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**SAFETY ANALYSIS—200 AREA
SAVANNAH RIVER PLANT
F-CANYON OPERATIONS (U)**

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February 1986

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The F-Canyon facility is located in the 200 Separations Area and uses the Purex process to recover plutonium from reactor-irradiated uranium. The irradiated uranium is normally in the form of solid or hollow cylinders called "slugs." These slugs are encased in aluminum cladding and are sent to the F-Canyon (Building 221-F) from the Savannah River Plant (SRP) reactor areas or from the Receiving Basin for Offsite Fuels (RBOF). The flow of radioactive materials in the 200 Separations Area is shown in Figure 1-1.

This Safety Analysis Report (SAR) documents an analysis of the F-Canyon operations and is an update to a section of a previous SAR (1). The previous SAR documented an analysis of the entire 200 Separations Area operations. This SAR documents an analysis of the F-Canyon and is one of a series of documents for the Separations Area as specified in the Savannah River Implementation Plans (2). A substantial amount of the information supporting the conclusions of this SAR is found in the Systems Analysis (3,4). Some F-Canyon equipment has been updated during the time between the Systems Analysis and this SAR and a complete description of this equipment is included in this report. The primary purpose of the analysis was to demonstrate that the F-Canyon can be operated without undue risk to onsite or offsite populations and to the environment. In this report, risk is defined as the expected frequency of an accident, multiplied by the resulting radiological consequence in person-rem. The units of risk for radiological dose are person-rem/year. Maximum individual exposure values have also been calculated and reported.

1.1.1 Historical Perspective

Construction of the F-Canyon facilities was completed in the early 1950s, and irradiated fuel processing started in November 1954. Processing facilities were shut down from March 1957 to March 1959 for modification and for installation of larger equipment, primarily jumbo mixer-settlers and continuous evaporators, to increase throughput by a factor of about three. Modifications and improvements have been made subsequently to allow the canyon to perform a wide variety of missions, but the basic mission and the process has not changed.

In 1977, a Systems Analysis was completed for the 200-Area Chemical Separations Facilities (5), and as noted earlier, an SAR was completed in 1980 (1). Recently an updated Systems Analysis (3,4) has been completed and several major improvements have been made to the F-Canyon that relate to the safety of the facility, e.g.,

- The installation of the distributive control system in the second solvent extraction cycle,
- Improvements in the emergency power system, and
- Improvements in the sump collection system.

FIGURE 1-1. Flow of Radioactive Materials in the 200 Separations Area

This SAR is based upon the updated Systems Analysis (3,4) and includes the recent major improvements to the F-Canyon.

1.1.2 Approach

This SAR follows the content requirements given in the Savannah River Implementation Plans (2). Following this summary discussion of the facility and operations is the site evaluation including the unique features of the F-Canyon. The F-Canyon facility and process design are described in Section 3 and a description of F-Canyon operations is given in Section 4. The accident analysis in Section 5 is followed by a list of safety related structures and systems (Section 6) and a description of the Quality Assurance program (Section 7).

The accident analysis in this report focuses on estimating the risk from accidents as a result of operation of the F-Canyon facilities. The F-Canyon operations were evaluated on the basis of three considerations: 1) potential radiological hazards, 2) potential chemical toxicity hazards, and 3) potential conditions uniquely different from normal industrial practice. The effects of the hazard were quantified in terms of risk for personnel on the plant site and for those outside the plant boundary. Routine conditions were not considered in this analysis other than for comparative purposes.

The analysis presented in this SAR relies extensively on the recent comprehensive Systems Analysis for the F-Canyon operations (3,4). This Systems Analysis identified events deviating from normal operating conditions for the facility and evaluated the hazards resulting from such events. Most of the events leading to accidents for the F-Canyon operation were identified from data in the 200-Area Fault Tree Data Bank. Other initiating events were identified from analyses of other similar facilities.

Event progression subsequent to event initiation was largely based on data from the 200-Area Data Bank. This data bank contains a listing of the significant safety-related occurrences over the history of F-Canyon operations. The data bank contains more than 135,000 entries ranging from minor equipment malfunctions to incidents with significant potential for injury or contamination of personnel. In addition to the data bank, analyses of system characteristics, operating procedures, and engineering data were applied. Where necessary, event trees supported by fault trees were used to determine and quantify frequencies for a particular accident sequence from the initiating event to a final outcome.

For those events where there was release of radioactivity from the process equipment, the impact of confinement barriers was considered in estimating the release of radioactivity to occupied areas and to the environment. Estimates of radionuclide release to occupied areas and to the environment are largely derived in the SAR. These radionuclide release calculations required consideration of the specific nuclides present in the process, their physical form, the mechanism of release, and the performance of the engineered safety features designed to prevent their release. Most factors are specific to the accident being considered, and the supporting rationale are included with the accident analysis.

The quantified source terms were used to calculate radiological consequences for personnel on the plant site and for offsite populations. Once consequences were available, risk was calculated as the product of the expected frequency of the event and the radiological consequences of that event.

1.2 SUMMARY DISCUSSION

A summary of the F-Canyon operations important to safety is presented followed by a summary of F-Canyon safety analysis including tables of radiological consequences and risks for accidents and normal operation.

1.2.1 Facility Operations

The 221-F chemical separations plant has a large shielded "canyon" building for processing irradiated materials. The plant dissolves irradiated fuel and target materials and produces solutions containing the various products that have been decontaminated from fission products by solvent extraction and ion exchange processes. Further processing is performed in unshielded facilities where the products can be converted from solution to solid form for shipment offsite.

The two principal safety concerns in operation of the facility are:

1. Loss of process fluids from the vessels or systems in which they are contained such that radionuclides can be released to the canyon atmosphere,
2. A criticality event resulting from accumulation of fissionable material in a geometry and under conditions conducive to a self supporting chain reaction.

The F-Canyon structure and supporting engineered safety features constitute an effective radiological confinement system designed to control and limit radionuclide release to the environment. For this area of criticality, canyon equipment by and large is not designed to geometrically preclude criticality. Hence, administrative and procedural control are the principle mechanisms for controlling criticality. Such an approach has historically been effective (no criticalities have been experienced during operations of the F-Canyon or any other facility of the Savannah River Plant).

The analysis presented herein indicates how expected frequencies for criticality events can be achieved if administrative and procedural controls are effectively applied.

1.2.2 Facility Risk

The estimates of radiological consequences and radiological risks from accidents as a result of operation of the F-Canyon are summarized in Table 1-1. The total calculated risk to the offsite maximum individual is only 3.9×10^{-5} rem/yr, a small fraction of the annual exposure from natural background of 0.09-0.1 rem. Population dose values are also small, calculated values

TABLE 1-1. Summary of Risks for F-Canyon Accidents for All Unit Operations

Accident		Risk		
		Offsite Population, person-rem/ yr	Onsite Population, person-rem/ yr	Offsite Maximum Individual, rem/yr
Natural Phenomena		2.4E-05	7.4E-06	1.0E-08
Externally Induced Failures		7.6E-03	8.0E-04	1.3E-06
Process Related Occurrences				
Medium Energetic (Airborne)	Fire	6.9E-04	7.6E-05	1.1E-07
	Uncontrolled Reaction	3.3E-02	3.3E-03	6.5E-06
	Criticality	5.4E-03	4.9E-03	8.7E-06
	Total (Airborne)	3.9E-02	8.2E-03	1.5E-05
Low Energetic (Airborne)	Transfer Error to 211-F	1.4E-03	4.3E-04	6.2E-07
	Transfer Error to Sump	1.9E-03	2.0E-04	3.4E-07
	Overflow to Sump	1.0E-02	1.0E-03	2.1E-06
	Leak to Sump	7.8E-02	7.8E-03	1.5E-05
	Processing Short-Cooled Fuels	2.7E-03	2.7E-04	5.1E-07
	Ruthenium Volatilization	8.5E-03	7.5E-04	1.9E-06
	Total (Airborne)	1.0E-01	1.1E-02	2.1E-05
Coil Failure (Liquid)		2.1E-02		1.8E-06
Total (Airborne)		1.4E-01	1.9E-02	3.6E-05
Total (Process Occurrences)		1.6E-01	1.9E-02	3.8E-05
Total for all accidents	Airborne	1.5E-01	2.0E-02	3.7E-05
	Liquid	2.1E-02		1.8E-06
	Total	1.7E-01	2.0E-02	3.9E-05

being 1.7×10^{-1} person-rem/yr for the offsite population group to a radius of 50 miles and 2.0×10^{-2} person-rem/yr for the onsite population group.

Radiological consequences for accidents are compared with those from normal operational exposure in Table 1-2. Doses from effluent releases for normal operation in Table 1-2 are for all 200-Area operations. Calculated doses in this SAR are typical for separations processes and are well within federal guidelines for accidental releases of 25 rem as stated in 10 CFR 100.

For normal operation, the calculated offsite dose to the surrounding population from gaseous effluents is 75 person-rem/yr from the 200-Area operations. For liquid effluent releases direct to surface streams as a result of the canyon operations, the calculated exposure is 1.1 person-rem/yr. The exposure to workers in F-Canyon operations during 1984 was 116 person-rem.

Radiological risks are dominated by releases as a result of normal operation. However, it is important to recognize that some of the process events considered in the accident analysis occur more frequently than once per year; hence the accident risks and normal operating risks presented in Table 1-2 overlap, particularly for liquid pathways doses. The accident doses and risks from accidental releases, however, make up only a small part of the doses tabulated for normal operations.

The very low risk from accidents for SRP F-Canyon operations are in part a result of the massive, relatively leak tight and blast resistant structure within which the processes are located. The structure itself is resistant to damage by extreme natural phenomena and by other external events and hence protects the process equipment from damage. Further, radioactivity released within the facility is largely contained by the facility; gaseous effluents from the system must pass through a massive sand filter located below ground level. Hence the filter system itself has high availability.

The very low values for radiological consequences and for risk as a result of F-Canyon operations indicate that the facility can be operated without undue risk to the public or operating personnel.

TABLE 1-2. Comparison of Risk for Accident and Normal Operation

Pathway	Risks Due to Accidents				Risks Due to Normal Operation, person-rem/yr*
	Natural Phenomena	Externally Induced Failures	Process Operations	Total	Annual Dose, 1984
Airborne Release to Offsite Population, person-rem/yr	2.4×10^{-5}	7.6×10^{-3}	1.4×10^{-1}	1.5×10^{-1}	75
Airborne Release to Onsite Population, person-rem/yr	7.4×10^{-6}	8.0×10^{-4}	1.9×10^{-2}	2.0×10^{-2}	--
Airborne Release to Offsite Maximum Individual, rem/yr	1.0×10^{-8}	1.3×10^{-6}	3.6×10^{-5}	3.7×10^{-5}	--
Liquid Release to Offsite Population, person-rem/yr**	--	--	2.1×10^{-2}	2.1×10^{-2}	1.1
Liquid Release to Offsite Maximum Individual, rem/yr**	--	--	1.8×10^{-6}	1.8×10^{-6}	--

*Total for all of the 200-Area Operations due to effluent releases.

**Direct release to Four Mile Creek.

1.3 REFERENCES

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3. Durant, W. S., Perkins, W. C., and Severynse, T. F. Systems Analysis - 200 Area Savannah River Plant F-Canyon Operations. Internal Report DPSTSY-200-1F, Volume 1, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, SC, December 1983.
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2.0 SITE EVALUATION

2.1 SITE DESCRIPTION

The Savannah River Plant (SRP) is located in South Carolina on an approximately circular site of about 300 square miles, bounded on the southwest by the Savannah River, and centered approximately 25 miles southeast of Augusta, Georgia. Other distances to locations of interest are shown in Table 2-1. The plant reservation occupies parts of Aiken, Barnwell, and Allendale Counties.

Two chemical separation areas are located near the center of the plant site between Upper Three Runs Creek to the north and Four Mile Creek to the south, as shown in Figure 2-1. These areas are designated as 200-F and 200-H. The principal unit in each area is identified as a 221 building, which contains the remotely serviced and operated canyon facilities and the directly serviced and operated finishing facilities.

The F-Canyon is located in F-Area and noted as Building 221-F, as shown in Figure 2-2.

A detailed description of the seismologic, geologic, hydrologic, meteorologic, and population characteristics of the Savannah River Plant site is presented in Reference 1. These characteristics are examined in Section 5 for situations in which operation of the facility could be affected by natural phenomena. The following material summarizes the information from Reference 1. Detailed population, seismologic, etc., data pertain to the SRP site, not to the F-Canyon specifically.

2.1.1 Population Characteristics

The potentially affected population for any F-Canyon accidents would include personnel on the SRP reservation and the general public in the areas around the SRP. Table 2-2 shows the number of onsite personnel by assignment. Currently, there are over 15,700 people working at the site.

The offsite population for the 13 counties surrounding the reservation are shown in Table 2-3. The two largest population centers located within reasonable proximity of the plant are Augusta (25 miles west-northwest) and Aiken (20 miles north). Also shown in Table 2-3 is the 1980 population for major population centers within about 25 miles of the reservation boundaries. Figure 2-3 shows the major population centers within a 150-mile radius of the SRP reservation.

2.1.2 Seismology

The SRP site is in an area that has a rather low seismic frequency. Based on three centuries of recorded history of earthquakes, an earthquake above Intensity VII on the Modified Mercalli Scale (MM) would not be expected at the Savannah River Plant. (See Reference 1, Appendix C for a description of the Modified Mercalli Scale.) Only two earthquakes of Intensity VII or greater have occurred within 200 miles of the site. They were the Charleston, South

TABLE 2-1. Approximate Distances to Locations of Interest from SRP

Location of Interest from Center of Plant	Distance (miles)
Greenville, SC	115
Atlantic Ocean	100
Charleston, SC	100
Savannah, GA	100
Columbia, SC	60
Augusta, GA	25
Aiken, SC	20
Barrow, SC	15
Williston, SC	15
Jackson, SC	12

FIGURE 2-1. The Savannah River Plant Site

FIGURE 2-2. Plot Plan, 200 F-Area

TABLE 2-2. Onsite Personnel at SRP, November 1985

Assignment	Number of Personnel
Savannah River Plant	6,787
Savannah River Laboratory	1,015
Construction	6,154
DOE/Savannah River	292
Forest Service	25
Cafeteria	68
Janitorial	405
Savannah River Ecology Laboratory	118
Laundry	25
Wackenhut	<u>848</u>
Total	15,737

TABLE 2-3. 1980 Population Counts

<u>Counties</u>			
<u>South Carolina</u>		<u>Georgia</u>	
<u>County</u>	<u>1980 Population</u>	<u>County</u>	<u>1980 Population</u>
Aiken	105,625	Burke	19,349
Allendale	10,700	Columbia	40,118
Bamberg	18,118	Richmond	181,629
Barnwell	19,868	Screven	14,043
Edgefield	17,528		
Hampton	18,159		
Lexington	140,353		
Orangeburg	82,276		
Saluda	16,150		

Major Population Centers Within About 25 Miles

<u>City</u>	<u>Distance (miles)</u>	<u>Direction from Plant</u>	<u>1980 Population</u>
Augusta, GA	25	West-Northwest	47,532
N. Augusta, SC	25	Northwest	13,593
Aiken, SC	20	North	14,978
Williston, SC	15	Northeast	3,173
Barnwell, SC	15	East	5,572
Allendale, SC	26	Southeast	4,400
Waynesboro, GA	28	Southwest	5,760

FIGURE 2-3. Location of SRP Relative to Surrounding Population Centers

Carolina, event (epicenter 90 miles from the SRP site) and the Union County, South Carolina, event (epicenter 100 miles from the SRP site).

Available data indicate 18 events of Intensity VII or greater in South Carolina. The largest event was the 1886 Charleston earthquake, which was Intensity X. The next largest events were the Giles County, Virginia earthquake (Intensity VIII) of 1897 and the Union County, South Carolina, earthquake (Intensity VII-VIII) of 1913. All other events were Intensity VII or less.

If data on the earthquakes in the vicinity of Charleston were omitted, South Carolina and Georgia would be considered to have relatively few earthquakes. The Charleston zone contains the epicenter of the 1886 Charleston earthquake, the 1912 Charleston earthquake, and hundreds of aftershocks of the 1886 event recorded until at least 1900.

The great Charleston earthquake of August 31, 1886, has dominated consideration of seismic activity in South Carolina for many years. This disturbance had dual epicenters, one at Woodstock (16 miles N 30°W from Charleston) and another 13 miles due west from Charleston. The epicenters were 14 miles apart. The Charleston quake, which was Intensity X, was felt 800 to 1000 miles away and affected an area of about 2,000,000 square miles. The Charleston earthquake caused only minor, superficial surface changes. The epicentral region was broken by many fissures through which water issued, but the fissures seldom attained a width of more than one inch. In the vicinity of Augusta, GA, just over 100 miles from the epicentral area, ground motion was estimated at Intensity VIII during the earthquake.

The Middleton Place-Summerville zone (25 miles northwest of Charleston, SC) has experienced seismicity continually since the 1886 Charleston earthquake. This zone and the Jedburg (62 miles northwest of Summerville) and Adams Run (25 miles south-southwest of Summerville) zones have had 36 shocks since 1974.

The Bowman Area (16 miles southeast of Orangeburg, SC) has experienced 12 shocks since 1974 in an apparent northeast trending zone. The poorly determined foci are from 0 to 6.2 miles in depth, and other parameters cannot be accurately ascertained.

In western South Carolina, there have been 56 shocks since 1978, most of which are centered in the Piedmont Plateau. The most prominent activity was related to reservoir-induced seismicity (Lake Monticello and Clarks Hill Lake).

Maximum earthquake intensities have been considered for three seismic source regions: 1) the Appalachian Mountains, 2) the Atlantic Coastal Plain, and 3) the Charleston seismic zone. The maximum historical earthquake in the Appalachian Mountains was the Giles County, Virginia, event that occurred in 1897 with Intensity VIII. The maximum historical earthquakes in the Atlantic Coastal Plain (excluding the Charleston zone) were of Intensity VII. The Charleston seismic zone contains the 1886 event of Intensity X.

The Design Basis Earthquake for the Savannah River Plant has been conservatively established as an earthquake with a Modified Mercalli Intensity of VIII (see Table 5-1) and a corresponding zero-period peak horizontal ground acceleration of 0.20 g.

Several fault systems occur in and adjacent to the piedmont and the valley and ridge tectonic provinces of the Appalachian System. The closest fault is the Belair, which is about 25 miles from the SRP site. Evidence for the last movement of this fault is not conclusive, and it is not considered capable. There are no known active or capable faults within 200 miles of the Savannah River Plant.

2.1.3 Geology

South Carolina is divided into two main geologic provinces: 1) The Piedmont Plateau, which is underlain by igneous and metamorphic rock; and 2) the Atlantic Coastal Plain, which is characterized by flat, mostly unconsolidated sediments of Cretaceous age or younger. The boundary between the two provinces is called the Fall Line. It is not a sharp line of contact but a zone of transition from the typical land forms of one province to those of the other. The SRP plant site is located on the upper Atlantic Coastal Plain in Aiken and Barnwell Counties, South Carolina. About 20 miles northwest of the plant site is the lower edge of the Piedmont Plateau, the other main geologic province in South Carolina.

The geologic layers of the plant site affect the migration rates and directions of groundwater flow. Geologic formations beneath the site are the Hawthorn, Barnwell, McBean, Congaree, Ellenton, and Tuscaloosa Formations, and bedrock (crystalline metamorphic rock and the Dunbarton Triassic Basin). Figure 2-4 is a profile of the geologic formations beneath the SRP. The sediments that constitute the formations above bedrock are either unconsolidated or semiconsolidated. The crystalline metamorphic rocks outcrop at the Fall Line and dip approximately 36 ft/mi to the southeast underneath the coastal plain sediments.

The major physiographic divisions in the site region are the Aiken Plateau and the Congaree Sand Hills. The Aiken Plateau is bounded by the Savannah and Congaree Rivers and extends from the Fall Line to the coastal terraces. The surface of the Aiken Plateau is highly dissected and characterized by broad interfluvial areas with narrow steep-sided valleys. Relief is locally as much as 300 ft.

The site region, defined as the area within 200 miles of the site, contains elliptical depressions called Carolina bays. These features, common throughout the Atlantic Coastal Plain, are most numerous in North Carolina and South Carolina.

The Congaree Sand Hills trend along the Fall Line northwest and north of the Aiken Plateau. The sand hills are characterized by gentle slopes and rounded summits and are interrupted by the valleys of southeast-flowing streams and their tributaries.

FIGURE 2-4. Profile of Geologic Formations Beneath the SRP

2.1.4 Hydrology

2.1.4.1 Surface Water

Most of the surface water at the Savannah River Plant (SRP) results from rainfall or from the water that is pumped from the Savannah River and is used for secondary cooling for the plant reactors. The usage rate of the Savannah River water by the SRP varies from about 300 cubic feet per second (cfs) to about 1000 cfs, depending on the number of reactors operating and the corresponding reactor power levels. After use, the heated cooling water is returned to the river via one of the plant streams.

Almost all of the SRP site is drained by tributaries of the Savannah River. Each tributary is fed by several small streams so that no location on the site is very far from a flowing stream. Only one small stream in the northeastern sector of the site drains to the Salkehatchie River instead of the Savannah River.

In addition to these streams, surface water is held in more than 50 artificial impoundments totaling over 3,000 acres. Par Pond is the largest, with an area of about 2,700 acres. Water is retained intermittently in wetlands and in more than 200 natural basins, including some Carolina bays. A large swamp borders the Savannah River and is crossed by several of the streams.

