV<sup>(41)</sup> of MM-5.5<sup>(13)</sup>. Moreover, for Puget-Sound-based quakes, MM-intensities at Hanford have been less than the intensities at other sites at comparable distances<sup>(42)</sup> because of the confirmed absorptive characteristics of Hanford's basin-fill sediments. This factor, combined with the lesser

acceleration of the long-period waves as shorter period waves filter out with distance indicates that the maximum possible damage may result from locally-originating quakes whose shorter-period waves are not dampened out and which could be amplified by resonance in passage through local sediments.<sup>(43)</sup>

Photogeologic interpretation of the Pasco Basin<sup>(44)</sup> indicated numerous possible faults, particularly in the Wallula Gap to Rattlesnake Hills area along a zone called "The Olympic-Wallowa Lineament",<sup>(45)</sup> Recently, detailed field studies<sup>(46)</sup> of the 50-mile segment of that "lineament" from Wallula Gap to the Rattlesnake No. 1 Well at the southwest corner of the Hanford Project showed that faults of significance are clearly absent. In addition, features forming that lineament are genetically unrelated to each other. The lineament thus is non-existent as a shallow-seated tectonic feature along which earthquakes can occur.

Earthquakes can cause damage by three processes: 1) rupture and offset of rock along a fault, 2) differential compaction of sediments, and 3) shaking of the ground and structures upon it. A fault offset in the basalts would be dissipated in the compact but unconsolidated sediments that underlie Hanford to hundreds of feet of depth. It is highly unlikely that a fault offset would ever be observed at grade. Gravels, predominating beneath separations plants, generally are dry and in surface - to - surface contact with interstitial fillings of sand and silt. Appreciable compaction is therefore not possible. The prime concern thus is ground motion or acceleration. This applies whether quakes originate locally or in the Puget Sound region. Equipment now being installed (accelerographs) will provide much of the needed information. Acceleration damage would be largely to above ground structures in which vibration periods permit resonance. Extensive damage to well-built structures at Hanford, whether above or below grade, is not probable, short of earthquakes of intensities believed to be highly unlikely.

#### B. Floods

The Hanford Area was inundated periodically to varying depths by glacial melt waters from about 20,000 to 10,000 years ago. Geologists believe that a sudden, major and virtually excludable ground upheaval would be required for such a lake to be reformed. Similarly, the possibility of landslides blocking the Columbia River sufficiently to cause an objectionable increase in ground water level below the Hanford Plateau is also virtually excludable. A breach in the Grand Coulee Dam is considered to be very unlikely. The crest of water resulting from such an unlikely event would not rise to the level of the plateau upon which the B Plant facilities are located.<sup>(47)</sup>

A flood caused by excessive rainfall over a short period of time is very unlikely. The ground could, for example, drain off ten inches per hour with erosion being the main effect. The B Plant swamp would be lost in such

an event.<sup>(47)</sup> The annual rainfall, however, is unlikely to exceed 13 inches in a hundred years.<sup>(40)</sup> For at least the last 10,000 years, the Hanford Area has experienced little rainfall - about 5 to 10 inches per year. Under these conditions, the soil has dried to the extent that added water, up to seven percent by volume, can be held above the water table by capillarity. In addition, soil fractions have a strong affinity for the radioisotopes of principal concern. Plutonium is sorbed in the top few feet of soil and is not leached significantly by water flow. Cesium-137 and strontium-90 are also sorbed from a flowing stream of water, and the isotopes migrate downward more slowly than they decay after the flow of water has stopped. Assuming the exclusion of surface waters other than rainfall, isotopes in the ground can be confidently predicted to remain above the water table. The sorption coefficients and competition from the other ions are such, however, that a large or continued percolation of water through the soil would cause migration of the isotopes to and with the ground water; the isotope migration rate for cesium and strontium would be less than one-hundredth that of the associated lineal rate of water flow.

REFERENCES

1. Evaluation of Hazards, Chemical Processing Department, HW-71918 REV., compiled by E.G. Pierrick and R.E. Tomlinson, November 2, 1964. (Secret)
2. H.L. Caudill, R.L. Junkins, M.L. Oldfather, O.V. Smiset and V.W. Smith, Chemical Processing Department Hazards Review Task Force Report, Waste Management Program - Phase I (Project CGC-897), HW-77379, April 29, 1963.
3. G.E. Backman, O.F. Beaulieu, R.L. Hibbard, Jr. and J.D. McIntosh - Hazards Evaluation Fission Product Packaging Facility (Project CGC-981) and Prototype Waste Management Packing Equipment (Project CAC-132) RL-SEP-859, November 30, 1965.
4. H.L. Caudill and L.L. Zahn - Project CGC-897 - Title I Design - Fission Product Storage in B-Plant, HW-69011, April 3, 1961. (Secret)
5. W.C. Schmidt - B-Plant Phase I Information Manual, HW-77016, March 15, 1963.
6. S.J. Beard and B.F. Judson - Process Specifications for Operational Control Purex and B-Plant Fission Product Recovery Processes, HW-78275, July 16, 1963. (Sec.)
7. H.L. Caudill, H.D. Haberman and J.W. Kolb - Process Design Engineering, Waste Management Program - Phase II, HW-77163, March 31, 1963.
8. H.L. Caudill, E. Doud, D.C. Fleischer, and W.H. Sevier - Design Criteria 221-B Waste Fractionization Phase III - Waste Management, HW-81802, Pt. 1 May 15, 1964, (Secret) and HW-81802, Pt. 2, May 28, 1964.
9. C.A. Lyneis - Project Proposal, Revision 4, B-Plant Modifications for FPCE-Waste Management Integrated Facilities (Project CAC-181), ISO-634, March 20, 1967.
10. L.W. Finch - Project Proposal, Revision 2, Waste Fractionization - B-Plant (Project CAC-144), ISO-787, April 18, 1967.
11. E.B. Jackson - Project Proposal Promethium Purification B-Plant (Project AAE-207) US AEC-RL, June 7, 1967.
12. R.F. Foster - Evaluation of Radiological Conditions in the Vicinity of Hanford for 1962, HW-76526, February 25, 1963.
13. G.W. Hausner - Recommended Seismic Design Criteria for Hanford Nuclear Facility, DUN-3130, October 1, 1967.
14. Waste Management Technical Manual, ISO-100, J.S. Buckingham, Editor, August 31, 1967. (Secret)