2.1.4.2 Groundwater

Groundwater is defined as that part of the water beneath the land surface that is free to move by gravity. It occurs in the zone of saturation, in which all the interconnected openings or pores in the rocks of the earth's crust are filled with water under hydrostatic pressure. The number, size, and shape of the openings in porous rocks and sediments and the degree of interconnection determine the amount of water that can be stored and yielded.

The physiography of the area determines the location of groundwater gradients. Groundwater recharge is fairly uniform over the entire region. Groundwater gradients are determined by the location and depth of incision of the stream valleys.

Groundwater gradients in deeper aquifers are controlled by recharge and discharge areas that are farther removed from the point of interest. Shallower aquifers are controlled by nearby recharge and discharge areas.

Three distinct geologic and hydrologic systems exist beneath the SRP site:

- The coastal plain sediments, of Cretaceous and Tertiary age (Tuscaloosa and above), where water occurs in porous, unconsolidated to semi-consolidated sands and clays.
- The buried crystalline metamorphic basement rock consisting of chlorite-hornblende schist, hornblende gneiss, and lesser amounts of quartzite, where water occurs in small fractures.

- A buried Triassic basin, consisting mostly of red consolidated mudstone with some poorly sorted sandstones, where water occurs in the intergranular space but is very restricted in movement by the extremely low permeability.

The coastal plain sediments contain several prolific aquifers. The hydrology of buried crystalline metamorphic rocks and the Triassic mudstone beneath the SRP site have been studied intensively as a result of an exploration program from 1961 to 1972 to assess the safety and feasibility of storing radioactive waste in these rocks.

2.1.5 Meteorology

Climate near the SRP is relatively temperate with mild winters and long summers. This area, while subject to continental influences, is protected by the Blue Ridge Mountains to the north and northwest from the more vigorous winters prevailing in the Tennessee Valley. The terrain offers little moderating effect on the summer heat. The plant site and surrounding areas are characterized by gently rolling hills with no unusual topographic features (except the Savannah River along the western boundary) to significantly influence the general climatology.

Many hurricanes that affect the South Carolina coast originate in the West Indies and Caribbean area, then turn northward along the west coast and Panhandle of Florida, and then turn northeast, following a track either across land to the south or across water parallel to the Atlantic coast. These storms lose much of their force before they reach South Carolina. Storms that enter the Atlantic Ocean after crossing land occasionally regain their strength and lash the coast of South Carolina with full Hurricane force. Most hurricanes that originate far out in the Atlantic or in the East Indies curve away from the coast and stay over the ocean. Only a few of the hurricanes strike the mainland. Only 38 hurricanes caused damage to South Carolina during the 272 years of record between 1700 and 1971. This is an average frequency of one every seven years. The hurricanes that affect South Carolina occur predominately in the months of August (37% of total) and September (47% of total).

The SRP is located 100 miles from the Atlantic Coast. The occurrence of a hurricane along the coastal region does not necessarily mean that the Savannah River Plant will be subjected to hurricane force winds. The high winds usually associated with hurricanes tend to decrease as the distance from the eye of the hurricane increases and as the storms move over the land. Winds of 75 mph were measured by anemometers (mounted at 200 ft) only once during the history of the SRP, when Hurricane Gracie passed to the north of the plant site on September 29, 1959.

In South Carolina, the greatest percentage of tornadoes occur in April and May, with a smaller maximum, about 20%, in August and September. The latter are mainly the result of spawning by hurricanes and waterspouts. One or two tornadoes can be expected in South Carolina during April and May, with one expected in each of the months of March, June, July, August, and September.

Weather Bureau records show 278 tornadoes in Georgia over the period 1916-1958 and 258 tornadoes in South Carolina for the period 1950-1980. The general direction of travel of confirmed tornado tracks in Georgia and South Carolina is from the southwest to northeast.

The Savannah River Plant is in an area where occasional tornadoes are to be expected. Statistics for the period 1950-1978 for a rectangular region of Georgia and South Carolina including the SRP site show a total of 248 tornadoes, or 8.5 per year. Although the number of tornadoes in this region averages 8.5 per year, there have been only four occasions (May 28, 1976; July 2, 1976; April 23, 1983; and August 26, 1985) on which tornadoes were confirmed on or in close proximity of the plant site. On all four occasions only light damage was reported, i.e., displacement of light sheet metal roofing, window breakage, trees snapped and uprooted, etc. On no occasion has there been tornado damage to any production facility on the plant site. There have also been several sightings of funnel clouds which did not touch ground and caused no damage on the SRP site. Investigation of the confirmed tornadoes indicated wind speeds of 100 to 175 mph.

The Design Basis Tornado is defined as a tornado having a rotational speed of 230 mph at a radius of 230 ft and a maximum translational speed of 50 mph, and a total pressure drop of 1.5 psig at a maximum rate of 0.5 psi/sec (see Sections 3.2.2 and 5.1.1.1).

2.2 FACILITY DESCRIPTION

In the F-Canyon building, the Purex solvent extraction process is used to recover ^{239}Pu , ^{237}Np , and ^{238}U from irradiated depleted or natural uranium. Irradiated targets and slugs are received from the reactors and dissolved. After passing through the Purex process, aqueous streams of plutonium and uranium are transferred to the FB line and the A-Line, respectively.

The unit operations in the F-Canyon are shown in Figure 2-5 and include receipt of reactor targets, dissolving, head end, solvent extraction, evaporation, ion exchange, and waste disposal. A more complete discussion of F-Canyon process operations is included in Section 3.0 and the F-Canyon Systems Analysis (2).

2.2.1 The F-Canyon Building 221-F

The 221-F building is a reinforced concrete, blast-resistant structure that is 835 ft long, 122 ft wide, and 66 ft high on four levels, and built on a reinforced concrete foundation slab. The building consists of seventeen 43-ft sections and one 85-ft section at the south end. Two parallel canyons, each 15 ft wide at the bottom, 30 ft wide at the top, extend from Sections 5 through 18, and constitute the process areas. These two canyons contain the high activity and low activity materials and are designated the hot canyon and the warm canyon, respectively.

FIGURE 2-5. F-Canyon Flowsheet

2.2.2 Dissolving - Head End

Processing begins with dissolution of the irradiated targets or slugs. With the normal aluminum-clad, metallic uranium fuel, the aluminum cladding is dissolved in a sodium hydroxide-sodium nitrate solution, with evolution of ammonia and hydrogen. After the decladding solution with contained aluminum is removed, the uranium is dissolved in nitric acid. Noble gas fission products and some iodine are evolved during dissolving. These gases are routed through a vessel containing silver nitrate-coated sorbent materials (to reduce iodine emissions), a filter, and an acid recovery unit that converts some of the nitrogen oxides in the off-gas to reusable nitric acid, reducing nitrogen oxide emissions. The uranium solution is clarified, and some of the fission products are removed.

2.2.3 Solvent Extraction

In the first solvent extraction cycle, uranium and plutonium are extracted away from the remaining fission products with tributyl phosphate in a hydrocarbon diluent and then separated from each other. The uranium and plutonium are further purified with separate second solvent extraction cycles. The final uranium product solution is concentrated and sent outside the canyon building for subsequent processing in A-Line (Building 221-1F). The plutonium product solution is transferred to a separate area within the canyon building where the plutonium is concentrated by ion exchange, precipitated as the fluoride and, with some further treatments, reduced to the metal.

2.2.4 Wastes - Recovery

In the canyon building, neptunium and residual plutonium are recovered by ion exchange from the fission product waste solution from solvent extraction. Neptunium is further purified by ion exchange, and the final product solution is sent to the H-Area neptunium line. The plutonium is sent to the plutonium processing system. Also, irradiated plutonium fuel and target assemblies may be processed in the main solvent extraction system and in the plutonium finishing system.

2.3 REFERENCES

1. Dukes, E. K. The Savannah River Plant Environment. DP-1642, E. I. du Pont de Nemours and Company, Savannah River Laboratory, Aiken, SC, June 1984.
2. Durant, W. S., Perkins, W. C., and Severynse, T. F. Systems Analysis - 200 Area Savannah River Plant F-Canyon Operations. Internal Report DPSTSY-200-1F, Vol. 1 and 2, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, SC, December 1983.

3.0 FACILITY AND PROCESS DESIGN

3.1 DESIGN CRITERIA

The 221-F Canyon and processes were designed in accordance with applicable Du Pont Engineering Standards in effect at the time. The operations have not been modified significantly since the facilities were originally constructed.

Du Pont Engineering Standards were used for the design and construction of all Savannah River Plant (SRP) facilities, ensuring the use of standard, reliable, and economical materials with quality workmanship. The Du Pont Engineering Standards provide specific, detailed, instructive information for the designer. They refer to portions of national standards where applicable.

Du Pont Engineering Standards are updated frequently and are reviewed and revised, or reaffirmed every five years. The standards provide design information and specifications in the following general areas:

Architectural	Lubrication
Civil	Machine Design
Concrete	Piping
Drafting	Plumbing
Electrical	Power
Environmental Protection	Process Equipment
Fire Protection	Safety
Heating and Ventilation	Steel
Instruments	

Du Pont and SRP Engineering and Design Standards were compared with the standard requirements set forth in the Department of Energy (DOE) Manual (1). From this comparison, the following conclusions were made:

- Du Pont Standards met or exceeded DOE requirements in areas not related to nuclear safety.
- There was almost literal correspondence between Du Pont and DOE requirements for nuclear safety and radiation protection.

The design of any upgrade of systems, structures and components is done according to current DOE standards as set forth in DOE Orders 5480.1A and 6430.1.

3.2 PROCESS AND FACILITY DESCRIPTION

3.2.1 Summary Description

The F Canyon Building is a radiochemical plant for the separation and recovery of ^{239}Pu , ^{237}Np , and ^{238}U from irradiated natural or depleted uranium. The processing equipment is isolated from the operating personnel, the environment, and the public in two parallel canyons, 15 ft wide at the bottom and 30 ft wide at the top, separated from each other by a central operating and service section that is divided into four levels (Figure 3-1):

DELETED VERSION

DPSTSA-200-10, SUP-4

FIGURE 3-1. Building 221-F Isometric Sections

DELETED VERSION

Fourth Level - Control room and general office space.

Third Level - Feed tank gallery, sample aisles.

Second Level - Pipe gallery, mask and tool decontamination room, canyon air supply fan room.

First Level - Change rooms, services, and maintenance facilities.

The more highly radioactive processing operations such as dissolution of irradiated materials, precipitation and centrifugation (head end), bulk fission product separation and waste evaporation, are performed in the hot canyon and the final purification of the products is performed in the warm canyon.

The hot and warm canyons are housed in a blast-resistant reinforced concrete structure 835 ft long, 132 ft wide, and 66 ft high. The above grade height is 52 ft. Rail access to the hot canyon for the delivery of irradiated material and hot canyon process equipment is through a railroad tunnel that is entered beyond the south end of the main building. The rail spur terminates 128 ft inside the building. Equipment for the warm canyon enters through a truck well on the side of the building. Six personnel entrance towers connect to the center section through tunnels under the first level. There are also 15 entrances to the building to such areas as the gang valve corridor and the truck well area. A more complete description is contained in the F-Canyon Systems Analysis (2,3).

3.2.2 Process Design Consideration

The 221-F Building isolates the radiological hazard in the recovery process for U, Pu, and Np from the operators and the public. These isotopes and associated fission products are confined within the F-Canyon processing cells by the radiological shielding walls and by engineered features that minimize releases through ventilation air service, instrument lines, cooling water, and steam condensate discharge lines. These features are designed to remain intact in the event of any foreseeable accident or natural phenomena such as earthquakes or tornadoes. The building is a Class I explosion-proof structure. The building (Figure 3-2) is a Maximum Resistance Construction (MRC) structure designed to withstand pressures of 1000 lb/ft². The building is constructed of reinforced concrete walls. The outside walls of the radioactive processing cells are 48 in thick for the most highly radioactive process, and 24 in thick for the process involving the lesser radioactive radionuclides. The roofs over the two process areas are 42 in thick and 30 in thick, respectively.

The F-Canyon Building is assumed to be insignificantly affected by an earthquake of less than VIII intensity on the Modified Mercalli Intensity Scale* of 1931. This assumption has been independently supported by studies

*A tabulation of the observed earthquake effects at various intensities on the Mercalli scale is found in the F-Canyon Systems Analysis (3) and in Table 5-1.

FIGURE 3-2. Building 221-F

by the Engineering Decision Analysis Co., Inc. of Palo Alto, CA (4,5). A study by URS/Blume (6), determined that the maximum seismic hazard at the SRP site is from an earthquake of intensity VII in the immediate vicinity of the site and from a postulated intensity X near Bowman, at a distance of 60 miles.

The canyon building has not been thoroughly analyzed with regard to tornado resistance (7). Uncontained equipment such as ventilation ducts, electrical services, and exposed pipe lines are vulnerable to missiles. The resistance of the truckwell rollup door curtain and railroad airlock door is questionable. The loss of the railroad airlock doors would not be significant unless the inner shielding doors were open.

The F-Canyon processing operations are divided according to the radiation level of the material being processed with the higher activity processing in the so-called hot canyon and the lower activity operations in the warm canyon. Irradiated uranium in the shape of solid or hollow cylinders is brought into the plant through an airlock. The railcar of irradiated material is prepared for unloading in the airlock, then pushed into the 221-F building for unloading into the canyon dissolvers. The canyons are dimensioned for a single row of process vessels. The canyons are divided into sections 43 ft long, with process operations occurring in 13 sections numbered 5 to 18 from south to north. The floors of each section are sloped $3/8$ in per ft to drain spills, leaks, and overflows to a 2 ft x 2 ft x 2-1/2 ft deep sump. Most services are supplied through the inner shielding wall from the central operating area. Some of the steam supply lines for heating process solution and for the motive force for the liquid transfer eductor are supplied from the Gang Valve Corridor on the outside of the hot and warm canyons.

Each canyon has a pipe rack above the Gang Valve Corridor that supports an array of headers interconnecting the various sections in the canyon for the transfer of radioactive solution between vessels in various parts of the process.

The facility design requires the use of the crane to make connections between the process equipment in the canyons with services or with radioactive transfer lines which are supplied from special nozzles on the canyon wall. There are two basic types of nozzles - electrical and piping - which mate to corresponding pieces of piping or conduit, called jumpers, that span the spaces between the service supplied to the wall and the vessel. Jumpers are individual lengths of pipe with lifting bails and special pipe connectors or electrical connectors welded on either end for remote coupling. Typical jumpers, when lifted by the bail, balance in a position that coincides with the dimensional coordinates of the connecting nozzle. Pipe jumpers can be very simple, consisting of a piece of pipe and the connectors, or quite complex. Jumpers can be fabricated with valve and flow measurement devices, with thermocouples or resistant temperature devices, and probes for liquid level or liquid density measurements.

The Hanford connector shown in Figure 3-3 is a rugged pipe connector that has demonstrated many years of reliability in nuclear service. The connector consists of:

- The tapered jointed male nozzle, which is part of the fixed embedded piping in the wall, or stubbed-off on the vessel.

NOTE: Connectors are lubricated with
Lubricant No. 1 (Contains 1-75 oil/solvent solution).
For connector gasket material, see B-136673.

FIGURE 3-3. Hanford Connector

- The female connector block, which is part of the movable piping.
- The guide skirt and gasket subassembly.
- The three-hook screw-puller type clamp mechanism with a 2-in drive nut.

The female connector block, guide skirt, and clamp are bolted together and operate as a unit. In operation, a pipe jumper, which is hanging from its balance point on a crane hook, is placed in the fixed male nozzle with the clamp hooks open. The loose jumper is held in position by its own weight, by the guide skirt, and by the stabilizing action of the open clamp hooks. An impact wrench carried by the crane, is positioned by the crane with the socket on the clamp drive nut, and operated remotely to close the clamp and seal the connector pieces. Positioning is normally accomplished remotely by the crane operator viewing through a periscope type optic on the hot canyon crane or through the shielding window on the warm canyon crane. Connectors are designed to be operated by impact wrenches rated for approximately 500 ft-lb of torque.

Nozzle ends of embedded pipe are horizontal and are rigidly held to close dimensional tolerance. Process vessels are positioned by equipment positioning guides, one set for each piece of equipment for a total of four sets for each section of the canyon. An equipment piece is precisely positioned by matching the positioning trunions on the bottom of the equipment piece with the trunion guides mounted in the floor of the canyon. The locations of all nozzles, cold walls, vessels, and racks are known with close tolerance. This knowledge is necessary for the design of replacement jumpers, new jumpers, or modified equipment. In addition, a jumper on one piece of equipment will fit on a similar piece of equipment.

There are four basic groupings of pipe wall nozzles in each section: 1) the cold side-wall nozzles for cold feed, cooling water, electricity, lubrication, steam, and instrumentation; 2) the horizontal nozzles on the hot side access the cooling water outlet, the waste and vent header; 3) low vertical connectors on the hot side that access steam supply from the jet gang valve corridor for supplying steam to the transfer jet; and 4) horizontal nozzles on the hot side that provide access to the pipe rack through the high verticals for transfers between vessels in various section.

Decontamination cells are provided in canyon sections 3 and 4. Equipment removed from the canyons can be decontaminated to some degree before it is removed to the hot or warm canyon maintenance shop for repair by maintenance personnel.

3.2.2.1 Receipt and Charging of Irradiated Material

Irradiated uranium in the form of hollow cylinders called slugs, and MK-42 targets packaged in bundles are shipped by railcar to F-Canyon, where the plutonium material is charged to the dissolver.

Irradiated target slugs are self-heating and, in the absence of a mechanism for heat removal, the fuel cladding could fail releasing volatile fission products. This condition would be approached if a large quantity, such as a cask-full or dissolver charge of target slugs, was not covered by water, or a canister was spilled in a confined space. The cladding would prevent combustion for several hours, but higher temperatures would be increasingly unsafe. Spontaneous combustion of irradiated slugs is prevented by shipping the fuels to 221-F Building in water-filled cask wells. Target slugs charged to a dissolver are lifted directly from these water-filled casks or from the water cooled storage locations in the canyon and dumped into the dissolver, which has previously been filled with decladding solution or, in the case of special material such as the MK-42 assemblies, charged directly into nitric acid-KF solution. Administrative controls require that the slugs be kept covered at all times while in the dissolver, except for the short time that a dissolver is drained and refilled.

Administrative controls are employed to ensure that the irradiated material received has been out of the reactor for a sufficient period of time to prevent excessive release of volatile fission products and to limit radiation exposure to operating personnel (8). Irradiated material is aged or cooled sufficiently after discharge from the reactor to allow radioactive ^{131}I , with a half-life of 8.07 days to decay. A minimum of 140 days is required for Mark 31A and 31B depleted uranium targets (8). The timing of fuel shipments to F-Canyon is regulated at the reactor. Radiation monitors set to alarm upon the detection of abnormal radiation levels indicating the presence of fuel insufficiently aged, are part of the preparation of the irradiated fuel for shipment.

Excessive charges of irradiated fuel or targets from the perspective of criticality prevention is not a consequence because of the material, engineered safety features, and administrative controls. The two annular dissolvers installed in F-Canyon are critically safe by geometry for ^{235}U enrichments of less than 1.0% (Figure 3-4). However, this is of secondary importance in the processing of irradiated uranium because the ^{238}U is in sufficient concentration with respect to the ^{239}Pu to act as a neutron poison either in solution or in the event of precipitation, so long as the mass ratio is maintained.

Plutonium scrap, Rocky Flats Scrub Alloy (RFSA), or the MK-42 irradiated material is charged into inserts that fit into the four dissolver charging ports. The RFSA and the MK-42 targets are packaged into bundles that are charged into the inserts. The geometry of the insert specifically matches the bundle and size of the well or receptacle in the insert and limits the quantity of material that can be charged. MK-42 targets are packaged into bundles at the reactors, and the RFSA is packaged into bundles in 221-F building. RFSA is assayed nondestructively at the Rocky Flats site prior to shipment and at F-Canyon upon receipt. The inserts used to hold the special irradiated material in the dissolving solution can be probed by the crane to determine the extent of the dissolution. RFSA is configured in a critically safe geometry during dissolution by an insert that has three wells or receptacles for targets (9). The MK-42 inserts have a rectangular well.

CW = Cooling Water
OG = Off-Gas

FIGURE 3-4. Annular Dissolver Charging Assembly

3.2.2.2 Aluminum Cladding Removal and Irradiated Metal Dissolution

The aluminum cladding on the irradiated uranium is removed by reaction with caustic, dissolving the aluminum and exposing the uranium for subsequent dissolution in nitric acid. Both reactions are vigorous, requiring effective temperature and vacuum control to prevent the dissolver system from being pressurized by rapid evolution of gases. These control systems are discussed in more detail in Section 3.2.3. Special materials that are more refractory require the addition of fluoride, in the form of KF, to dissolve. The process chemistry of decladding and dissolution is important to the prevention of undesirable reactions and the prevention of precipitation of fissile material.

Irradiated uranium slugs are charged into a solution of NaNO_3 , which acts to suppress the evolution of H_2 gas during the reaction of caustic with aluminum cladding. Caustic is added at a controlled rate to limit the H_2 evolution rate. In the dissolution of bundles of specialty materials by Hg-catalyzed nitric acid and HF under conditions of low acid ($<0.1\text{M}$), the off-gas stream can contain up to 10% H_2 from:

Hydrogen-air mixtures are flammable in the range of 4.1 to 75 volume percent in dry air and can explode at room temperature over a more limited concentration range. An external source of ignition is required, such as sparks from electrical equipment, overheated bearings, or any open flame. A minimum air purge rate is maintained during decladding and dissolving to dilute the hydrogen concentration to the less-than-the-safe limit (10).

The principal safety consideration in the dissolution of irradiated uranium and the special materials is the selective precipitation of Pu.

Several mechanisms have been identified that would cause the selective precipitation of Pu in the dissolver, depending on the type of material being dissolved. Irradiated uranium is dissolved in two separate batch additions of 50% nitric acid. The acid concentration diminishes as the uranium is dissolved. If the dissolver solution becomes acid deficient, Pu polymer precipitate forms (3).

Acid endpoint is controlled primarily by procedure and confirmed by analyses of each dissolver cut. Additional acid is added if the endpoint acidity is too low.

In the dissolution of RFSA, the possibility of localized high Pu concentration is avoided by air sparging the dissolver solution. This is ensured by the operator's verification of air flow, indication of the air flow on the dissolver off-gas rate meter, and the integrity of the instrumentation (10). In addition, if fluoride is not present, the RFSA Pu fails to dissolve and accumulates on the bottom of the dissolver. Administrative controls confirm that the fluoride has been added. A sample taken after dissolution is analyzed for the presence or absence of fluoride. In the absence of fluoride, the Pu at less than the critical density can be re-dissolved by adding KF to the dissolver solution.

3.2.2.3 Off-Gas Treatment

The off-gas from the dissolving operation is passed through the dissolver condenser, which recovers part of the nitrogen oxides as nitric acid for recycle to the dissolver, then through an off-gas heater reactor. This is a single unit with the bed consisting of silver-coated berl saddles and the helical off-gas coil inside a steam chest. The dissolver off-gas is heated to 185°C, which promotes the reaction and removal of ^{129}I and ^{131}I with the silver on the silver-coated berl saddle packing in the inner vessel. Particulates are removed on the in-cell "Fiberglas"-packed filter. The nitrogen oxides in the off-gas are absorbed in the A-Line nitric acid absorption column.

The alternating operation of aluminum cladding removal and nitric acid dissolution produces ammonia during the decladding and oxides of nitrogen during acid dissolution, which combine to form the potentially explosive compound ammonium nitrate. Ammonia can also react with the silver in the silver reactor to form another explosive, silver azide.

Administratively, the off-gases from the aluminum decladding operation are required to be by-passed around the acid fume exhauster and the nitric acid absorber (11). An ignition temperature of 273°C for silver azide (11) is considerably higher than the maximum silver reactor operating temperature of 200°C that is specified in the operating safety requirements.

Hydrogen deflagration in the dissolver was discussed previously for the Hg-catalyzed dissolution of specialty materials. Hydrogen produced in this situation could result in a deflagration inside the silver reactor, releasing a large fraction of the ^{131}I and ^{129}I absorbed as AgI and AgIO_3 (10). Maintaining sufficient acid and an adequate air purge rate in the dissolver prevents this from occurring.

3.2.2.4 Head End

The head end process clarifies, and concentrates and decontaminates, if needed, the raw metal solution for subsequent solvent extraction processing. Silicon present in aluminum-clad fuel forms colloidal silica or polymeric silicic acids that collect at the phase interface in the mixer-settler, causing emulsion precipitates and lower operating efficiency. Gelatin reacts with silica to form large molecules that agglomerate and precipitate from the solution.

In special campaigns where zirconium and niobium activity are high, a strike with gelatin, to remove silica, and a strike with MnO_2 , to remove fission products, are used.

Freshly precipitated manganese dioxide, by virtue of its large surface area, sorbs a major portion of the zirconium and niobium that is present and agglomerates or occludes suspended matter. The precipitate can then be flocculated by digestion so it will separate readily from the solution.

The raw metal solution, along with any cake washes returned from the previous head end batch, is jetted into the strike tank. Manganous nitrate is added as

a 25% solution to a Mn^{++} concentration of about 0.014M, and the solution is heated to 70 to 75°C. Then enough gelatin is added as a 1% solution to give 50 mg gelatin per liter. Then a 3% potassium permanganate solution is added at a rate of 10 to 15 lb. per minute. The quantity of $KMnO_4$ added is calculated to react with 90% of the manganous ion present. The nitric acid concentration for this step is in the range of 0.5 to 2.5M.

The digested solution containing the suspended manganese dioxide and/or gelatin is centrifuged to yield clarified liquid and a cake by the operations described below. Figure 3-5 shows the construction of the centrifuge and illustrates the various steps. The clarified solution overflows the centrifuge bowl and flows by gravity to the first cycle feed adjustment tank.

The cake is washed five times to remove any residual raw metal solution by successive dilution and skimming; these washes are also sent to the feed adjustment tank. The cake is washed by high-pressure sprays that inject about 30 gal of 0.75% nitric acid into the bowl compartments at 600 psig while the bowl is rotating at 840 rev/min. When the spray cycle is complete, the wash solution is skimmed (840 rev/min) to the centrifuge run tank.

The washing procedure successively dilutes the solution left with the cake and decreases the uranium and plutonium loss to less than 0.1% of the total amount fed to the centrifuge.

After the final wash, the cake is broken up by spraying with dilute acid (10 rev/min), the speed is increased to 140 rev/min, the centrifuge is stopped, and the slurry is jetted from the bowl to the high-activity waste neutralization tank for disposal. The bowl is rinsed with three additional washes, which are transferred to the high-activity waste disposal tank for neutralization. Up to four gelatin cakes are accumulated in the centrifuge before emptying.

Polymerized Pu produced by insufficient acid in the dissolution is a safety concern because the Pu polymer will accumulate in the nongeometrically favorable centrifuge bowl. Each batch of dissolver solution is analyzed for acidity and Pu polymer before the dissolver solution is clarified and/or decontaminated in the head end operation.

Two tanks in the head end receive Pu recovered from scrap in the FB line. A leak during this transfer could collect in a noncritically safe sump and concentrate by evaporation. If this leak is unnoticed, a significant accumulation could occur over a period of time.

Administrative controls require that a specific liquid level be maintained in the sumps at all times. Any increase in liquid level is cause for transfer to a tank for analysis and rework or disposal.

3.2.2.5 Solvent Extraction

Solvent extraction separates one or more components from a mixed solution, depending on their solubility in an immiscible extracting solvent. The Purex solvent extraction process is designed for the recovery of uranium and plutonium from irradiated depleted uranium, decontamination of uranium and

FIGURE 3-5. Head End Centrifuge and Centrifugation Steps

plutonium from fission products, and the separation of the two actinide elements from each other. The SRP process uses seven contactors arranged in three cycles of solvent extraction. Figure 3-6 is a schematic representation of a typical solvent extraction flow sheet and Figure 3-7 is a schematic of the canyon equipment.

The first cycle process consists of three banks: one bank where most of the fission products and Np are removed from the uranium, plutonium, and neptunium; a second bank where the uranium and plutonium are partitioned or separated from each other; and a third bank where the uranium and residual neptunium is stripped out of the organic into the aqueous phase in preparation for additional processing.

The first cycle is followed by two second cycles consisting of two banks each: one set of banks in the second uranium cycle, for the purification of depleted uranium; and a second set of banks in the second plutonium cycle, for the purification of plutonium.

The extractability of the U, Np, and Pu are significantly affected by the nitric acid in the aqueous phase. At higher concentrations, extraction into the organic phase is favored; and at low concentration, U, Np, and Pu will strip or back-extract into the aqueous phase. Valence also affects extractability of Pu and Np. The solvent consists of the extractant tri-n-butyl phosphate in a normal paraffin hydrocarbon diluent. The diluent modifies the physical characteristics of the extractant making it acceptable for mixer-settler operation. Pu is extracted into the organic phase in the IV valence state. Np is extracted in the IV and VI valence states. At a valence III for Pu and valence V for Np, the extractability is very low. Changing the valence with reductants or oxidizers is used to affect actinide separation.

A detailed description of the Purex Process is found in the F-Canyon Systems Analysis (2). The composition and relative process flows are listed in Table 3-1 for the Purex process.

3.2.2.5.1 First Cycle Solvent Extraction: Primary Fission Product Separation

The dissolver solution, after clarification, is transferred to the 1AF centrifugal contactor feed tank and fed continuously to the tenth stage of an 18-stage centrifugal contactor. The scrub stream is introduced at stage 1 and the extractant at stage 18. A reagent stream (1AN) enters at stage 17 to promote rejection of Np to the 1AW. The centrifugal contactor, in contrast to the mixer settlers, which are used as liquid-liquid contactors in the other solvent extraction operations, decants the organic and aqueous phases by a centrifugal force of 400 g's. The use of centrifugal force, rather than gravity, greatly reduces the contact time between the highly radioactive dissolver solution and the organic. This results in reduced radiation exposure to the organic and reduced organic degradation. The U, Pu, and a small fraction of the fission products are extracted by concurrent contacting in each stage. The interstage flow is counter-current. Most of the fission products are scrubbed back out of the organic in the scrub section, and the extractant containing U and Pu exits the centrifugal contactor at stage 1. The aqueous waste stream containing fission products, americium, Np, non-radioactive chemical contaminants, and a trace of the product elements, exits

FIGURE 3-6. Solvent Extraction Flowsheet

FIGURE 3-7. Approximate Location of F-Canyon Equipment

TABLE 3-1. Purex Process

Stream	Relative Flow	Composition
1AF	100	1.0M HNO ₃ , 1.22M U, Pu
1AX	333	30% TBP in n-paraffin hydrocarbon
1AS*	66	3.0M HNO ₃
1AN	0.6	5.3M NaNO ₂
1AW	166	1.6M HNO ₃ , 0.01M Al(NO ₃) ₃ , 0.02M NaNO ₂
1AP	348	0.39M U, 0.2M HNO ₃ , Pu
1BF	348	0.39M U, 0.2M HNO ₃ , Pu
1BX**	45	0.11M HNO ₃ , 0.14M Fe(NH ₂ SO ₃) ₂
1BS	88	30% TBP in n-paraffin hydrocarbon
1BP	47	1.2M HNO ₃ , 0.06M Fe(NH ₂ SO ₃) ₂ , Pu
1BU	434	0.31M U, 0.01M HNO ₃
1CF	434	0.31M U, 0.01M HNO ₃
1CX	406	0.08M HNO ₃
1CW	421	30% TBP in n-paraffin hydrocarbon
1CU	415	0.29M U, 0.08M HNO ₃
1DF	100	1.85M U, 1.0M HNO ₃ , Fe(NH ₂ SO ₃) ₂
1DX	468	30% TBP in n-paraffin hydrocarbon
1DS'	29	3.0M HNO ₃
1DS	30	H ₂ O
1DW	142	1.8M HNO ₃ , Fe(NH ₂ SO ₃) ₂
1DU	348	0.39M U, 0.1M HNO ₃
1EF	348	0.39M U, 0.1M HNO ₃
1EW	468	30% TBP in n-paraffin hydrocarbon
1EX	478	0.08M HNO ₃
1EU	478	0.30M U, 0.10M HNO ₃
2AF	75	4M HNO ₃ , Pu, Fe, Na, SO ₄
2AS	10	0.7M HNO ₃
2AX	22	30% TBP in n-paraffin hydrocarbon
2AW	97	3M HNO ₃ , Fe, SO ₄ , Na
2AP	75	0.1M HNO ₃ , Pu
2BF	75	0.1M HNO ₃ , Pu
2BW	75	30% TBP in n-paraffin hydrocarbon
2BX	14	0.01M HNO ₃ , 0.05M NH ₂ OH·1/2H ₂ SO ₄
2BP	14	0.35M HNO ₃ , 0.05M NH ₂ OH·1/2H ₂ SO ₄ , Pu

*Split scrub flowsheet being tested: 1AS = 3M HNO₃, 1AS' = 0.08M HNO₃

**Alternate flowsheet: 1BX = 0.08M HNO₃, 0.08M Fe(NH₂SO₃)₂, 0.056M NH₂OH·HNO₃

the contactor at stage 18. This aqueous waste stream typically contains more than 99.99% of the nonvolatile fission products and americium and is collected in a receiver tank in the waste and concentration system.

The 1A centrifugal contactor performance is determined by the uranium concentration in the organic solvent and by the ratio of the aqueous scrub-stream (1AS) flow to the organic product (1AP) flow. Product losses are measured by analyzing frequent samples of the waste.

3.2.2.5.2 Partitioning: Separation of Pu from U

The organic product solution, 1AP, containing the U and Pu, is pumped to stage 11 mixer of the 1B mixer-settler. In this liquid-liquid contactor, the two phases, organic and aqueous, after passing counter-currently through a paddle agitator, enter a gravity settling chamber where the phases separate into two distinct phases that are directed into counter-current direction to the next stage. The aqueous stream flowing out of the feed stage (stage 11) is scrubbed with organic in stage 12, and the organic is contacted with stripping solution in stage 10. The nitric acid strip solution contains a reductant such as ferrous sulfamate. The plutonium is back-extracted into the aqueous phase, where it is reduced to the near-inextractable plutonium (III) valence and exits the 1B mixer settler from stage 18. This aqueous product solution is scrubbed with fresh organic in the scrub section (stages 18 through 11) to remove residual uranium. The scrubbed aqueous plutonium solution (1BP) is then routed to the second plutonium cycle feed (2AF) tank for additional decontamination from uranium, neptunium, and fission products.

The organic from the 1B bank, containing uranium, residual neptunium, and a trace of Pu, flows by gravity to the paddle mixer on stage 12 of the 12-stage 1C mixer-settler bank, where it is contacted with weak nitric acid. The uranium is back-extracted or stripped into a counter-current flowing aqueous solution that is introduced at stage 1 of the 1C bank. The organic raffinate from the 1C bank is routed to solvent recovery for washing and recycle to the process. The aqueous stream emerging from the 1C mixer-settler bank flows to a decanter to remove gross amounts of entrained solvent that would create an explosion hazard in the evaporator operation. Solvent disengaged from the aqueous phase in a decanter is continuously transferred to rerun or returned, by means of an air lift, to the bank.

The aqueous uranium-bearing stream 1CU from the 1C mixer settler bank is continuously received in the 1CU evaporator feed tank and pumped continuously to the 1CU evaporator. The 1CU evaporator is a continuous thermosyphon type evaporator with de-entrainment column, a reboiler, and a condenser (Figure 3-8). Concentrate is recirculated from the column through a duct that connects the de-entrainment column with the reboiler. The concentrate is diluted with feed solution below the tubes and passed up through the tubes, where it is heated with steam. The vapor and unevaporated liquid flow from the tube bundle through the duct. The large droplets are deflected into the liquid, and the vapors flow up through three bubble cap trays. The lower and upper trays serve as demisters and are operated dry. The middle tray is operated wet, and it can be supplied with water for scrubbing. The vapor leaves the top tray and enters the condenser, which consists of a center hollow section and an annular section containing tubes, tube sheets, and baffles. The vapors

FIGURE 3-8. Schematic of Continuous Evaporator

from the column flow up through the central riser, across the upper tube sheet, and down through the tubes. The vapors are condensed in the 3/4-in tubes, collected in the 8-in annular space at the bottom of the condenser, and flow by gravity to a condensate run tank. The overhead condensate is pumped to the seepage basin.

A weir in the lower portion of the column controls the concentrate liquid level. The concentrate flows by gravity from the weir to the bottoms tank. The concentrate is then pumped to the second uranium cycle tank (1DF).

3.2.2.5.3 Second Uranium Cycle Solvent Extraction

The concentrated uranium product overflowing the weir of the 1CU concentrator is collected in a product receiver and pumped to a second uranium cycle feed where ferrous sulfamate is added in batches from a third-level feed tank. The 1DF is delivered continuously at a controlled rate to the mixing chamber of the ninth stage of the 18-stage 1D mixer settler bank.

Operating performance of the 1D bank is controlled to obtain a uranium product of the highest possible purity. Pu is removed by valence change to inextractable III valence, caused by the ferrous sulfamate reductant, and leaves the mixer settler bank at stage 18 in the 1DW waste raffinate. A small amount of Np and the remaining fission products are separated from the uranium by loading the organic with U, forcing the less extractable Np and some fission products into the aqueous waste. With acid introduced at stage 4, the organic phase is scrubbed to remove any fission product and trace impurities into the aqueous waste. Another scrub stream 1DS-water introduced at stage 1 back-extracts some of the acid out of the U-bearing organic (1DU). The 1DU flows by gravity to the mixing chamber of stage 12 on the 12-stage 1E mixer settler. The uranium is stripped or back-extracted into a dilute acid which is pumped to A-Line for further processing.

3.2.2.5.4 Second Plutonium Cycle Solvent Extraction

The function of the second Pu cycle is to affect further decontamination of the plutonium product from residual fission products. The aqueous plutonium leaving the 1B contactor is collected in an accumulation tank and each batch chemically adjusted by the addition of sodium nitrite to reoxidize the plutonium to the extractable IV valence. The 1BP aqueous product is also diluted to Pu concentration that is acceptable for processing in the second Pu cycle. The acidity of each batch of feed is adjusted to enhance the extractability of the Pu.

The Pu is separated from most of the residual fission products in the 2A, 16-stage mixer-settler. The aqueous waste stream, 2AW, is the primary feed to the low activity continuous waste evaporator. The feed stream is introduced in the mixer of stage 8, the acid scrub at stage 1, and the organic extractant at stage 16. The organic extractant with Pu is introduced at stage 16 and the strip solution at stage 1.

The 2B bank, another 16 stage mixer-settler, uses a stripping solution (2BX) consisting of hydroxylamine nitrate and dilute nitric acid to reduce the Pu

to the relatively inextractable III valence. The presence of the reductant increases the stripping effectiveness. Control is affected by chemical analysis of each batch of 2BX and the monitoring the 2B bank product, 2BP, sample analysis. A significant change in the 2BP Pu concentration is indicative of a shift in the feed to solvent ratio, or a change in acid or reductant concentration in the 2B bank. The product stream from the 2B bank, 2BP, is collected, then transferred to the FB-Line for subsequent processing. Before the batch is transferred, the operator verifies that the analytical results are recorded and that Pu concentration is less than 5g/l for criticality control. The organic, stripped of Pu (2BW), is sent to solvent recovery where it is washed for re-use.

3.2.2.5.5 Safety Considerations - Solvent Extraction

Nuclear safety in solvent extraction is based on concentration control, i.e., the solvent extraction flowsheet is designed to maintain the concentration of fissile material well below the critical concentration because the equipment is not geometrically favorable. This is done by limiting the feed concentration, the extractant concentration, and the ratio of feed to extractant. Other methods used to maintain safe concentration are: 1) rigid administrative control of the operation of the process, 2) frequent monitoring, sampling, and material balance checks, and 3) the use of continuous nuclear monitors (see Section 3.2.3).

1A Bank. Criticality prevention in the 1A bank is ensured by maintaining the organic solvent-to-aqueous feed ratio and the nitric acid concentration of the 1AS at run sheet concentration to maintain a safe solution concentration of Pu at less than 6.75 g/l (8). To assure that plutonium is always less than 6.75 g/l, the feed to the bank must be brought in at a concentration less than that value and not permitted to concentrate in the bank. In the Purex 1A contactor there are no known mechanisms by which an abnormal reflux of Pu could occur to give an unsafe accumulation.

1B Bank. In the 1B bank, the plutonium is separated from the uranium by an adjustment in Pu valence. Failure to effectively maintain a reducing aqueous phase will result in a separation failure. An excessive amount of nitrous acid in the 1AP will destroy the ferrous sulfamate which is used to reduce the Pu to the inextractable III valence. A gradient in the oxidation-reducing potential is established across the mixer-settler bank with a reducing aqueous phase existing at the point of scrub introduction and an oxidizing phase around the introduction of the feed. The Pu is reduced and stripped or back-extracted into the aqueous phase near the introduction of the scrub and then re-extracted into the organic phase due to the presence of nitrous acid at the feed point. The Pu accumulates in the mixer-settler because of the extraction and stripping that is occurring. This condition, called refluxing, is also induced by deficiency of ferrous sulfamate. Unless this condition is corrected, the critical concentration will be approached in one of the stages in a few hours. The worst envisioned reflux condition in the 1B mixer-settlers would involve the complete destruction of the ferrous sulfamate in the scrub section. This would cause oxidation of plutonium to the extractable (+IV) valence, thus preventing the Pu from leaving with the aqueous 1BP

stream. Concurrently, the ferrous ion entering in the 1BX stream would reduce the Pu to the inextractable (+III) and prevent its exiting with the uranium-bearing organic stream. An internal reflux of Pu would then be initiated and continued until the concentration increased sufficiently to consume all of the ferrous ion entering the bank. At that point, all additional Pu would exit with the organic 1BU stream.

Control of the uranium-plutonium partition is maintained by providing sufficient reductant and holding time to ensure complete reduction of the plutonium. Each 1BX make-up batch is sampled and analyzed, and the ferrous reductant concentration is ascertained to be within the control limits. The reductant flow rate is measured and compared to make-up tank depletion rates. This flow rate is controlled above a minimum acceptable level. Regular samples of the 1BP are analyzed for Pu concentration; any steady decline in Pu concentration would be recognized as indicating a possible accumulation of Pu developing in the 1B bank. Two neutron monitors in the 1B bank are used to detect an accumulation of plutonium.

1C Bank. Introducing Pu into the 1C bank is a possibility if the separation of Pu from U in the 1B bank fails and the Pu exits the 1B bank with the 1BU. Prevention of Pu entering the 1C bank and the second U cycle is accomplished by close surveillance and monitoring of 1B bank operations.

2A Bank. Control of the 2A bank operation is directed toward avoiding high losses and preventing an accumulation of Pu. The operation is sensitive to acid concentration and sufficient nitrite to oxidize the Pu to an extractable valence. Insufficient nitrite would result in incomplete oxidation of the Pu and high Pu loss, while an excess would enter the 2B bank and upset Pu stripping because nitrite is readily extractable into the organic.