15. D. E. Larson - B Plant Phase III Flowsheets, ISO-986, August 31, 1967. (Secret)
16. Specifications and Standards, Strontium Semiworks, RL-SEP-20, Fission Products Process Engineering, February 15, 1965.
17. US AEC Manual Chapter 0524, May 12, 1964.
18. Specifications and Standards for the Operation of B Plant and Associated Facilities, ARH-144, Waste Management Process Engineering, December 8, 1967.
19. N.I. Sax - Dangerous Properties of Industrial Materials, Reinhold Publishing Corporation, New York, N.Y., 1963, 2nd Edition.
20. The Facilities Change Notice System for the Control of Design Reference Media, Operating Instruction, O.I. 5.8.1.3, May 24, 1967.
21. Handbook of Chemistry and Physics, R.C. West, ed., The Chemical Rubber Company, Cleveland, Ohio, 45th ed., 1964.
22. L.P. Bupp - Chemical Research and Development Monthly Report, HW-69062, March 12, 1961. (Secret)
23. Z.M. Shapiro and T.R. Moffette - Hydrogen Flammability Data and Application to FWR Loss of Coolant Accident, WAPD-SC-545, September 21, 1957.
24. W.J. DeCoursey et al. - "Effects of Water Vapor Content on the Inflammable Limits of Ammonia-Oxygen-Nitrogen Mixtures", Canadian Journal of Chemical Engineering, Vol. 40, p. 203 (1962).
25. W.L. Bulkley and H.W. Husa - "Combustion Properties of Ammonia," Chemical Engineering Progress, Vol. 58, p. 81 (1962).
26. I. Ginsbergh and W.L. Bulkley, "Hydrocarbon-Air Detonations...Industrial Aspects", Chemical Engineering Progress, Vol. 59, p. 82 (1963).
27. R.M. Wagner, Investigation of Explosive Characteristics of Purex Solvent Decomposition Products (Red Oil), HW-27492, March 17, 1953.
28. TNX Evaporator Incident, DP-25, January 12, 1953, May 15, 1953 (Declassified).
29. E.C. Martin, BNW - Private Communication to G.L. Ritter, October 16, 1967.
30. J.J. Shefcik - Safety Aspects of Purex Plant Concentrator Operation, HW-40556, December 23, 1955.
31. L.J. Nitteberg and R.K. Smith - NPR High Efficiency Filter Bed Evaluation, Humidity Test Report, HW-77628, May 14, 1963.
32. J.B. Fecht - Project CAC-144 - Design Criteria Revision No. 3 - Supplement, B Plant Cell Fire Detection System, ARH-19, September 15, 1967.
33. M.R. Weiler - Fission Product Ratios of the Purex Filter, HW-82896, June 12, 1964. (Secret)

34. R.J. Sloat - Progress Report - Study of Land Requirements, ISO-668, January 16, 1967.
35. K.C. Schneider, BNW - Private Communication to R.A. Kyle, ARHCO, October 16, 1967.
36. G.L. Hanson - Letter, "Redox Ruthenium Emission", to R.W. McCullugh, September 24, 1965.
37. H.L. Caudill - Project CAC-144 - Design Criteria Revision 3 - Waste Fractionization - B Plant, ISO-838, May 2, 1967.
38. L. Oathes - Drawing, "Fission Product Recovery Emergency Raw Water Supply for B Plant", SK-2-18573, December 20, 1960.
39. G. Janson, Jr. - Letter, "Temperatures in Leaks of Fission Product Solution from a Buried Transfer Line", to P.W. Smith, November 7, 1966.
40. J.S. Kane and R.A. Kennedy - Turbogenerator Failure, 284-W Building, HW-57799, October 15, 1958.
41. R.A. Earle - "U.S. Coast and Geodetic Survey, 1960", Unpublished Letter to J.M. Musser, AEC.
42. F. Neumann - Seismological Investigations at the Hanford Area, 1958-1959, HW-63832, November 10, 1959.
43. R.E. Brown and J.R. Raymond - Geophysical Seismic Evaluation Study at Hanford, BNWL-47, December, 1964.
44. F.O. Jones and R.J. Deacon - Geology and Tectonic History of the Hanford Area and Its Relation to the Geology and Tectonic History of the State of Washington and the Active Seismic Zones of Western Washington and Western Montana, DUN-1410, June 15, 1966.
45. J.W. Skehan - A Continental - Oceanic Crustal Boundary in the Pacific Northwest, AFCRL-65-904, December 20, 1965.
46. R.E. Brown - Study of Reported Faulting in the Pasco Basin, in preparation.
47. D.J. Brown - Private Communications to W.A. Blyckert, October 20, 1967.
48. D.E. Jenne - Frequency Analysis of Some Climatological Extremes at Hanford, HW-75445, April, 1963.

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