The amount of nitrite added to the 2AF tank is controlled by chemical analysis of the sodium nitrite stock solution. A batch of sodium nitrite is added to each batch of feed solution in the 2AF tank. The contents of the 2AF tank are then sampled and the samples tested with an indicator for excess nitrite. The nitric acid addition and the dilution of Pu concentration is based on the sample analysis of the 1BP and is added automatically after the sample results of the 1BP are entered into the distributed control system (see Section 3.2.3.4.3).

2B Bank. Close adherence to the flowsheet is the primary controls. The primary corrective measure to a process upset is to shut the process down to determine the cause of the upset.

1D, 1E Banks. The trace quantity of Pu is effectively removed from U-bearing streams to make accumulation of more than gram quantities of Pu highly unlikely.

3.2.2.5.6 Solvent Flammability

The Purex process uses 30 volume percent TBP organic solvent in normal paraffin hydrocarbon (NPH).

The TBP is considerably less volatile (flashpoint 146°C) than NPH (nominal flashpoint 90°C). The properties of the NPH diluent are, therefore, considered to be controlling with respect to solvent flammability.

The NPH is a mixture of five saturated, straight-chain hydrocarbons composed of from 10 to 14 carbon atoms per molecule. The purchase specifications require a minimum of 99 volume percent of the C₁₀ to C₁₄ constituents and a minimum flashpoint of 70°C. The range of concentration for flammability of NPH vapor is from 0.6 to 6.0%. The calculated flammable limits for dry diluent, expressed in terms of temperature, are then 70 to 120°C. For kerosene-type hydrocarbons, gaseous mixtures containing more than 33% water vapor are not flammable under any conditions. When water vapor is present as a separate phase, dilution to 33% water vapor is reached at 72°C; thus, the flammability region of the NPH-water-air system is from 70 to 72°C.

In addition to the potential for vapor flammability, consideration has to be given to the potential of mist flammability. Solvent mist flammability may occur in the concentration range from 3 to 60% and at all temperatures up to 90°C; at this temperature, mist flammability is limited by lack of sufficient oxygen in the mixture.

Air jets are used in the tank sampling system. Studies were conducted to investigate the characteristics of solvent aerosols that result from the sampling system. The highest concentration of aerosol obtained in these experiments approached 2.0 mg NPH/liter of vapor at 60°C (12). The equilibrium vapor-phase concentration of NPH was calculated to be 27.25 mg NPH/liter at 60°C (assuming 100% NPH and an ideal gas). Therefore, total NPH concentration in a solvent tank headspace could approach 30 mg/liter, which is below the lower flammable limit of 46 mg/liter for hydrocarbon chains C₈ and greater. Based on these studies, a flammable concentration safety margin exists (12).

3.2.2.6 Anion Exchange: Pu and Np Recovery

Aqueous waste from the first solvent extraction cycle (1AW) and second uranium cycle (1DW) contain Np, Pu, and nitric acid. The Np and Pu are recovered by ion exchange on an anion resin in the Primary Recovery Column (PRC). These wastes are evaporated with a continuous evaporator and the high activity waste (HAW) concentrate is collected in the feed tank for the PRC.

Depending on their Pu and Np content, the eluted actinides from the PRC are sent to the first solvent extraction cycle or to the warm canyon for further decontamination and separation of the Pu and Np in Frame II-F.

3.2.2.6.1 Primary Recovery Column

The recovery of neptunium and plutonium from waste concentrates is carried out in the PRC, a three-stage, agitated-bed, ion-exchange column containing anion resin (Figure 3-9) in two of the three stages. The HAW concentrate is batch-fed through the three-stage column. In absorption of Np and Pu, nitrate ions are displaced from the resin by anionic nitrate complexes of Np(IV) and Pu(IV). The resin effectively absorbs the Np and Pu only if the feed solution has a sufficient concentration (6 to 9M) of nitrate ions and if the Np and Pu are in valence IV.

Upon completion of the feed step, the resin beds containing absorbed Np and Pu are washed with 8M nitric acid to remove most of the fission products that remain on the resin. Subsequently, the actinides (Np and Pu) are eluted with dilute (2.2%) nitric acid. The absence of excess nitrate ions breaks the Np and Pu nitrate complexes, and the Np and Pu are stripped from the resin.

3.2.2.6.2 Frame II-F: Neptunium Recovery

The Np and Pu, recovered from the Purex process waste concentrations in the PRC in the hot canyon, require further decontamination before they can be processed in unshielded areas. The primary anion exchange recovery of Np and Pu from HAW concentrate achieves a decontamination factor (DF) of 1000 from fission products and a separation factor from most other stable elements of 10,000. The desired product purity requires a DF from fission products of approximately 10^9 . In addition to separation from fission products, it is necessary to remove ^{234}Th , Pu, and ^{232}Th from the Np product. ^{234}Th is removed to decrease the activity problem of handling ^{237}Np . Pu and ^{232}Th are removed from ^{237}Np to produce a chemically pure product.

In this purification process (Frame II-F, see Figure 3-10), two cycles of anion exchange are used to provide the required fission product decontamination. In each cycle, the Np and Pu are sorbed on the resin, as in the PRC; the raffinate and column washes are sent to the HAW evaporator. Then, Np and Pu are washed selectively from the resin by:

1. A partitioning wash containing 0.05M $\text{Fe}(\text{NH}_2\text{SO}_3)_2$ - 0.05M $\text{N}_2\text{H}_5\text{NO}_3$ - 5.2M HNO_3 to remove the Pu for recycle to the HAW evaporator or to the first solvent extraction cycle, and
2. An elution wash containing dilute (2.2%) HNO_3 to remove the Np product.

Cation exchange resin is used to remove the ^{232}Th and ^{234}Th from the Np product solution. After the neptunium passes through a cation exchange column, its purification is complete. It is then sent to the product loading station, where it is loaded in shipping containers and transported to the HB-Line in Building 221-H.

FIGURE 3-9. Primary Recovery Column

FIGURE 3-10. Frame II-F

3.2.2.6.3 Safety Considerations - High Activity Waste

The processing of HAW concentrate with anionic resin for the recovery of Pu and Np is cause for concern because of the explosive and flammable nature of the resin, which is increased by radiation exposure. The data from closed bomb tests show that the dried nitrate form of anion resin is explosive and could be detonated in the falling hammer test (3). The OSR (8) specifies that the strong anion exchange resin qualified by laboratory tests for thermal stability and process operation shall be covered with water or dilute acid (<1 molar). Keeping the resin wet significantly reduces the hazard. Under these conditions, the resin did not react to closed bomb test temperatures up to 200°C (3). The radiation exposure received by the resin is administratively controlled to limit the resin service to a period of time that prevents the resin from becoming hazardous (8). The explosive limit, or ignition point, is temperature-dependent. Processing conditions never require temperatures near the ignition temperature; however, process temperatures are limited to 60°C (8). High concentrations of nitric acid (>9M) and other chemicals that could react abnormally or react to produce explosive gas mixtures are excluded.

Administrative controls prevent an unsafe accumulation of fissile material and limit the radiation exposure of the anion resin.

In April 1986, equipment changes are planned to improve accountability of Pu for criticality prevention in a system of non-geometrically favorable equipment. Accumulation of Pu within the system is detected by strict accounting of Pu in the product, waste and recycle.

A batch of HAW concentrate accumulating in the evaporator bottoms tank is transferred to the PRC feed tank for processing. Batch size is restricted to a volume with a Pu content within the operational specification for safe operation.

The batch of waste is sampled and analyzed for Pu then loaded on the resin beds and the raffinate waste collected in a waste tank for evaporation. The decontamination scrub is used to remove fission products from the bed, and portions of the eluted Pu product are collected in the recycle tank. The concentrated product is collected in separate vessel. All waste and product batches are sampled and analyzed for Pu; differences between input and output is cause for concern prompting possibly a flush to remove a Pu accumulation.

The PRC (Figure 3-9) and the II-F Frame (Figure 3-10) are similar with respect to use of similar administrative controls to prevent an unsafe accumulation of Pu in the system and in limiting the resin exposure. In the II-F Frame operation, the absence of highly radioactive fission products allows much longer use of the resin than in the PRC. Each batch of feed processed is restricted to a volume consistent with the Pu and total actinide mass restriction. The volume of solution that is transferred to the II-F Frame feed tank must satisfy these limits. Pu processed through the system is strictly accounted for to prevent an accumulation. The Pu eluted from the ion exchanger in the partitioning step, the Pu in the decontamination scrub wash, and the Pu in the Np elution, are checked against the Pu in the feed. If there is a difference between the Pu in the feed and Pu accounted for in the previously mentioned streams, then the Pu processed in the succeeding batches is limited to the normal Pu limit minus the difference.

3.2.2.7 Evaporation

3.2.2.7.1 General

Continuous natural recirculation or batch evaporators in 221-F are used to:

1. Increase uranium concentration available in feed streams for operational efficiency.

Evaporator ServiceEvaporator No.

Rerun	16.1E (batch)
1CU, 2nd Uranium	17.7E (continuous)

2. Remove excess water and economize in waste storage.

Evaporator ServiceEvaporator No.

Low Activity Waste	8.5E (continuous)
High Activity Waste (1st Stage)	9.3E (continuous)
High Activity Waste (2nd Stage)	9.1E (continuous)
Rerun	16.1E (batch)
Laboratory Waste	18.6E (batch)

3. Remove excess acid and economize in waste storage.

Evaporator ServiceEvaporator No.

High Activity Waste (1st Stage)	9.3E (continuous)
Rerun	16.1E (batch)
Laboratory Waste	18.6E (batch)
Low Activity Waste	8.5E (continuous)
Acid Strip Evaporator	16.1E (batch)
Low Activity Waste	7.6E (batch)
Low Activity Waste	7.7E (batch)

The functional relationship of 1CU evaporator to the second uranium cycle, the HAW to the PRC and the low activity waste (LAW) evaporator to the second plutonium cycle have been mentioned in respective sections on solvent extraction and anion exchange. The 1CU evaporator, the first and second stage HAW evaporator, and the first stage LAW are continuous evaporators that in operation are similar to the description of operation of the 1CU evaporator (Section 3.2.2.5.2).

The waste evaporators reduce the volume of waste going to high level waste storage tanks and recovery of the nitric acid in the waste. It is economical to recover nitric acid from evaporator overheads when the concentration of acid is above 2%.

3.2.2.7.2 High Activity Waste Evaporation

Aqueous waste from the first extraction cycle and second uranium cycle, acid-strip condensate, ventilation system sumps via tank 805, and concentrate from the second-stage HAW evaporator are combined in the HAW feed tank, and evaporated in the first-stage (9.3E) continuous evaporator. When the Special Product Recovery System is operating, column decontamination washes and Frame II-F non-resin-bearing wastes are recycled to the HAW feed tank.

Unless limited by foaming, boilup can reach a maximum rate of 25,000 pounds per hour. Antifoam or other solutions can be added to the evaporator by a portable pump on the second level or through an addition line from the feed gallery tanks.

Condensate from 9.3E flows by gravity to the second-stage (9.1E) continuous evaporator, where it is again evaporated. Condensate from the double evaporation is transferred to the acid recovery unit in the 211-F outside facility for recovery of nitric acid. If the 9.1E condensate contains excessive radioactive contamination, it is recycled back to the HAW feed tank instead of being transferred to acid recovery. The bottoms concentrate from the first stage HAW evaporator overflows continuously to a bottoms tank and is then transferred in batches to the PRC for recovery of Pu and Np by anion exchange. The ion exchange raffinate is transferred to the Acid-Strip Evaporator 16.1 for water stripping. This operation is expected to begin in April 1986. Concentrate from the second-stage evaporator (9.1E) is periodically recycled to the HAW feed tank.

3.2.2.7.3 Low Activity Waste Evaporation

Batch evaporators 7.6E and 7.7E are used to strip acid from the strong acid (25 to 30%) waste concentrate from the continuous LAW evaporator (8.5E). The acidic concentrate from 8.5E is sent to the two batch evaporators (7.6E and 7.7E), which are also fed water and the waste from solvent washing. The acid vapors go to acid recovery in the 211-F outside facilities.

Batch evaporator operation is controlled during stripping by maintaining the specific gravity and acid concentration in the boiling solution within operating limits. After about 16 hr of stripping at 3000 lb of water per hour, the stripped concentrate is transferred to the run tank, neutralized, and then transferred to Building 241-F.

The acidic overhead condensate from the batch evaporators is returned to the continuous evaporator feed tanks and is used to dilute the 18% acid waste from the second product cycle (2AW). By subsequent evaporation in the continuous evaporator, the acid from the stripping operation reaches the acid recovery unit (in Building 211-F) with the acidic overhead from the continuous evaporator.

3.2.2.7.4 Batch Evaporator Operation

Some of the evaporators in F-Canyon are batch evaporators with a pot, evaporator coils, and a condenser in a column that can be attached and

detached by the overhead crane (Figure 3-11). Feed solution is transferred by jets to the evaporator at a controlled rate, through a liquid level control.

Concentrate in the body of the evaporator passes down through a 2-ft 4-in diameter tube around which the coils are stacked and the solution is heated as it passes up through the coils. The same coils are supplied with water for cooling the concentrate. The vapor and entrained droplets bubble through liquid on three bubble cap trays with 3-in OD bubble caps spaced 18 in apart in the evaporator column. The trays serve as a de-entrainer for droplets and scrubbers for particulates. Condensate is diverted to the top trays, which overflow by gravity through a downcomer to the tray below it.

The vapor leaves the top tray and enters a hollow center section with an annulus of cooling coils. The vapors are condensed and collect at the bottom of the annular section. Non-condensable vapors are vented by a 6-in line through an overflow line to the process vent system; any sudden pressure within an evaporator is relieved through the overflow line. Part of the condensate is diverted as reflux water to the bubble cap trays and the remainder drains to the condensate collection tank. When the contents of the evaporator bottoms reach the desired concentration, the evaporator is shut down and the bottoms are transferred.

3.2.2.7.5 Safety Considerations - Evaporators

Vigorous exothermic reactions can occur between nitric acid or heavy metal and some TBP-diluent mixtures under conditions that might arise during the concentration of the product solution. Violent reactions resulting in equipment damage were experienced during early development work (13,14).

While the exact reactions have not been determined, certain characteristics have been observed and the conditions under which reactions can be expected to occur have been defined, as described below.

- The explosions or rapid reactions have involved the evolution of nitrogen oxides, and two incidents occurred only after concentration proceeded to the point of incipient calcination of the uranyl nitrate ($>135^{\circ}\text{C}$).
- In each incident, solvent was present in a dense, second organic phase in the concentrator bottoms liquid. This second organic phase, commonly called red oil, is the intermediate product of the nitration reactions and is a complex mixture of degraded organic compounds and nitrate complexes.
- In the laboratory studies, the temperature at which the reactions commenced varied (as evidenced by the evolution of nitrogen oxides), depending on the nature of the diluent and whether or not heavy-metal nitrates were present.

The 1CU continuous evaporator operation is controlled to prevent the introduction of solvent and to prevent exceeding the safety limits (see Section 3.2.3.4.4). Entrained solvent in the evaporator feed is limited to 0.5 volume percent by analysis; reboiler steam pressure is limited to 25 psig

DELETED VERSION

FIGURE 3-11. Standard Coil Batch Evaporator and Column

limiting evaporator temperature. Evaporator feeds are decanted prior to introduction into the evaporator reboiler.

Nuclear safety in the evaporators is related to operations in the solvent extraction. High sustained losses from the 1A centrifugal contactor bank could be a hazard in the HAW evaporator. A similar condition in the 2A mixer settler bank could pose a similar problem in the LAW evaporator. A failure of the 1B bank to separate Pu from U would introduce Pu into the 1CU evaporator.

In all cases, the problem would be manifested in the solvent extraction banks and the evaporator feed tanks. Routine sample analysis would warn of the problem.

3.2.3 Instrumentation and Controls

Each process, dissolution, precipitation, extraction, ion exchange, and waste evaporation within the 221-F Building, has control parameters unique to that process to maintain the process within a safe regime of operation with respect to personnel, equipment, and to efficiently produce a satisfactory product. The limits of operation are defined in Operational Safety Requirements (OSRs) and down-graded appropriately in alarm settings to give appropriate margins of error. The durability and maintainability or replacement in a radiation environment are factors in selection. Redundancy and service by emergency power are used where controls may be critical to a process operation.

3.2.3.1 Dissolver

The dissolution of uranium metal with nitric acid is an exothermic reaction that is controlled by regulating the dissolver solution temperature by the flow of cooling water through the dissolver coils. The reaction is vigorous, producing a large quantity of nitrogen oxides that are removed by the lobe blowers at a rate sufficient to maintain a vacuum on the dissolver and dissolver off-gas system. Loss of vacuum is a concern because of the vessel pressurization and the potential for contamination spread outside the confines of the canyon by way of pneumatic instruments and service lines. This is mitigated by a safety interlock that initiates the operation of a vacuum jet assist to the lobe blower if the vacuum is lost.

The mass of Pu and U dissolved in a single batch dissolution must be limited. This is accomplished by administrative control, which specifies the amount of nitric acid added to the dissolver and the final end point concentration (indicated by specific gravity measurement).

All coils that are used for both heating and cooling, like the dissolver coils, are subjected to increased rates of corrosion. The dissolver coils are pressurized with air by a Cash regulator any time the pressure in the coil drops to 12 psi. The air leaks out of the coil, rather than radioactive solution leaking into the coil to contaminate the segregated cooling water.

3.2.3.2 Dissolver Off-Gas

^{129}I and ^{131}I present in the dissolver off-gas are removed in the F-Canyon by packed beds of silver-coated berl saddles. The reaction efficiency is affected by temperature, the primary control parameter.

In mercury catalyzed dissolution of specialty materials, H_2 evolution can be a problem under certain dissolver conditions. For dilution purposes, a controlled air purge is maintained at minimum level.

3.2.3.3 Head End

Before solvent extraction, the raw metal solution is clarified with gelatin to remove impurities that form emulsions that cause upsets in mixer-settler operation (2). Agglomeration of the silica impurities is temperature sensitive. The temperature of the raw metal solution is controlled at 70°C during the gelatin addition and digestion period. A safety interlock prevents Ru volatilization by shutting off the steam to the coil when the tank temperature exceeds the safe limit.

When the flocculation is complete, the raw metal solution and precipitate are clarified by centrifugation. Feed rate control is important to affecting a good solids-liquid separation; at rates higher than 20 gal/min, gelatinous precipitate is lost to the supernatant liquid. The precipitate is deposited as a cake on the bowl of the centrifuge. When centrifugation is complete, about 40 gal of raw metal solution remains in the centrifuge. A solution skimmer is moved into the path of the rotating solution and the centrifugal velocity forces solution through the skimmer pipe to the centrifuge casing. Restrictions on controls for the movement of the skimmer pipe prevent it from entering the cake and damaging the centrifuge.

3.2.3.4 First Cycle Solvent Extraction

In the centrifugal contactor (1A) and the mixer-settler banks, 1B, 2A, 2B, 1C, 1D, and 1E, the TBP-diluent extraction is intimately mixed with an aqueous phase, either to affect an extraction of a valued radionuclide or to back-extract or strip the radionuclide into an aqueous phase for further processing. After extraction or stripping, the aqueous and organic phases coalesce into two separate phases with a common interface. Controlling the interface by adjusting the back pressure on overflow weirs by applying air pressure directs the phases, aqueous or organic, in the proper direction. Failure or improper control can result in the aqueous phase following the organic phase or the organic phase following the aqueous phase. The consequence is both a processing problem and a safety concern. The organic in the aqueous stream is an explosion hazard in evaporators without the proper controls in effect (see Section 3.2.2.7). To prevent the organic which is flammable from exceeding the flash point of the diluent, limiting control settings have been established for the 1A feed tank and the solvent handling system.

3.2.3.4.1 1B Bank

The organic stream, 1AP from 1A centrifugal contactor, containing the U and Pu, is pumped to the 1B bank where the U is separated from the Pu by contacting the 1AP with a dilute stream of nitric acid and reducing agent. An alarm sounds if the flow of the reducing stream falls below established limits. The reducing agent effects a valence change in the Pu causing the Pu to favor the aqueous phase. This aqueous phase containing the Pu is scrubbed with a fresh organic introduced to the 1B bank to remove any residual U.

Conditions can develop where the Pu accumulates within the 1B bank because the reductant is ineffective. The Pu is stripped into the aqueous, then re-extracted into the organic scrub; the separation fails, and Pu accumulates posing a criticality hazard.

To detect this condition a neutron sensor is installed in the 1B bank. If the neutron readings in the bank exceed a specific level, an alarm is sounded and a safety interlock shuts off the feed to the 1A contactor. This curtails the Pu feeding to the 1B bank.

3.2.3.4.2 1C Bank

The uranium-bearing organic from the 1B bank, 1BU, is stripped of uranium in the 1C mixer-settler bank. The effectiveness of this transfer of uranium from the organic phase to the aqueous phase is enhanced by controlling the scrub solution at 55°C. A high scrub temperature causes an alarm before the flash point of the diluent in the 1BU is reached.

3.2.3.4.3 Second Plutonium Cycle

The Pu bearing aqueous stream from the 1B bank is sent to the second Pu cycle for separation of the Pu from remaining fission product and metal impurities. A microprocessor based distributed control system with the control logic based on standard operating procedure was installed in December, 1983. Instrument readings noted every hour by operators controlling the process with pneumatic instrumentation are now scanned continuously by the distributed control system. Every 20 minutes, process operations permitting vessel agitators are shut down, contents allowed to come to rest, then instruments (liquid level and specific gravity) are archived. The actual process I/O and sequence information is continuously scanned and archived for retrieval as needed via the operators consoles.

Accumulations of Pu within the second Pu cycle and conditions that could cause an accumulation are a safety concern that the distributed control system can monitor and adjust the process automatically to stay within safe operating boundaries. The distributed control system is provided with a backup panel for the safe shutdown of the second Pu cycle on the occasion that the distributed control system is inoperable. The backup panel shuts off pumps and steam, and checks the tanks in the second Pu cycle for any unusual change in the liquid level. The memory of the distributed control system is maintained by battery during a power outage.

Batches of Pu laden aqueous from the 1B bank are adjusted for extraction in the 2A bank by the addition of sodium nitrite and nitric acid. Sample analysis of a batch of feed for second Pu cycle processing is fed to the distributed control system which checks to determine if the analysis falls within the established limits. The distributed control system calculates the addition, checks the chemical addition tanks for adequate supply, determines if sufficient room is available in the receiving tank, then initiates and terminates the addition. The Pu concentration is diluted to prevent an unacceptable concentration of Pu in the second cycle solvent extraction. After the additions are made, the distributed control system checks for a difference between depletion in the stock make-up tanks and the receiving tanks. These same functions now performed by the distributed control system were performed previously by administrative control.

The 2A bank scrub, 2AS, acidity is measured continuously by conductivity probe. In the event of low acid concentration in the 2AS, a situation that would cause high Pu losses, the Second Plutonium Cycle is shut down.

Two neutron monitors in the 2A bank and 2B bank, showing high neutron readings indicating high Pu concentration, cause a shutdown by an interlock with the feed stream to the process. The distributed control system can go the additional step of initiating flushing of the system if the operator response is inadequate to lower the high neutron readings.

Batch flow rates within the second Pu cycle are measured by flowmeter and by calculating the depletion rate from a batch tank. Abnormal flow rate is alarmed.

3.2.3.4.4 Second Uranium Cycle

The uranium separated from Pu in the 1B bank is constantly received in the second U cycle for final purification before transfer to A-Line for conversion to uranium trioxide powder. A continuous evaporator and two banks, one for extraction and one for stripping, are the primary components controlled by the microprocessor-based distributed control system.

The configuration of the distributed control system is based on standard operating procedures of the second U cycle and is less complex than second Pu cycle because of the lack of criticality concerns. Most of the safety interlocks in the second U cycle are found on the instrument system associated with the 1CU evaporator. This continuous evaporator increases the uranium concentration to a level acceptable for feed to the 1D bank. The distributed control system has the flexibility for readily changing from one control mode to another. A cascade control scheme where the specific gravity regulates the steam supply has proven effective in bringing the evaporator bottoms up to the desired specific gravity. The level in the evaporator is allowed to rise to the overflow weir and flow into the 1DF feed tank. With the specific gravity of the evaporator bottoms stabilized at the desired level, the evaporator control mode can be changed to maintain a constant liquid level in the evaporator feed tank.

The other continuous evaporators (Section 3.2.2.7.1) are controlled with pneumatic instrumentation to maintain the evaporation bottoms specific gravity

at the desired level. The pneumatic control system lacks the flexibility and smooth response that has been demonstrated by the use of the distributed control system.

A primary safety concern in the evaporation process is the prevention of a violent explosive reaction of organic that is carried along with the aqueous stream because of inefficient phase separation in upstream mixer-settler bank. This reaction is known as a "red oil" explosion. Aqueous streams emerging from mixer banks or centrifugal contactor banks are piped to decanters to remove gross amounts of organic. Interlocks within the distributed control system prevent temperatures in evaporators in excess of 118°C, which are necessary for explosive reaction of entrained organic with nitric acid, by shutting off the steam to the reboiler. Additional temperature-related safety interlocks that shut down in the 1CU evaporator preventing the conditions conducive to a "red oil" explosion are as follows:

- High vent temperature,
- High reboiler steam pressure,
- Low liquid level, and
- High evaporator column differential pressure.

The cooling water supply valve to the condenser must be opened a specific percentage before the operation of the evaporator can be initiated. These safety interlocks are typical for all evaporators with a potential for receiving entrained organic in the feed.

Another safety concern is a leak of process solution into the failed reboiler contaminating the steam condensate or segregated cooling water supply. Process air is piped into the reboiler through a Cash regulator which pressurizes the reboiler with air to prevent leaks. A backpressure valve on the steam condensate discharge closes if reboiler pressure drops to 14 psi. If the pressure declines further to 12 psi, the reboiler is pressurized with air and the reboiler pressure maintained at 12 psi.

In the event of a large leak, one that exceeds the capacity of the Cash regulator to maintain the pressure, a quick-opening valve opens at 9 psi to maintain the coil pressure in excess of the hydraulic head on the reboiler. The quick opening valve has been installed on all the continuous evaporators.

In 1986, a data logger system will be installed which will access the data indicating leaks. Air flowing through a leak is detected by a monitor in the process air supply to the reboiler that reads-out in a data logger system and alarm. When an alarm occurs, the printer logs an alarm message giving date, time of day, point identification alarm setpoint(s), and the current value of the point.

The data logger also receives data from radiation monitors in the cooling water discharge lines. A leak from a reboiler or a coil that introduces radioactivity into the segregated cooling water can be pinpointed and the failure isolated before a large volume of cooling water becomes contaminated.

The uranium is extracted in the 1D bank and the waste, along with various other high level wastes, is collected in the high activity waste feed tank for evaporation and concentration of Pu and Np for recovery in the PRC. This mixed bed, three-stage ion exchange unit is batch operated and the instrumentation is appropriate to supporting a batch operation. Dip tube bubblers in the PRC feed tank indicate when the desired batch of waste with the requisite amount of Pu has been fed to the PRC.

3.2.3.4.5 Waste Treatment: Batch Evaporation

There are a number of batch evaporators used primarily in the treatment of waste solutions. A listing of these evaporators can be found in Section 3.2.2.7.1. Excess water is removed and acid stripped from the waste.

The control modes for these evaporators differ from the continuous evaporator because of the batch operation. The safety features in the control system are designed to prevent a red oil explosion and to prevent leaks of process solution into the steam condensate and cooling water discharge.

Batch level in the evaporator is a primary control parameter; and for those batch evaporators where acid stripping is a primary function, water is continuously added to the pot at a controlled rate and the operation terminated after a period of time sufficient for the removal of HNO_3 . In the other evaporators the contents are transferred after solids concentration, as determined by specific gravity, and boiling temperature have reached the desired level.

3.2.3.4.6 Radiation Monitoring

Health Monitoring (HM) ion chambers are located throughout the 200-Areas where a potential external radiation hazard exists. These HM chambers record and alarm in the 221-F Building dispatcher office. Personnel monitoring stations for the detection of alpha, beta, and gamma radiation are also located throughout the buildings and at exits from potentially contaminated areas.

3.2.3.4.7 Fire Detection

The fire detection instrumentation is discussed in Section 3.2.5.4.

3.2.3.4.8 Nuclear Incident Monitor System

Nuclear Incident Monitor (NIM) system activates alarms warning personnel to evacuate the vicinity because of a criticality incident. This system is provided wherever fissile materials are stored or processed in sufficient quantity for a potential critical configuration. Each nuclear incident alarm system associated with a potential incident site shall consist of at least two individual NIMs located no further than 100 ft from a potential incident location (15). A steady radiation rate between the limits of 0.5 and 3R/hr will sound the alarm within 5 seconds (15). An alarm shall also sound if the total dose received at the detector within one minute exceeds 50 mR. However,

for instruments in normally unattended, shielded areas, alarm rates shall be set well above the prevailing radiation background and may be as high as 100R/hr.

Each NIM is equipped with a green light to indicate that the NIM is operating, and with an amber light and audible signal to indicate a malfunction. Operating lights are located in the control room. During a power failure, each NIM remains capable of responding for at least 24 hours (15).

3.2.3.4.9 Data Logger System

A Data Logger System to be installed in 1986 will process data from: 1) the radiation monitors on the cooling water discharge from the canyon vessels; 2) the air flow monitor on the Cash system; and 3) differential pressure in the ventilation system and canyon exhaust air flow measurements from the warm canyon, hot canyon, and the new sand filter.

Operator interface to the system is through the CRT and keyboard. The CRT provides the displays required for data monitoring with displays called by dedicated pushbuttons on the keyboard. A status/menu display is provided for time of day, locations by page of acknowledged and unacknowledged alarms, communication status, and a menu of the dedicated function keys.

The computer tests data collected from the remote units, generates alarms as required, provides an alarm acknowledge routine, and provides an alarm log via the printer. Three general types of displays are provided for monitoring each of the different process systems: system status, system alarms, and data trends.

A high alarm setpoint is provided for each point, and an additional low alarm setpoint is provided for each cooling water flow. When an alarm condition occurs, an audible warning is sounded and the lower areas of the current display indicate an unacknowledged alarm. An alarm is acknowledged by calling the alarm or status display showing the point in alarm.

When an alarm occurs, the printer logs an alarm message giving date, time of day, point identification, alarm setpoint(s), and current value of the point. When an alarm is acknowledged, the printer logs an "acknowledge message" giving date, time of day, and point identification. When the point returns to normal, the printer logs a return message giving date, time of day, and point identification.

3.2.4 Electrical Power Distribution

3.2.4.1 Normal Power

Three-phase, 115-kVA, 60 Hz power, 60% generated in-plant and the rest purchased from South Carolina Electric and Gas Company, is routed through three interconnected substations. F-Canyon receives power from two of these substations (2). In F-Area, two 7500-kVA transformers reduce the line voltage to 13.8-kVA and feed the F-Area supply loop. The 221-F Building power is taken from the area supply loop through five secondary feeders. Two secondary

feeders are tied into the north branch of the area loop, designated Process Feeder No. 2; and three secondary feeders are tied into the south branch, Process Feeder No. 1. It is possible to connect between the two feeders in case either fails.

Fire protection of the first level motor control center and associated electrical equipment is provided by the manually operated sprinkler system. In the event of a fire, the electrical equipment is deactivated and the water flow to the sprinklers initiated by opening the hand valve.

Each secondary feeder is connected to a 750-kVA transformer in the first level electric control rooms where the voltage is reduced to 440 volts (except for the 1500-kVA transformer in F-Area which supplies the water chillers with a line voltage of 4160 volts). The middle leg on the secondary side of each transformer, as well as all electrical equipment in Building 221-F, is grounded through embedded liners to four buried ground rods in each building section.

Power from the secondary side of these transformers is distributed by a bus, through circuit breakers, to feeders that supply motor control center breakers.

All Building 221-F electrical equipment is supplied power from motor control center panels except for the instrument air compressors which receive power directly from the emergency bus. Some equipment uses remote-operated magnetic starters which are housed in the motor control center cubicles along with associated overload protective features. Actual starting of such equipment is performed using a pushbutton mounted adjacent to the equipment or provided in the control room. Whenever the overload feature operates to cut off power to the equipment, it is necessary to press the manual RESET button on the appropriate motor control center panel. When maintenance work is to be done on electrically-powered equipment, the motor control center switch is de-energized and locked out.

3.2.4.2 Emergency Power

Emergency power is available to critical equipment such as gang valves, lighting, air compressors, and ventilation equipment in Building 221-F. If normal power to Building 221-F is lost, the emergency system permits an orderly shutdown of the various processes so that the building and equipment can be maintained in a safe condition. A listing of the critical equipment on emergency power is included in Appendix A.

Generators. The main emergency power source for Building 221-F is a diesel generator located on the first level in Section . This diesel drives a 1000 kw generator to furnish 2000 amps. The diesel is automatically started by batteries and reaches full load in approximately 15 seconds in the event of a normal power outage.

Switchgear. Emergency power for Building 221-F is distributed by the emergency bus in electrical Control Room No. 1. This bus can be connected to two sources of power:

- Normal Building 221-F supply, and
- Building 221-F diesel generator.

The emergency bus in electrical Control Room No. 1 is part of the emergency switchgear. The switchgear with 1600 amp service is electrically interlocked so that no more than one power source is connected at any one time. Under normal conditions, power is supplied to Building 221-F through the 1500-kVA transformer in electrical Control Room No. 1. Failure of this source causes the Building 221-F diesel generator to start automatically, and connect to the emergency bus. Feeders from the Building 221-F emergency bus are connected through circuit breakers to various motor control panels. To permit maintenance work on the Building 221-F emergency bus, the motor control panels which receive power from electrical Control Room No. 1 can be supplied from alternate feeders.

Batteries. Emergency power from batteries is supplied for fire detection systems, diesel generator starters, lighting, and for certain doorlocks and communication equipment.

3.2.5 Auxiliary Systems and Support Facilities

3.2.5.1 Water Systems

Power Department supplies the 221-F Building with three separate service systems: cooling water, fire water, and domestic water including demineralized water.

3.2.5.1.1 Cooling Water

Supply. Cooling water from two wells is continuously pumped through heat exchangers in the waste tank farm and then delivered to 285-F cooling tower reservoir. About 400,000 gal of water is stored. Excess water received from the wells is discarded. One of the well pumps can be driven by a diesel motor if electrical power fails.

From the reservoir at Building 285-F, cooling water is pumped through the area by five pumps in parallel. An auxiliary steam-turbine-driven pump is also provided which automatically starts if pressure in the main cooling water header drops below 48 to 52 psig. These pumps discharge to a 36-in main header. This main header splits into a 30-in normal cooling water header and a 20-in independent cooling water header. Both headers have branches serving other facilities in F-Area.

If cooling water pressure fails (such as from electrical power failure, cooling water line break, etc.), certain key equipment is supplied by the independent cooling water system. This allows continued operation of

important equipment during a failure of normal cooling water supply of short duration, or provides for a controlled shutdown of key facilities if a prolonged cooling water failure is evident. There is an automatic valve in the 30-in header just downstream of the 20-in independent cooling water branch. This valve closes and diverts all available cooling water to the 20-in independent cooling water branch if pressure in the 36-in cooling water header drops below 45 psig and/or in event of low water level at the Building 285-F reservoir.

Return. Cooling water leaves the building in three mainstreams: clean cooling water, circulated cooling water, and segregated cooling water.

The clean cooling water return system has the largest bulk of the three streams. This water returns only from the non-regulated portions of the B-Line and Building 221-F first level; the refrigeration units associated with the water chillers contribute the largest single load. This stream returns through a header in first level where it exits from the building from the west wall. From this point, it runs underground to the 282-2F pump basin and returns to the Building 285-F reservoir.

The circulated cooling water return stream accumulates from canyon vessel coils in two underground headers, one on each side of the building, and flows through the 281-4F monitor, 281-1F delaying basin, and 281-2F pump basin, and then returns to the building 285-F reservoir.

The segregated cooling water system accumulates from canyon vessel coils which are supplied by both cooling water and steam. This return system is segregated from the circulated cooling water and the clean cooling water return systems because of a higher chance of failure of coils that are alternately heated and cooled. Segregated cooling water flows from warm canyon vessels to an underground header, west of Building 221-F, and ties into an underground header. The two underground segregated cooling water headers join south of Building 221-F, and the water flows underground to the 281-6F monitor. From here, the water flows through the 281-5F delaying basin and then is discarded to Four Mile Creek, if within discard limits.

3.2.5.1.2 Fire Water

Fire water is supplied to Building 221-F through three 14-in lines running parallel, north of the building, and connecting to a 24-in main area fire water header. Two of these lines go to the gang valve corridors, and reduce to 12-in lines just before they enter the building. The third line enters Section 18 in the center section, first level, and splits into two 10-in lines which go through the ceiling to second level.

On each of the 12-in lines entering the building in the gang valve corridors and on each of the 10-in lines going to second level, there are isolation valves that are electrically operated from the control room. On each of these valves, there is a STOP button which may be used to prevent operation of the valve while electrical or mechanical maintenance is in progress.

There are two strainers in parallel, one of which is a spare, on all four headers. Strainers are changed by manual valving, and screens are removed from their chambers for cleaning.

In each section of each canyon excluding Sections 17 and 18 for the F-Area hot canyon, there are three deluge valves, one for each set of spray nozzles. Deluge valves on the second level control flow of water to the cold-wall fire sprays. The northernmost deluge valve in the hot gang valve corridor and the southernmost deluge valve in the warm gang valve corridor, control flow to the fire sprays on the rack side of the canyon wall in each section. The other deluge valves in the gang valve corridors control flow to the rack pan fire sprays. Each deluge valve has a block valve upstream and a check valve downstream. Valves are opened by pushbuttons in the control room.

The gang valve corridor fire water header also supplies water for rack flushing. The second level warm canyon fire water header sprays in the south half of the warm canyon decontamination cell.

The fire water header serving the hot canyon from the second level extends to Section 3. At Section 4, a branch with a deluge valve supplies embedded piping and sprays in the south half of the hot canyon decontamination cell. At Section 3, the main header terminates at the point of a final branch, which supplies a deluge valve to the north 18 feet of the railroad tunnel.

The deluge valve for the railroad tunnel sprays is operated by a manual pushbutton in the control room. This is on the same panel as pushbuttons for the hot canyon sections. Deluge valves for the south half of the decontamination cells of both canyons are operated from control button switches located on the control room console. However, the deluge valves for the railroad tunnel are blanked on second level, and the deluge valves for the warm and hot decontamination cells are blanked on second level and in the gang valve corridors.

3.2.5.1.3 Domestic Water

Domestic water enters Building 221-F in a 6-in header through the west wall of Section 1, first level, near the point where cooling water and independent cooling water enter. There is a shut-off valve directly inside the building.

The main supply header runs the length of the building on first level and has several main branches serving bucket storage, the swimming pool, the warm gang valve corridor, hot gang valve corridor, warm canyon sample aisle, hot canyon sample aisle, second level, and B-Line.

Domestic water is provided to F-Area buildings for the usual domestic services. The service water system supplies water to 1) the fire water system and 2) the power house for boiler feed, miscellaneous cooling, and the production of deionized water by ion exchange.

3.2.5.2 Steam Supply

3.2.5.2.1 Generation

F-Canyon is normally supplied with steam from D-Area but steam can also be supplied from C-Area. A common steam header ties C-Area, F-Area, and H-Area together.

3.2.5.2.2 Distribution

The 325 psig steam is distributed through a 12-in header to reducing stations for the buildings. For Building 221-F, the reducing station is at the south end of the building where pressure control and indicating instruments are provided. From this station, steam at 150 and 15 psig is supplied. A steam flow recorder and a steam consumption integrator for the 325 psig steam supplied to the station are in the dispatcher's office. A pressure gauge and a low pressure alarm for the dissolver off-gas header are in the hot canyon control room. Pressure gauges and low pressure alarms for the 150 psig header are also in the canyon control rooms.

Dissolver Off-Gas Steam Supply. This steam is used only for heating the dissolver off-gas reactors in Sections 6 and 7 of the hot canyon. The 325 psig steam is reduced to 140 psig through a single pressure regulator valve into a 2-in header which enters the building and runs along the ceiling of second level to Section 7. There are cutoff valves at the reducing station and at Section 6 and 7 where lines to the reactor heaters go through the wall, and there is a pressure relief valve outside the building at the reducing station. Condensate from the off-gas reactors leaves the building through the segregated water header.

150 psig Steam. This steam is the main process steam source and is used for all vessel transfer jets, second level sump jets, etc. There are two stations for reducing the 325 psig supply to 150 psig. Normally, one of the stations is in use, the other on standby. Steam pressure can be controlled at 150 psig manually or automatically. Each reducing station is equipped with cutoff valves and a pressure relief valve. The 150 psig steam enters the building through a 14-in header which runs on second level to Section 5. The second level 150 psig header branches into a loop supplying steam to both the hot and warm canyons. The loops supplying the gang valve corridors branch off this second level loop. The FB-Line steam supply in F-Area ties into the 150 psig header at the main reducing station.

Condensate from the header traps and branch line traps drains to the cooling water return header; condensate from the canyon vessels is handled in the segregated water system; condensate from gang valves drains through seal pots to the exhaust air tunnel sump; and condensate from the sample aisle branch drains to Tank 181.

15 psig Steam. This steam is used primarily in the heating and ventilation equipment throughout Building 221-F. Its only use in the canyons is on the dissolver off-gas filters in Section 6 and 7 of the hot canyon.

15-15 psig steam is reduced from a 150 psig supply at two reducing stations outside the south end of Building 221-F. Normally, one station is in use, the other on standby. Steam can be controlled at 15 psig manually or automatically. Each reducing station is equipped with cutoff valves and a pressure relief valve. The main distribution header enters the building and runs along the ceiling of the second level to Section 2, then drops to first level and continues to Section 11.

Steam condensate from the various heaters is collected in a header which runs to a collection tank located in the Section 9 personnel tunnel. A pump actuated by a level switch on the tank automatically transfers accumulated water to the cooling water return header. Condensate can also be directed to the sanitary sewer outside Building 221-F.

3.2.5.3 Compressed Air

Four compressed air systems are maintained in the canyon building: instrument air, plant air, process air, and breathing air. Compressed air systems are the responsibility of the Power Department with the exception of breathing air which is jointly controlled with the Separations Department.

3.2.5.4 Fire Protection Systems

The Savannah River Plant has a Fire Protection Division which inspects and maintains fire fighting equipment, trains auxiliary fire brigades, and puts out fires. Every operating facility having Regulated Areas or Radiation Danger Zones maintains a separate or joint Fire Wardens Team for each shift. Each shift has an auxiliary fire brigade to fight all fires. The brigade consists of a chief, a hose squad, and a service squad. Fire fighting equipment is located in the F-Area fire station.

Resistance temperature devices (RTDs), and thermocouples are inserted into the air tunnel and into the pipe rack area of the hot and warm canyon by thermowells that originate in the Hot and Warm Gang Valve Corridor. The pipe rack is located above the Gang Valve Corridor and the air tunnel is located below the Gang Valve Corridor.

Actuation of these detectors will sound an alarm in the fourth level control room and the fire station in F-Area. Deluge valves (Figure 3-12) can be manually opened in the affected building section. Each deluge valve can be opened by pushing buttons in the control room. The 221-F Cold Feed Preparation area has a sprinkler system.

3.3 ENGINEERED SAFETY FEATURES

Engineered safety features associated with the canyon operations include the ventilation system, diversion of segregated or circulated cooling water, a secondary confinement.

FIGURE 3-12. Canyon Spray System - Typical Section

3.3.1 Ventilation Systems

Building 221-F is supplied by four ventilation systems: the canyon air system, the center section air system, the gang valve corridor air system, and the FB-Line system. Air supplied by these four systems is exhausted by five separate systems: canyon air exhaust, center section exhaust, B-Line exhaust, process vessel vent system, recycle vent system.

The ventilation system in Figure 3-13 is designed to prevent spread of airborne contamination to clean areas. This confinement is accomplished by maintaining positive static pressure in clean areas, atmospheric pressure in areas with low contamination potential, and slight vacuum in areas with high contamination potential. The control room is equipped with gauges that indicate the pressure differentials in the various areas.

3.3.1.1 Air Supply

3.3.1.1.1 Canyon Air Supply System

Air for the canyon air supply system enters a plenum chamber in Section 1 through fourth level ports in the south and east walls of Building 221-F. From this chamber, it is fed through a vertical shaft along the south wall of the building to the second level, where it is conveyed in separate ducts to the blowers for each canyon. Before entering a blower, the air is filtered through matted "Fiberglas" filters, heated by 15 psig steam coils, or cooled by chilled water pumped by a booster pump through separate coils, to maintain canyon temperatures between 70° and 105°F.

The four canyon air supply blowers are in Section 1 on the second level. Each blower is powered by a 40 hp motor, is rated at ~56,400 cfm at 425 rpm, and changes the canyon air twice an hour. There are two blowers for each canyon, one on standby that will start automatically. An alarm will sound in the control room in case of mechanical or electrical failure of an online blower. The alarm is reset in the second level fan room. Each of the blowers is equipped with an automatic damper in the discharge line. If a blower fails, the damper closes automatically, and the standby blower starts. Thus, the normal air flow into the canyon is continued.

None of these blowers is on the emergency power system. In case of a power failure to Building 221-F, all supply blowers stop and the dampers for all blowers automatically open. This allows the Building 292-F exhaust fans to maintain the normal air flow pattern through the canyons. In case of power failure to Building 292-F, all canyon supply blowers automatically shut down.

In case of loss of instrument air in Building 221-F, the canyon supply blower that is online continues to operate and all the dampers on all the supply blowers automatically open. Each blower is equipped with a manually operated damper which may be used to control the canyon air flow and maintain the desired air flow pattern. The supply blowers also shut down automatically if the vacuum created by the exhaust fans in Building 292-F drops to 1 s ha : 0.75-in water column.

FIGURE 3-13. Building 221-F Air Flow

Canyon air supply ducts extend from Section 2 through Section 18 on the third level (except for the portion of the warm canyon duct that passes under B-Line). These ducts are under the hot crane cab runway and the warm crane walkway. Inlet registers, three per section, pass air into the canyon area.

Branch streams from the canyon supply systems also feed fresh air to the fourth level crane maintenance vestibule, storage room, one workshop, the second level fan room, the swimming pool, both canyon shops, and crane maintenance areas. Positive air flow from these areas to the canyons is automatically maintained by motor-operated exhaust dampers. In addition, when the doors between the canyons and the crane maintenance areas are opened, electropneumatic operated dampers in the supply ducts to the crane maintenance areas raise the air flow from zero to 4700 cfm to the crane maintenance areas, and maintain 2200 cfm to the receiving room.

Within the canyons, air moves from the supply registers in the crane-ways to exhaust registers, three per section, located low on the rack side cell wall. These exhaust registers are equipped with manually adjusted dampers and lead to large exhaust air ducts, which run the length of the building directly under the gang valve corridors. The damper settings are made from the gang valve corridors. At Section 3, these ducts connect to the canyon air exhaust system tunnel.

3.3.1.1.2 Center Section Air Supply System

Building 221-F has two process sections, the hot canyon on the east side and the warm canyon on the west side, and the center section comprised of four levels between the canyons.

Fresh air ventilation for the center section is drawn in through a filter house at ground level on the west side of the building, and passes through a filter bank of 100 matted "Fiberglas" filters.

The air is picked up by a system of three booster fans that are also located in the filter house. Failure of any booster fan will activate an alarm in the control room. The failure also closes a pneumatically operated damper on the fan to maintain operating efficiency of the other fans. In case of total loss of instrument air, all dampers open so that air supply is maintained to building supply blowers; [REDACTED] If the booster fans fail and the dampers close, the supply fans would be starved.

Air supply from the booster fans is then routed via an external duct to the roof of Building 221-F, Section 9, through the Section 9 stairwell to the fan room on first level to Sections 10 and 11. A dual-purpose pneumatically-operated relief damper is installed in the external duct on the Building 221-F roof. This damper can be operated from controls in Section 9 on the fourth level to relieve air pressure created by the booster fans outside the building to allow use of the Section 9 stairwell for personnel movement in emergencies. The other purpose of this damper is for additional air supply to the building supply blowers in case of total booster fan failure or during filter changes. A pressure switch inside the plenum senses a negative pressure and causes the relief damper on the roof to open automatically.

The fan room on first level incorporates three supply blower systems (A, B, and C). Supply system A consists of two blowers, one of which is a standby. These blowers are equipped with automatic discharge dampers which close if the blower fails, but which open in the event of power or instrument air failure. These blowers are on emergency power. The standby blower starts and an alarm sounds in the dispatcher's office if the on-line blower fails. Failure of the center section supply blower automatically shuts down the following auxiliary fans: head tower, first level change room, regulated shop supply, regulated shop exhaust, regulated shop welding hood, and mask and tool decontamination exhaust. If the standby blower does not start, the second level supply fan, the canyon supply fans, and the fourth level B-Line supply fan will stop. These fans must be restarted manually. A time-delay relay interlock on supply system A blowers is provided so that routine changes or automatic operational checks will not stop the interlocked fans. The capacity of the on-line blower is 63,800 cfm.

Supply system B consists of one blower which supplies air to all of the first level sections except those locations supplied by system A blowers. This blower is not connected to the emergency power supply and does not incorporate a backup unit. Several separate and interconnected fans exhausting air are supplied by system B, and these exhaust fans are electrically interlocked to prevent their operation unless system B fan is operating. These include the second level supply fan, the first level change room exhaust, the regulated shop supply and exhaust, and the regulated shop welding exhaust. These fans must be manually restarted after normal power is restored.

Failure of the system B blower actuates an alarm in the control room. The capacity of the system B blower is 28,760 cfm. Supply system C consists of one blower which is also not on emergency power. There is no spare blower for this system. This system exhausts air from first level center Sections 8 through 18, and delivers cooled air to second level. Capacity of this system is 28,400 cfm.

Maintenance of appropriate differential pressures (Table 3-2) within the 221-F building is essential to protect against air reversals and subsequent spread of contamination. Typical conditions are as follows:

First level	- Should be maintained slightly positive with respect to atmosphere. The gauge and recorder range is 100% for +0.04-in WGSP (water gauge static pressure) with respect to third level. (No alarm).
Fourth level	- Identical to first level except that the differential pressure is maintained at 0.04-in WGSP. The sample aisle exhaust dampers are maintained in fixed positions approximately 75% open.
Second level	- Should maintain essentially atmospheric pressure.
Third level	- Should maintain slightly below atmospheric pressure. Approximately 0 to 0.01-in WGSP.

TABLE 3-2. Differential Air Pressure

<u>Between</u>		<u>Inches Water Gauge</u>	
High	Low	Range	Operating
Third level	Hot canyon	0.1 - 0.7	0.50
Third level	Warm canyon	0.1 - 0.7	0.45
Third level	HGVC	0.01 - 0.7	0.20
Third level	WGVC	0.01 - 0.18	0.06
First level	Third level	0.0 - 0.04	0.04
Fourth level	Third level	0.0 - 0.04	0.04

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Centimeter

Inches

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- Sample aisle - Should be negative with respect to third level.
- Electrical control rooms and compressor rooms - Should be slightly positive with respect to the cold feed preparation area.

Air from the center section circulates from the fourth level to the third level, and from the first level to the second level, then to the third level. From the third level, air flows through the sample aisles into the center section exhaust system tunnel.

Air from cold locker rooms, toilets, etc. (Zone 1 areas) is exhausted directly to the atmosphere by auxiliary fans through exhaust ports on the north and south ends of the building. Air from regulated change rooms and toilets is exhausted to the center section exhaust system.

Air is circulated to the mask and tool decontamination rooms and regulated shops in Sections 1 and 2 and exhausted by auxiliary fans to the center section system. Air is supplied directly from the outside at the south loading dock.

All B-Line air is exhausted by the B-Line exhaust system through the canyon air exhaust tunnel to the sand filter.

3.3.1.1.3 Gang Valve Corridor Air System

The gang valve corridors are supplied with 10,000 cfm of fresh air directly from the outside by air conditioners outside the building. Each 10 hp unit in the warm corridor moves 3330 cfm. Each 15 hp unit in the hot corridor moves 5000 cfm. The air exhausts through air-controlled dampers at Section 4 to the center section exhaust system tunnel. There are also three packaged air conditioners in each corridor that recirculate air for additional cooling. Blowers equipped with heating coils are installed inside the building but have not been needed because of the amount of heat given off by steam lines. Modifications are being made to this system to increase cooling capacity and to provide additional circulation.

3.3.1.2 Air Exhaust

3.3.1.2.1 Canyon Air Exhaust

Canyon exhaust air leaves Building 221-F through [REDACTED]. This [REDACTED] leads to the underground sand filters (Buildings 294-F and 294-1F). Filtered air is exhausted by fans in Building 292-F and discharged to the Building 291-F exhaust stack.

Sand Filter, 294-F. This filter, east of Buildings 221-F and 292-F, is a rectangular concrete structure, 240 x 100 x 16 ft. The [REDACTED] and supplies 12 distribution laterals that run under the filter to the south wall. Slotted precast concrete slabs placed over the

top of the narrow laterals provide a supporting base for channeled tubes which serve to evenly distribute air under the filter bed. The first layer of the filter bed is coarse stone; succeeding layers are of decreasing-size gravel and sand, making up to a filtering depth of 8-1/2 ft.

Three openings above the bed level on the south wall allow air to pass out into "valve" chambers which have removable concrete slabs at top and bottom. The top slab can be opened to the atmosphere; the bottom slab opens to a tunnel through which filtered air flows to the fan house (Building 292-F) and is discharged to the stack. When both slabs are in place, air flow is cut off; removal of only the bottom slab permits air to pass to the discharge tunnel.

If it becomes desirable or necessary to withdraw the sand filter from operation, the canyon exhaust air can be bypassed to the fan house through a tunnel immediately adjacent to the west wall of the filter. A "valve" chamber and slab arrangement permits air flow to be directed either to the filter or along the bypass.

There are two stainless-steel-lined sump pits that receive any internal drainage from the filter. One is located at the southwest corner of the filter, the other at the northeast corner. These sump pits are fed by gravity through stainless steel pipes, running from the low points of the filtered air tunnel, and from the east half of the canyon exhaust tunnel, respectively. When the sumps become filled, as indicated by local gauges, they can be jettied or pumped to Transfer Tank 805. The west half of the canyon exhaust tunnel drains to a sump just outside Building 221-F which has a high level alarm in the hot canyon control room.

The filter is equipped with ten sampling points each for radiation monitoring, pressure taps, and filtered air sampling.

In 1971, some of the internal supports in the 294-F sand filter collapsed. The collapsed areas were refilled, but the original sand and gravel that fell into the laterals caused a reduction in total air flow through the sand filter. The efficiency, however, is not significantly affected because the face velocity remained essentially constant.

Sand Filter, 294-1F. This filter, constructed in 1976, is similar to the 294-F filter that was constructed in 1976. The air [REDACTED] side of the filter and supplies 17 distribution laterals that run under the filter. The new filter has a stainless steel grid placed over the laterals to provide support for the filter media. The 294-F and 294-1F filters are operated in parallel.

Exhausters. Air is exhausted from the canyons by four fans located in Building 292-F. Each 200-hp fan is rated at 60,600 cfm. Three units are normally in operation and the fourth is on standby, to come on if the vacuum drops to less than 1.75-in water column. This also activates an alarm in the 221-F control room and an alarm in the Building 211-F control room. If the vacuum drops to less than 0.80-in water column, the canyon supply fans in Building 221-F are automatically shut off.

DELETED VERSION

Emergency power from the 600 kW generator in Building 292 is available to two of the canyon exhaust fans, but only one will operate under emergency conditions (the other will be on standby).

Exhaust Stack (Building 291-F). The 200 ft exhaust stack, located at the south side of the fan house, is constructed with an acid-resistant brick liner which rests on a stainless steel drip pan that drains condensate to the 1250-gal underground catch Tank 602. A high level alarm on this tank is located in the control room. The 100-gal catch Tank 601, with a high level alarm in the dispatcher's office, receives overflow from the stack drainage pan if the drain to Tank 602 becomes plugged.

The connecting chamber, or breech, from the fan house is constructed of 3/16-in stainless steel plate surrounded by a 3/4-in steel plate blast shield. The condensate from the breech drains to a sump pit which has a high level alarm in the control room. This alarm can also mean that Tank 602 has overflowed to its sump.

A small blower located just south of Building 291-F uses atmospheric air to maintain a slight positive pressure in the annulus between the acid-resistant brick and the outer stack structure. This air pressure helps prevent fumes, which might leak through the brickwork, from attacking the unprotected outer stack concrete.

3.3.1.2.2 Center Section Exhaust

Air from the center section of Building 221-F is exhausted through air-controlled dampers in the sample aisles ([REDACTED])

Three 40,000 cfm exhausters powered by 50 hp motors are located at Building 292-F and pull air from the center section ([REDACTED]). Two of these exhausters are normally in operation, with the third unit on standby. The standby unit automatically starts if the vacuum ([REDACTED]) drops to less than 0.75-in water column. When this occurs, an alarm also sounds in the dispatcher's office. Only one of these exhausters is on emergency power supplied by the Building 292-F emergency generator.

3.3.1.2.3 Process Vessel Vent System

Canyon vessels are maintained at a slight vacuum with respect to the canyons by two exhausters located at Building 292-1F. These 50 hp units are rated at 4800 cfm. Building 292-1F is below ground level and the fan rooms are Radiation Danger Zones. Canyon air flows into vessels through the overflow lines and then into the vessel vent header. Only the dissolvers and their associated off-gas reactors and filters are not connected to the process vessel vent system.

The canyon vessels connect to 18-in vent headers encased in concrete located along the outside walls of the respective air exhaust tunnels. The canyon waste headers and the gravity drain header are also connected to these vent

headers. At Section 5 of the warm canyon and at Section 7 of the hot canyon, the vent headers enter the canyons where the air is filtered to remove radioactive particulate matter. After the air is filtered, it returns to vent headers located in the canyon air exhaust tunnels. The vent headers form a common header in the canyon air exhaust tunnel.

The common vent header leads to the exhausters at Building 292-1F. Air from the fans is exhausted to the canyon air exhaust tunnel before the sand filters. Both exhausters are on the Building 292-F emergency power system. Normally only one exhauster is in operation; the second is held as a standby.

The exhausters have air-operated valves in both the discharge and suction sides; these valves open when the fan is operating and close when the fan is shut down. A mechanical stop prevents the discharge valves from opening wide. This stop has been set to maintain the static vacuum in the vent header necessary to meet design flow requirements. The position of these stops is changed only if filter pluggage or some mechanical difficulty causes loss of vacuum in the vent header.

The vacuum on the suction side of these exhausters is measured by a water manometer in 292-1F. Air flow on the discharge side of the exhausters can be measured with a Pitot tube located in the discharge line.

The process vessel vent exhausters normally maintain a vacuum at the fan of about 25 in. If the vacuum drops to less than 16 in, the standby exhauster starts, and alarms sound at the control room and in the Building 211-F control room. If the inlet filter pressure at 5.7 or 7.2 becomes too high, an alarm sounds. Differential pressure instrumentation is located at fourth level control panels.

In the Building 221-F gang valve corridors, pressure taps extend through the floor and connect to the vent header at Section 5 (warm canyon) and Section 7 (hot canyon). They are used for sampling and for measuring vacuum in the vessel vent header.

3.3.1.2.4 Recycle Vent System

The recycle vent system services vessels in four main areas: A-Line, Building 211-F area, third level of Building 221-F, and the first level waste tank area of Building 221-F. All vessels and tanks have trapped overflows to prevent air from being pulled in through them.

All third level tanks except Tank 6B vent to a 6-in stainless steel header that runs from Section 5 through Section 17 on the third level. This system operates at a header vacuum of 5.7-in water column. First level waste Tanks 181 and 182 and Sump 183 are vented through an 8-in header that tie into the header from the third level at Section 4 on the second level.

The vent header from A-Line and the Building 211-F area joins the Building 221-F header in the air tunnel. The air discharged from the recycle vent is heated by a steam heater to 150°F, then filtered through two "Fiberglas" filters and exhausted to the Building 291-F stack.

The vacuum in the individual area headers is automatically controlled by instruments operating butterfly dampers that bleed air into the header. There is a draft gauge on the second level of Building 221-F.

3.3.2 Diversion of Segregated or Circulated Cooling Water

If radioactive contamination gets into either the segregated or the circulated cooling water return systems, these streams can be diverted to either a seepage basin or to retention basin 281-8F (16). For each stream there is a monitor station that continuously checks for alpha and beta-gamma activity.

Segregated cooling water normally passes through a monitor (281-6F) to delaying basin 281-5F. This basin has a capacity of 400,000 gal. It empties to Four Mile Creek, if the activity is within limits. About a 5 hour delay is provided by this basin to allow for confirmation of monitor results before releasing or diverting the segregated cooling water.

If either the circulated or segregated system is diverted, retention basin 281-8F is designed to hold all water that may be diverted. However, it is possible that if the circulated cooling water return system diverts for 30 min, a low water level will be reached at the 285-F cooling tower reservoir. In case of low level at 285-F, an automatic valve in the normal cooling water supply header closes, leaving only the independent cooling water supply headers throughout the area pressurized.

Segregated water is monitored in 281-6F. If no contamination is being detected at the water monitor, the valve settings and alarms may be returned to normal by depressing the reset button.

Circulated cooling water is monitored at Building 281-4F. The volume of this stream is quite large, and if it were diverted to the retention basin in error or because of a false instrument alarm, loss of water from the cooling water system would be substantial. If it were necessary for this stream to be routed to the retention basin, the contents of the water system would be depleted in less than 30 min under normal demand, even though supply from the wells continued. If circulated cooling water is diverted, operations are immediately shut down to avoid pumping contaminated water to the cooling tower.

When an alarm is received in the control room, the supervisor and Health Protection representative at Building 211-F are contacted immediately. The alarm is promptly investigated at 281-4F by the Building 211-F operator. Presence of contamination in the return stream is either confirmed or the source of trouble is located. An instrument records the level of contamination. If it is necessary to divert the stream to the retention basin, diversion valves may be operated manually in the Building 221-F control room, 281-4F monitor building, or at the 281-1F valve pit. A few seconds later, red lights illuminate, showing that diversion valves are in operation. When full movement of the valves has taken place, green lights go out and red lights remain illuminated.

Retention basin 281-8F is a lined excavation which receives aqueous waste streams for temporary storage, and which has a capacity of 4 million gal.

This is large enough to store the total volume of water from both the canyon circulated and segregated cooling water systems, as well as a volume of flush water from radioactivity cleanup. Disposition of water in the retention basin is to the seepage basin, creek, or waste tank.

3.3.3 Secondary Confinement

The process vessels are located in cells, four to a section. Concrete or stainless steel fire walls are used to separate the following hot canyon sections: 7 and 8; 8 and 9; 9 and 10; 10 and 11; 11 and 12; 15 and 16 and 16 and 17. In the warm canyon, fire curtains are located between sections 6 and 7, and between all succeeding sections through section 15. Sections 15 and 16 are not separated by a fire curtain, but fire curtains do separate 16 and 17, and 17 and 18. The fire curtains separate those sections with organic from the rest of the process. As a fire protection measure, the top of each cell is covered by concrete cell covers, 1-ft thick. Three sections of interlocked cell covers are placed on top of each cell. The primary function of the cell covers is to control air distribution among the cells. Cells are separated by 2-ft high curbing which contains the expansion joints.

The concrete cell covers and the walls of the canyon were designed to be radiation shields as shown in the cross section of the canyon (Figure 3-1). The shield thickness and closest personnel access corridors are given in Table 3-3.

Secondary confinement for airborne materials is defined as the ventilation exhaust system that is described in the F-Canyon Systems Analysis (2).

The canyon building was built to contain releases of radioactive liquids. The floor of the process cells in the canyon is sloped toward collection sumps. Drips or liquid spills are flushed to the sumps and pumped to appropriate catch vessels in rerun.

The canyon sumps (2 ft x 2 ft x 1-1/2 ft) are equipped with liquid-level detectors and transfer jets. A minimum liquid level is maintained in the sumps to verify that the detector is working properly. A high-level alarm sounds if the liquid level reaches the level of the cell floor, minimizing the possibility of overflowing sumps. Process liquids on the cell floor may cause corrosion of the floor, and buildup of uranium or plutonium.

Sump solutions were transferred through a waste header on the outside of the building to a tank vault. In 1986, this header will be re-routed back to an F-Canyon tank. Restrictions based on criticality safety will be relaxed because of improved sampling and agitation provided by the canyon tank.

3.4 DECOMMISSIONING CONSIDERATIONS

A decommissioning and decontamination plan will be formulated and used prior to shutdown of these facilities. Design features of 200 Area facilities that affect decommissioning are briefly discussed in the following paragraphs.

TABLE 3-3. Canyon Vessel Radiation Shields

Area	Concrete Shield Thickness, ft	Closest Personnel Access
<u>Hot Canyon</u>		
Outer Wall	4	Gang Valve Corridor*
Inner Wall	5-1/2	All Central Section
Roof	3-1/2	Crane* (40 ft above cell covers and offset)
	4-3/4 (minimum)	Sample Aisle,* Cold Feed Gallery* (20 ft above cell covers and offset)
<u>Warm Canyon</u>		
Outer Wall	2	Gang Valve Corridor*
Inner Wall	4-1/4	All Central Section
Roof	2-1/2	Crane* (40 ft above cell covers)
	3-1/2 (minimum)	Sample Aisle,* Cold Feed Gallery* (20 ft above cell covers and offset)

*Limited and Controlled Access.

3.4.1 Remote Operation

The ability to remove canyon process equipment remotely and place it into a decontamination area is available for decommissioning. This is particularly desirable for equipment that is contaminated with materials which emit highly penetrating radiation. High activity waste evaporators have been decontaminated and repaired in this manner.

3.4.2 Partitioning of Process Functions

The partitioning of process functions allows highly contaminated areas to have little effect on low-level contamination areas.

3.4.3 Protective Coatings and Liners

Stainless steel liners for some canyon sumps, and special protective coatings for materials such as concrete are included to simplify decontamination procedures. Some sections of the hot canyon in F-Area were decontaminated in 1970-1971 to permit personnel entry for the construction of the Multi-Purpose Processing Facility.

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4.0 DESCRIPTION OF OPERATIONS

The primary administrative control document is the contract (DE-AC09-76SR00001) between Du Pont and the U.S. Department of Energy (DOE). The contract explicitly describes certain obligations with regard to safety on the part of both the contractor (Du Pont) and the contracting officer (DOE). While Savannah River process facilities are operated by Du Pont, DOE is ultimately responsible for conduct of the program under the law, as expressed by regulatory requirements of various governmental agencies.

4.1 ORGANIZATIONAL STRUCTURE

The Savannah River Plant is operated by the Atomic Energy Division (AED) of the Petrochemicals Department of E. I. du Pont de Nemours and Company (1). The structure of the AED, for purposes of this SAR is shown in Figure 4-1. The two major divisions are the Technical Division, which includes the Savannah River Laboratory (SRL) and the Manufacturing Division, which includes the Savannah River Plant (SRP). The Departmental Engineer's Office reports separately to the AED in Wilmington.

The SRL provides technical support to the Manufacturing Division. The SRP organization has two major divisions; Operations, and Plant Facilities and Services. The Manager of Operations has custodial and operating responsibilities for all production facilities. The Manager of Plant Facilities and Services has responsibilities for nonproduction facilities and central services.

Primary responsibility for safe operation of the 200-Area facilities lies with the Separations Department of the Separations Program Management Team (PMT), with major support from the other PMT departments: Separations Works Engineering and Separations Technology. These three departments are supported by other departments such as Projects, Equipment Engineering, Health Protection and Personnel. The SRP organization is shown on Figure 4-2.

4.2 OPERATIONAL DESCRIPTION

Nuclear fuel elements, previously irradiated in reactors, are processed in the F-Canyon facility. Primary operations include chemical and physical separation and purification of materials. Process operations are diversified, encompassing three main products.

Objectives of the operations include maintaining and ensuring personnel and equipment safety and efficient operations. Basic and important operating decisions are made by management (Section 4.1) after review throughout the organization. Decisions that affect safety or operability are reviewed by a competent technical organization other than the one with direct responsibility for operation. Process operations are performed according to approved detailed written procedures (Section 4.3).

FIGURE 4-1. Atomic Energy Division Management

FIGURE 4-2. Organization of Savannah River Plant

4.3 PROCEDURES

4.3.1 Administrative Procedures and Control

A formal administrative control system ensures that basic and important decisions affecting safety or operability are adequately reviewed (2). The system requires the review by a competent technical organization other than the one charged with direct responsibility for conducting and controlling the operation of the facility.

The Separations Department operates the 200-Area facilities (Figure 4-2). Qualified operators who have received rigorous training in operating procedures and safety systems carry out the separation processes according to detailed written procedures under the direction of Separations Department. The Separations Technology Department provides continual independent input into program planning and implementation and a continual review of operations. Separations Technology must approve all operating procedures, process changes, and process equipment changes. Other departments such as Health Protection and Long Range Planning provide advisory reviews. The Technical Division provides an independent review of the 200-Area operations.

In addition to its independent reviews, the Technical Division provides basic technical information for process and equipment design and operation. The Technical Division prepares Technical Manuals, Systems Analyses, Technical Standards, and Safety Analysis Reports. This division also reviews and/or approves Test Authorizations and maintains liaison on technical matters with other interested parties such as DOE.

4.3.1.1 Notification and Reporting of Occurrences to DOE

A system for notification, investigation, and reporting of occurrences to DOE-SR is maintained as described in the SRP Procedures Manual Item 201, Notification and Reporting of Occurrences. An occurrence is defined as any deviation from the planned or expected behavior or course of events if the deviation has safety, health, cost, or environmental significance. Occurrences are classified according to seriousness, and time limits and responsibility for notification for investigation, are as follows:

- **Type A:** Immediate verbal notification; written followup within 20 hours; investigated by DOE-SR.
- **Type B:** Verbal notification within 24 hours; written followup within 72 hours; investigated by DOE-SR, but may be delegated to Du Pont personnel.
- **Type C:** Verbal notification as soon as practicable, but no later than the workday following the end of the quarter; written information no later than the 8th calendar day following the end of the quarter; investigated by plant personnel.

- **Unusual Occurrence:** Occurrences that have significant adverse programmatic effect and involve information that can be used at other DOE sites to avoid similar occurrences are reported into this DOE-wide system. The coordination of the release of SRP Unusual Occurrence Reports to DOE-SR and the review of Unusual Occurrence Reports from other sites will be by the Manager of Operations or his delegate.

4.3.1.2 Special Incident Reports

Special Incident Reports are formal reports that are prepared when incidents are of major proportions or are likely to be of interest to outside groups. They are in addition to and entirely apart from the Notification and Reporting of Occurrences and the usual "Incident" report prepared by direct supervision. The decision to prepare a Special Incident Report is made by the Plant Manager or Wilmington management.

4.3.1.3 Process Incident Investigations and Reports

Reports of process incidents are made using a system that provides a mechanism for prompt reporting of all incidents and assures that the thoroughness of the investigation, the followup, and the distribution given the report are commensurate with the seriousness of the incident. An incident is defined as any event that is a deviation from the accepted normal operation and appears to have significant implications relative to safety, equipment performance, or process operations. These reports are made as described below:

- **Separations Incident Reports.** A Separations incident report is used to describe events that constitute deviations from the accepted normal operation of the process and associated equipment. This report is intended to call the Separations Department's attention to incidents with significant potential and, through a study of compilations of these reports, to establish trends in the operation that might not be obvious over a short period. Separations incident reports are prepared by the Separations Technology Department and reviewed by the Separations Department.
- **Operating Incident Reports.** After an incident occurs that appears to have important significance to safety or operation of the facility, the Separations Department Area Superintendent convenes a committee to investigate the incident and issue a report. A Separations Department supervisor acts as chairman, and Separations Technology participates on the committee. Other groups that are involved in the incident, or that may be concerned with the findings, also participate. This committee attempts to determine the cause of the incident and to develop recommendations to prevent recurrence. The committee issues a report called an Operating Incident Report.
- **Special Hazards Investigations.** Special hazards investigations are made for unusual incidents involving actual or potential

radiation exposure, radioactive contamination, significant loss of confinement of radioactive material or if criticality control limits in Technical Standards, TAs, or OSRs have been exceeded. Jurisdiction over the investigations is assigned to the Plant Central Safety Committee. The scope is detailed in the SRP Safety Manual (3) and in radiation and contamination control procedures (4).

- **Unusual Incident Reports.** Unusual Incident reports (UIs) are prepared to describe situations involving high injury potential. The involved employee's Department Superintendent initiates the investigation. Where personnel are not directly involved, the Superintendent of the department owning the equipment concerned initiates the investigation. The scope is detailed in the SRP Safety Manual (3).
- **Daily Reports.** The Operations-Program Management Team (PMT) and Plant Facilities and Services organizations issue daily reports to Wilmington, with copies to the Savannah River Laboratory and the Savannah River Office of DOE. These reports include information about a significant incident shortly after it has happened.
- **Monthly Reports.** Activities of all Operations-PMT organizations and Plant Facilities and Services organizations are summarized in monthly reports. Individual program reports for Separations, Raw Materials and Tritium Facilities, and Waste Management describe technical studies of the nuclear safety and other technical aspects of facility operation, failure to meet a technical standard and/or an operations safety requirement, the status and efficiency of production, and design activities. Separate reports are issued to document the status of the Quality Assurance Program and Environmental Control.

The Technical Division, Savannah River Laboratory, issues a monthly progress report of significant results from studies relating to nuclear safety, process improvements, and new processes and products.

- **Annual Reports.** An annual report of the results of the auditing and incident investigation system for the separations areas is made by the Separations Department and the Separations Technology Department. This report includes statistical information on the audit results and on the number and kinds of unusual incidents. It also includes short discussions of the investigation and action taken on incidents that appear to have the most safety significance.

4.3.2 Operating and Maintenance Procedures

Du Pont imposes an internally authorized system of control procedures to ensure that facilities are operated and maintained in conformance with Du Pont

management policies as prescribed in the administrative control procedures for nonreactor nuclear facilities. The system has the familiar Du Pont operating objectives:

- Maintain safety of personnel, equipment, and facilities.
- Maintain continuity and increase efficiency of operations.
- Maintain compliance with applicable governmental regulatory requirements to ensure public health and safety and protect the environment.

Inherent in the controls system is the precept that all process operations are performed according to approved written procedures that have been reviewed in an effort to preclude unsafe consequences, either directly or as a result of a possible chain of unfavorable events. Audits of procedures are performed and documented, and periodic reassessments of facilities and processes are conducted with the objective of identifying any previously unrecognized hazards that may have been created by changes in process conditions, operating practices and equipment, or knowledge of the condition and behavior of construction materials.

4.3.2.1 Operational Control Documents

The following is a brief discussion of Du Pont Operational Control Documents. A more detailed description of each document can be found in Reference 2.

The level of control of each of these documents is shown in Figure 4-3.

Systems Analyses. Systems Analyses are comprehensive reviews of the equipment and processes of functional systems. The objective of these reports is to determine the risk of operating each facility. The reports provide technical bases for determining the interaction of the various systems as presented in the Safety Analysis Report. Systems analyses are prepared by the Technical Division, and the reports are reviewed and approved by both the Manufacturing Division and the Technical Division of AED.

Safety Analysis Reports. A Safety Analysis Report (SAR) is a document that describes the facilities and processes analyzed, and evaluates the risk of operation. Included in the risk is the probability of the occurrence of accident sequences and consequences. Both onsite and offsite risks are included. The SAR is prepared by the Technical Division and is reviewed by the Technical and Manufacturing Divisions prior to significant facility modifications and is updated within a five-year period. New issues and revisions to SARs are approved by Du Pont AED and DOE at Savannah River. Approval is also required by the Separations Program Management Team.

Operational Safety Requirements. Operational Safety Requirements (OSRs) define the envelope of authorized operations of the nonreactor nuclear facilities and formally document the requirements in the following categories:

FIGURE 4-3. Document Requirements and Procedures Flow Chart

- Safety Limits and Limiting Control Settings
- Limiting Conditions for Operation
- Surveillance Requirements
- Design Features
- Administrative Controls

OSRs and revision proposals are prepared by consultation of AED and DOE-SR staffs.

Technical Standards. Technical Standards define the process limits within which the facilities are operated, ensuring the safety of personnel, permanently installed equipment, and the environment. They specify the requirements and bases for basic variables within which the process must be operated for reasons of safety, quality, and/or limitations of known technology. These requirements are within the boundaries of safe conditions reported in the OSRs.

Technical Standards are authorized by the Director of the Technical Division after approval by the Manufacturing Division.

Mechanical Standards. Specifications for the physical installation of buildings process and service lines and equipment are termed Mechanical Standards. These standards are maintained in several forms as follows:

- **Du Pont, Subcontractor, and Vendor Drawings.** New issues or changes are approved by Separations Works Engineering, Separations Department, Separations Technology Department, and the Departmental Engineer's Office in Wilmington if the process is involved.
- **Du Pont Engineering Standards and Specifications, and SRP Engineering Standards and Specifications.** The Du Pont Engineering Department issues the Du Pont Engineering Standards and Specifications and keeps them current. The SRP Engineering Standards and Specifications are written for local use and provide uniform requirements for design and specifications. Additions and revisions are authorized by Separations Works Engineering or Technology and approved by the program manager.
- **SRP Project Specifications.** These specifications for the purpose of material procurement may include requirements for vendors and for onsite installations. New specifications and supplements are authorized by the program manager and issued by the Du Pont Engineering Department.

Technical Manuals. Technical Manuals are issued by the Technical Division. They may be prepared by either SRL or Separations Works Engineering. These manuals contain the basic technical information used for the design of facilities and for the preparation of Technical Standards. Technical Manuals are approved by both Technical and Manufacturing Divisions.

Design Reports. There are three different types of design reports: Technical Data Summaries, which are prepared and issued by SRL for planning projects that evolve from results of research and process development activities; Basic Data Reports, which are prepared and issued by SRP for planning projects for additions or modifications to plant processes or facilities; and Design Data Reports, which are issued by the Engineering Department for final project design.

Test Authorizations. A Test Authorization (TA) authorizes temporary deviations from Technical Standards, Operating Procedures, or Mechanical Standards. The TA is used to conduct process study trials with plant equipment, or to authorize nonstandard operations not covered by Technical Standards or Mechanical Standards. Limits defined by the TA will be at or within the boundaries of safe conditions specified in the OSRs. TAs are normally prepared by the Separations Technology Department. They are authorized by the Manufacturing Division Director after approval by the Technical Division; an exception is allowed when departures from Operating Procedures are within Mechanical Standards and Technical Standards. The TA may then be authorized by the Plant Manager after approval by the Separations PMT (also by Nuclear Engineering and Materials Section of SRL when nuclear safety is involved).

Nuclear Criticality Safety Control. Nuclear criticality safety controls are administratively achieved through:

- Systems Analysis Reports
- Safety Analysis Reports
- Operational Safety Requirements
- Technical Standards
- Nuclear Criticality Safety Supplements
- Operating Procedures

Plant Interpretive Documents. These are interdepartmental memoranda which document such things as recommendations made by technology groups to operating groups regarding changes in processes or procedures, agreements on matters of departmental interactions, program planning and scheduling, transmittal of information developed during inplant tests or studies, and actions necessary to implement any special programs. These memoranda usually are prepared by

staff personnel and approved by the superintendent of the originating department, or his delegate.

Quality Assurance Assessment Reports. QA procedures require that QA assessments be performed for existing facilities and processes, new designs, and modifications to existing designs to define the application of a formal QA program proportional to need. The Savannah River Plant Quality Assurance Program is discussed in detail in Section 7.0 of this report.

4.3.2.2 Operating Procedures

All process operations and many allied activities such as maintenance operations are performed according to written procedures (5). A procedure supplies detailed how-to-do-it directions (usually in a stepwise manner) for a given process or operation. The intent is to ensure systematic control of safety, quality, and yield. It is a removable-page document, and routine practices are established for its continual revision so that directions can be kept up to date and obsolete copies are not available for inadvertent use. An explanation of these procedures is given below:

Du Pont Savannah Operating Procedure (DPSOP). A DPSOP is a book divided into chapters (I, II, etc.), then into divisions (A, B, etc.), and within these into sections (1, 2, etc.). A DPSOP may contain one or all three of the following types of unit-pages.

- A text page contains descriptive information and/or stepwise directions.
- A figure page consists of a chart, drawing, process diagram, or photograph.
- A DPSOL (Du Pont Savannah Operating Log) page is primarily meant for separate on-the-job use as a logsheet for recording data, time, and checkoff or initials to show job progress. Each DPSOL (sometimes multipage) is individually page-numbered and has its own sequence of revision numbers. There are two categories of DPSOLs: "Training and Reference" (T&R) and "Use Every Time" (UET). The T&R serves to standardize instructions and aid in instructing a person new to a job.

Du Pont Savannah Operating Log (DPSOL). DPSOLs are written or revised by a Separations Department area supervisor with technical assistance from the Separations Technology Department. New DPSOLs are usually routed for full approval to Separations Technology and the Separations Department; however, Separations Department supervision may select a lower approval level for revisions that affect only the methods of doing the work and not basic variables. These may be approved by the Separations Department Area Supervisors and the Separations Technology Department Area Supervisors.

Plant Manuals. Apart from DPSCPs, there are several unnumbered manuals that contain administrative procedures concerned primarily with plantwide policies applying to personnel. Three examples are: the SRP Procedures Manual, which contains procedures that involve intergroup rather than intragroup dealings, office letters issued by the Plant Manager, and announcements of long-term interest; the Security and Safeguards Manual, which defines minimum security requirements for the plant based on DOE regulations; and the SRP Safety Manual, which defines basic plantwide safety policies and minimum requirements. These and other manuals are described in Reference 5.

4.3.3 Emergency Procedures and Plans

The SRP staff has the organizational responsibility for determining emergency plans and courses of action for each plan. Periodically, the staff reviews the performance of the plant in emergency plan practice drills. The effectiveness of the plans is also reviewed periodically, and revisions are made as necessary. The policy of SRP is to limit the radiation dose to workers to 25 rems (whole body), and 100 rems (hands) in an emergency which involves protection of property or personnel. For life-saving acts, an exposure of 100 rems (whole body) plus 200 rems (hands) is considered as the guide value.

Both F- and H-Areas have an emergency committee, with the Area Production Superintendent as chairman. Other members include representatives from each department resident in the area. This group resolves problems of area-wide significance, adapts the emergency plans to their particular location, and initiates and conducts practice drills.

SRP has an Emergency Operating Center (EOC) located in Building 703-A (located in A-Area). The EOC is equipped with emergency materials such as radio equipment, telephones, maps, and plotting boards. Food and sleeping gear are also provided.

There are three basic plans discussed in this report. They are:

1. Shelter or Evacuation Plans
2. Nuclear Incident Plans
3. Offsite Warning Plans

Shelter or Evacuation Plans (6). There are three types of emergencies considered which involve sheltering or evacuating personnel; all are practiced. They are:

1. Facility Emergency (where an emergency may exist in only a single building).
2. Area Emergency (where all buildings in an area are involved).
3. Plant Emergency (where a local condition, or a condition created offsite, affects the plantsite).

A Separations Department supervisor is the emergency Facility Coordinator (FC). Based upon his evaluation of any given situation, the FC determines whether evacuation or sheltering of personnel in the facility is or is not necessary. He announces that the facility emergency exists, provides instructions on the handling of personnel and facilities, and notifies the Area Emergency Coordinator. The FC has an announcement made over the area public address (PA) system. The announcement is made at two-minute intervals for a total of four times (7). Evacuating personnel assemble at the appropriate location as instructed by the Facility Coordinator.

An Area Emergency is declared by the Area Emergency Coordinator, who notifies the plant EOC. He announces the emergency over the area-wide public address system and sounds the alarm signal (normally a fast warble, but also could be five short blasts from the powerhouse whistle). Area emergencies are handled in the same fashion as Facility Emergencies. All personnel in the area are evacuated or sheltered, with the exception of a small predesignated crew. If personnel are sheltered, the Area Emergency Coordinator may adjust building ventilation as necessary and will evaluate the need for an area evacuation. Evacuating personnel will form a caravan at the parking lot and proceed to a safe location. The normal evacuation route in the F-Area is to the main parking lot, through the main gate. The Area Emergency Coordinator may designate an alternate route.

A plant emergency involving total plant evacuation or sheltering of facility personnel is initiated by the Plant Manager and conducted in the same fashion as Area Emergencies. Personnel, on the average, can be evacuated or sheltered in less than 15 min from the time the evacuation is announced. Assuming all areas are within 15 min of the plant boundary, it is estimated that the entire plant could be evacuated in about 30 min.

Nuclear Incident. When the Nuclear Incident Monitor (NIM) bell alarms, all personnel hearing the alarm evacuate immediately by predesignated, well marked routes and gather at a predesignated rallying point. The alarms are connected to instruments in the control rooms of the production facilities. When the NIM alarm bell rings in the affected area, a light is lit on the annunciator panel, and an alarm sounds in the control room. Verification of the alarm is made from the control room. If verified, or if two or more NIM instruments alarm, or if confirmation is received on other radiation monitoring instruments, the incident is considered real. The facility is shut down and personnel evacuated or sheltered, and the Facility Coordinator declares a Facility Emergency and notifies the Area Coordinator. If other buildings or areas may be affected, the Area Emergency Coordinator declares an Area Emergency. He may adjust building ventilation as necessary, and evaluates the need for an area evacuation. He also initiates action to locate persons not accounted for. When all personnel have been monitored, the Facility Coordinator directs Health Protection personnel to survey the area for re-entry.

All personnel are trained to recognize and use the proper escape routes in their individual facilities. These routes are well marked with Nuclear Incident Evacuation signs. Practice drills are held periodically to keep personnel familiar with the escape routes.

Offsite Warning Plan. The Offsite Warning Plan sets forth the conditions for alerting offsite areas of a hazardous condition originating at SRP that may affect them. Following an incident of sufficient severity to threaten offsite areas, the EOC Supervisor may elect to dispatch monitor teams, each consisting of a Health Protection Inspector and driver-radio operator, to predesignated monitor points downwind of the incident. They attempt from ground level to evaluate the spread of radioactive contamination, and radio the results back to the EOC.

In the EOC, when the source of the release, wind direction velocity, and concentration are known, the results are extrapolated to estimate the potential exposure to offsite areas. The information is given to the EOC staff members who make the decision to warn the offsite population center. If EOC staff members are not available, the EOC Supervisor may issue the warning.

It is estimated that it would take up to 2 hours to determine the extent of the accident sufficiently to decide whether or not an offplant warning should be made.

4.4 TRAINING

The type of equipment and the nature of the work performed at SRP are unusual and subject to requirements far beyond those of more conventional industries. Of primary importance are requirements regarding the qualifications and training of the individuals performing radiation work at SRP.

Personnel receive training in the safety aspects of new jobs with periodic retraining in certain areas (e.g., chemical properties, self monitoring, etc.), and utility scheduled safety meetings to implement plantwide programs. Personnel also receive training in emergency actions through scheduled drills and practices under simulated emergency conditions. Training records are kept for all employees.

New employees receive an orientation series on the following subjects:

- Safety rules and requirements
- Security rules and requirements
- Industrial relations, plans, and benefits
- Automobile traffic rules and regulations on plant property
- Equal employment opportunity policy

Separations and Separations Works Engineering Departments. The training program for Separations Department personnel ensures that trained personnel operate and maintain the equipment and facilities. The objectives of the program are to:

- Provide qualified personnel for each operating facility.
- Improve productivity.
- Expand employee knowledge of the processes and equipment contained in each of the facilities.
- Enhance the safety and security of all personnel.
- Heighten individual responsibility awareness.
- Increase employee flexibility within an assigned facility.
- Provide a formalized, auditable program that meets the DOE requirements.

These objectives are achieved through a comprehensive job-performance-based training program.

Program Structure. The Separations Department Superintendent has overall responsibility for the training program. A coordinator of Separations operator training, a coordinator of Separations Works Engineering mechanic training and a coordinator of Separations exempt salary training are responsible for the development, revision, and delivery of training in their respective areas.

An instructional systems specialist and an educational specialist provide support to the coordinators, and the training specialists. The instructional systems specialist is responsible for audio-visual training aids, development and maintenance of trainee records, and computer assisted instruction implementation.

The education specialist is responsible for assisting training specialists and coordinators develop lesson plans, write examination items, and validate materials. He is also responsible for providing instructor training, developing new courses, and selecting the best delivery media.

Qualifications. An individual becomes qualified when he or she can demonstrate adequate knowledge and skills in the areas to which he or she is assigned. Each knowledge and skills area, as identified by job/task analyses, is a major subdivision of the knowledge and skills required by the employee to adequately perform a job assignment.

Knowledge and skills areas have been identified for each job classification within the Separations Department. Within each job classification, a core curriculum has been identified consisting of the knowledge and skills associated with the initial duties to which a new trainee would most likely be assigned.

Trainees are given 120 days (17 weeks) to qualify in their assigned work area. The qualification period begins with the first day of the orientation program. Qualification procedures are discussed below:

- **Orientation: Weeks 1-3.** The orientation program is a two-week session which provides an overview of the various production groups and their functions and interactions. It includes Du Pont orientation, safety, lock-tag and try procedures, defensive driving, CPR, use of protective equipment, basic radiological health, and security concerns. At the end of this two weeks, trainees are sent to their respective departments and given a one week departmental orientation program. Following this, the trainees are divided into smaller groups by facility assignment. They are then given an additional three days of facility general training. Upon completion, the trainees begin facility specific training.
- **Facility Specific: Weeks 4-17.** Facility specific training takes place over a 14-week period which is divided into three 4-week sessions and a 2-week remediable period (if needed). Each of the three 4-week sessions cover a particular knowledge and skills area the trainee must understand and be able to perform. The first week of each 4-week session consists of classroom and laboratory training. The remaining three weeks consist of on-the-job training on the trainee's shift assignment. During the classroom/lab weeks and/or at the end of the on-the-job periods, each trainee receives a qualification evaluation on the particular knowledge and skills area. These qualification evaluations identify strengths and weaknesses of the trainee. The trainee becomes qualified after passing all evaluations. If the trainee fails to pass the qualification evaluations he/she remains in a trainee status during weeks 16-17. During weeks 16-17 the trainee works with his immediate supervisor and the facility training specialist to remedy areas of weakness. At the end of week 17 all trainees receive qualification evaluations on those areas not previously completed satisfactorily.
- **Qualification Evaluation.** All qualification evaluations are performance based. A qualification evaluation consists of a written evaluation and a job performance evaluation. Written evaluations are not always required as some knowledge and skills areas may be more appropriately measured only by a job performance evaluation. A job performance evaluation is conducted in all cases. Each evaluation is specifically designed for each knowledge and skills areas to which the trainee is to be assigned.

4.5 REVIEW AND AUDIT

Audits to determine if procedures are properly used are carried out independently by the Separations Department and by the Separations Technology Department. In addition to this, special criticality audits are performed as dictated by proposed equipment or process modifications.

Reviews to determine procedural adequacy are performed at several levels within AED.

Separations Department Audits. Separations Department conducts a cross-audit of each operation by a committee that does not have direct responsibility for the operation. In addition, each facility is audited monthly, or at least quarterly, by a facility audit team to determine compliance with procedures. Individual audits within facilities by first-line supervision are made monthly to provide a detailed examination of specific job assignments. Individual audits are recorded on Job Observation Reports.

Separations Technology Audits. Auditing functions are performed by the Technical Assistance groups as part of their routine surveillance. These groups observe operations being carried out, collect and analyze data, and review operating logs or runbooks. Special surveillance is provided when chances of error are greater than normal or when consequences of errors would be hazardous.

Criticality Audits. A committee composed of one qualified member from each of the Separations Department, the Separations Technology Department, and the Health Protection Department audits material handling and storage. This audit occurs once per quarter to determine compliance with procedures, adequacy of procedures, and the training of personnel. A report on each audit and any follow-up action recommended is submitted to Separations PMT Management for implementation.

4.6 INSPECTION AND TESTING

This section gives a brief description of the inspection and testing of equipment categorized as Engineered Safety Features in Section 3.3.

4.6.1 Ventilation Systems

Separations Project Department heating and ventilating engineers provide services for balancing the ventilation system of 200 Area buildings. Airflow is measured in hoods and gloveboxes to establish conformance with standards. The Health Protection Department routinely inspects and tests the flow of air in potentially contaminated areas. Fans and blowers are routinely inspected by operating personnel. Filters such as HEPA types are inspected and tested for efficiency and leakage at the Oak Ridge Filter Test Facility before being sent to SRP. They are again tested in place, after installation by the Health Protection Department (8). Sand filters are continuously monitored and routinely inspected for degradation of efficiency.

4.6.2 Cooling Water Diversion

The cooling water diversion system is inspected and tested as follows:

Instruments. An alpha- and gamma-response test is made to determine whether the detectors will respond to a radiation source (9). The gamma device and the alpha device are tested periodically. The alpha detector must be valved out and drained during this test. The monitors are calibrated using calibration solutions of four different levels of activity. The response of the signal from the monitor is determined on a quarterly basis.

Other Equipment. In the process of the instrument-response tests, the alarms are automatically tested. A visual inspection of the equipment is made during walk-through inspections and during the response tests.

4.6.3 Confinement Barriers

Inspections and tests on the integrity of confinement barriers are made routinely and in many instances, continuously. Table 4-1 lists the confinement barriers (including those given in Section 3.3.3) along with examples of inspection and/or tests performed on them.

4.6.4 Radiation Shielding

Continuously operating ion chambers with recorders and alarms are located throughout the facilities mounted outside the canyons. These instruments continuously test the integrity of the shields. Also, personnel working in shielded areas wear dosimeters that measure dose. These dose data are recorded either monthly or quarterly, depending upon the locations. If shielding had failed, personnel doses would have increased. Frequent monitoring of penetrations in the canyon walls is conducted by Health Protection Department surveyors. Any abnormal radiation is carefully identified by signs or roping off of the area.

4.7 UNIQUE HAZARDS

During this analysis, there was consideration of and a search for conditions uniquely different from normal industrial practice. Injury statistics for the past five years were examined based on data from the Data Bank and from the Safety Department computerized listing of first aid cases and OSHA reportable cases. All of the injuries were first aid or medical treatment cases; no lost workday cases were reported during the analysis interval (November 1979 to October 1983, 4 yr). Most of the injuries were to the fingers or hands and the most frequent type of injury was a laceration.

The most frequent difficulty experienced with protective clothing is failure to wear the prescribed pieces or failure to adequately don the clothing. This occurs about twice per year in which contamination of the person occurs. Specific failures of protective clothing that occur include tears, penetration by sharp objects, saturation of clothing by liquids, disconnecting of breathing air hoses, and tape being pulled loose from the clothing.

TABLE 4-1. Inspections and Tests on Confinement Barriers

Description of Barrier	Types of Inspection, Integrity Qualification, and/or Test
Operating equipment (tanks, valves, sumps, etc.)	Liquid level indicators, sump alarms, visual inspection and radiation detectors
Shielding	Gamma Monitors
Hoods and gloves cabinet	Pressure differential indicators, air monitoring equipment, personnel hand surveys, contamination surveys
Waste packages, casks, and load lugger pans	Individually surveyed and inspected
Filtering System	Differential pressure instruments, air samples, DOP tests
Fuel or target cladding	Water sampled

Although protective clothing is a necessity for contamination control and injury reduction, several problem areas exist that indirectly affect operator safety and performance. One problem is that communication through an air-supplied plastic hood or through a respirator is difficult. When air-supplied hoods or suits are used, mobility is significantly reduced and entanglement of hoses is common when several people are working in close proximity to each other. Protective clothing creates a high temperature environment and manual dexterity is reduced because several pairs of gloves are usually worn. Improper removal of protective clothing is one of the leading causes of skin contamination.

There are no identifiable risks associated with cold feed preparation and storage so long as the chemicals remain within the confines of the intended vessels and piping. Unconfined, the hazards are toxicity, corrosion, and carcinogenicity. Cold feed operations were analyzed for the mechanisms and frequency of chemical releases from confinement that could afford a risk to operating personnel. Methodology used in this analysis and the hazards for various chemicals are discussed in Sections 5.1.7 and 5.3.7. Consequences are presented in Section 5.4.7.

In addition, some of the accidents discussed in Section 5 are initiated by events not directly related to radiologically induced effects but the resulting consequences are primarily radiological in nature.

4.8 CONTROL AND MANAGEMENT OF EFFLUENTS

The basic controls utilized at SRP are the requirements and limits defined in Technical Standards. Where there are pertinent OSRs, the limits defined in the Standard must be at or within those of the OSRs. Standards are the responsibility of the Technical Division of AED and usually originate there. They are authorized by the Technical Director after approvals by both the Technical and Manufacturing organizations.

The plant processes may be operated outside Technical Standards (but within OSRs) for tests or for other short-term special purposes using TAs. The approval for TAs are similar to those for Standards, but they are authorized by the Director of Manufacturing. Safety limits and limiting conditions for the 221-F Canyon Facility operations are defined in the OSRs and are included in operating procedures, set well within the limits derived from the Standards. Operating Limits are approved by the Separations and Separations Technology Departments, both of which are intimately familiar with the processes and equipment.

4.8.1 Effluent Radioisotopes

The 221-F Canyon Facility operations include the handling of many radioisotopes. Some are natural materials such as ^{235}U or ^{238}U . Others are the products of nuclear fission or result from the absorption of neutrons. Some of the radioactive isotopes occur in liquid, solid, and gaseous phases.

Table 4-2 lists the significant radionuclides that may be present in gases and vapors for the 200-Area operations. Many of these would be present only if a

TABLE 4-2. Significant Radionuclides Present in Gases and Vapors From 200-Area Operations

El. No.	Isotope	Radioactive Half-life
1	^3H	12.35 yrs
6	^{14}C	5.73E3 yrs
35	$^{83}\text{Br}^*$	2.40 hrs
	$^{84}\text{Br}^*$	31.80 mins
	$^{85}\text{Br}^*$	3.00 mins
	$^{86}\text{Br}^*$	54.00 secs
	$^{87}\text{Br}^*$	55.00 secs
36	$^{83}\text{Kr}^*$	1.90 hrs
	$^{85\text{m}}\text{Kr}^*$	4.40 hrs
	^{85}Kr	10.72 yrs
	$^{87}\text{Kr}^*$	76.00 mins
	$^{88}\text{Kr}^*$	2.80 hrs
	$^{89}\text{Kr}^*$	3.20 mins
	$^{90}\text{Kr}^*$	33.00 secs
53	^{129}I	1.7E7 yrs ^a
	$^{131}\text{I}^*$	8.10 days
	$^{132}\text{I}^*$	2.34 hrs
	$^{133}\text{I}^*$	21.00 hrs
	$^{134}\text{I}^*$	52.00 mins
	$^{135}\text{I}^*$	6.70 hrs
	$^{136}\text{I}^*$	83.00 secs
54	$^{131\text{m}}\text{Xe}$	11.89 days
	$^{133\text{m}}\text{Xe}^*$	2.30 days
	$^{133}\text{Xe}^*$	5.30 days
	$^{135\text{m}}\text{Xe}^*$	15.00 mins
	$^{135}\text{Xe}^*$	9.20 hrs
	$^{137}\text{Xe}^*$	3.90 mins
	$^{138}\text{Xe}^*$	17.00 mins
	$^{139}\text{Xe}^*$	41.00 secs

*May be significant from a criticality accident.

^a1.7E7 = 1.7×10^7 .

nuclear criticality accident were to occur. Others such as tritium, ^{85}Kr , ^{129}I , ^{131}I , ^{131}Xe , and ^{133}Xe are isotopes that are present even without the occurrence of a criticality accident.

Table 4-3 lists significant radionuclides that can be released as solids or liquids. These isotopes may be released to the air in particulate form, or to the ground in liquid form from 200-Area operations (of which F-Canyon is a part).

The release of radionuclides, as a result of 200-Area operations, is routinely monitored. In this report, overall radionuclide releases from the 200-Area operations (of which F-Canyon is a part), via gaseous effluents, have been listed for the years 1983 and 1984. For liquids, direct releases to surface streams from 200-Area operations have been listed. Liquid release via retention and seepage basins for the 200-Area operations are listed in the Seepage and Retention Basins SAR (11). Combined radionuclide releases for F- and H-Canyon operations via gaseous streams are listed in Table 4-4. Releases are via the F- and H-Area stacks. Liquid releases to streams for combined 200-Area releases are listed in Table 4-5.

4.8.2 Effluent Treatment and Control

There are five normal sources of liquid effluents from 200 F-Area, as shown in Figure 4-4. All liquid effluents (except those from Building 238-H) are treated (10) by providing a delay time before they reach plant surface streams. This is accomplished by using retention and seepage basins. The earth acts as a soil column for releases through the seepage process. As water migrates from the basins by seepage, the movement of radioactive elements (except tritium) is slowed. Details of this system are discussed in Section 3.3.2.

The only radioisotopes that so far have moved through the soil to reach a stream (Four Mile Creek) are ^3H , ^{90}Sr , ^{99}Tc , and ^{129}I .

Airborne Effluents. Airborne effluents and their treatment are shown in Figure 4-5. HEPA filters are used to collect radioactive particles from all highly contaminated air streams. Evaporation from the seepage and retention basins will be included in the Seepage and Retention Basins SAR (11).

4.8.3 Effluent Monitoring Program

All effluents from 200 F-Area facilities are monitored for radioactivity. The concentration of radioactivity dictates the action (if any), to be taken to reduce a release. Effluent control is discussed in the following sections for liquid and airborne releases.

Liquid Releases. The F-Area liquid effluents are shown in Figure 4-4. Each system is separately discussed below:

TABLE 4-3. Significant Radionuclides That May be Released As Particulates From the 200-Area

El. No.	Isotope	Radioactive Half-life
38	⁸⁹ Sr	52.00 days
	⁹⁰ Sr	28.00 yrs
39	⁹⁰ Y	64.00 hrs
	⁹¹ Y	58.80 days
40	⁹⁵ Zr	65.00 days
41	⁹⁵ Nb	35.20 days
44	¹⁰³ Ru	39.60 days
	¹⁰⁶ Ru	367.00 days
45	¹⁰⁶ Rh	30.00 secs
47	¹¹⁰ Ag	24.40 secs
50	¹²³ Sn	42.00 mins
51	¹²⁵ Sb	2.70 days
52	¹²⁷ Te	9.40 hrs
	¹²⁹ Te	69.00 mins
55	¹³⁴ Cs	2.05 yrs
	¹³⁷ Cs	30.20 yrs
58	¹⁴¹ Ce	32.50 days
	¹⁴⁴ Ce	284.30 days
59	¹⁴⁴ Pr	17.28 mins
60	¹⁴⁷ Nd	11.06 days
61	¹⁴⁷ Pm	2.50 yrs
	¹⁴⁸ Pm	5.39 days
63	¹⁵⁴ Eu	16.00 yrs
92	²³³ U	1.62E5 yrs ^a
	²³⁴ U	2.47E5 yrs
	²³⁵ U	7.1E8 yrs
	²³⁶ U	2.39E7 yrs
	²³⁷ U	6.75 days
	²³⁸ U	4.51E9 yrs
93	²³⁷ Np	2.14E6 yrs
94	²³⁸ Pu	86.00 yrs
	²³⁹ Pu	2.44E4 yrs
	²⁴⁰ Pu	6.58E3 yrs
	²⁴¹ Pu	13.20 yrs
	²⁴² Pu	3.79E5 yrs

^a1.62E5 = 1.62 x 10⁵.

TABLE 4-4. Radionuclide Releases to the Atmosphere From the 200-Area

Isotope	<u>Annual Release Activity, Ci</u>		
	1983	1984	Average
³ H	4.3E5 ^a	5.5E5	4.9E5
¹⁴ C	3.7E1	3.4E1	3.5E1
⁸⁵ Kr	5.7E5	8.4E5	7.0E5
^{89,90} Sr	2.7E-3 ^a	3.5E-3	3.1E-3
⁹⁵ Zr	1.1E-2	1.4E-2	1.3E-2
⁹⁵ Nb	1.8E-2	1.9E-2	1.9E-2
¹⁰³ Ru	2.0E-2	1.9E-2	2.0E-1
¹⁰⁶ Ru	8.1E-2	1.5E-1	1.2E-1
¹²⁹ I	4.1E-2	3.5E-2	3.8E-2
^{131m} Xe	2.0E1	1.8E1	1.9E1
¹³¹ I	7.1E-2	2.7E-1	1.7E-1
¹³⁴ Cs	5.6E-5	4.0E-5	4.7E-5
¹³⁷ Cs	9.9E-4	1.9E-3	1.5E-3
¹⁴¹ Ce	4.7E-4	8.4E-4	6.5E-4
¹⁴⁴ Ce	8.9E-3	1.3E-2	1.1E-2
^{235,238} U	4.4E-3	2.2E-3	3.3E-3
²³⁸ Pu	2.6E-3	1.2E-3	1.9E-3
²³⁹ Pu	7.2E-4	4.6E-4	5.8E-4
^{241,243} Am	2.6E-4	1.4E-4	2.0E-4
^{242,244} Cm	5.1E-4	2.6E-4	3.8E-4

^a4.3E5 = 4.3 x 10⁵; 2.7E-3 = 2.7 x 10⁻³.

Source: Environmental Monitoring at the Savannah River Plant, Annual Report for 1984, DPSPU-84-302, June 1985, and Environmental Monitoring in the Vicinity of the Savannah River Plant, Annual Report for 1984, DPSPU-85-30-1, (1985).

**TABLE 4-5. Measured Liquid Releases to Surface Streams
from 200-Area Operations**

<u>Annual Release Activity, Ci</u>			
Isotope	1983	1984	Average
$^3\text{H}^a$	$1.5\text{E}2^{b,c}$	$2.0\text{E}2^c$	$1.8\text{E}2$
$^{89,90}\text{Sr}^a$	--	--	--
$^{134,137}\text{Cs}$	$1.3\text{E}-1$	$2.2\text{E}-1$	$1.8\text{E}-1$
Alpha	$5.8\text{E}-3$	$3.0\text{E}-2$	$1.8\text{E}-2$
Other Beta-Gamma	$3.1\text{E}-2$	$8.9\text{E}-2$	$6.0\text{E}-2$

^aPrimarily groundwater migration from F- and H-Areas Seepage Basins.

^b $1.5\text{E}2 = 1.5 \times 10^2$.

^cSurface transport component.

Source: Environmental Monitoring at the Savannah River Plant, Annual Report for 1984, DPSPU-84-302, June 1985.

FIGURE 4-4. 200 F-Area Liquid Effluents

The 221-F Canyon building has two cooling water systems. The Circulating Cooling Water is monitored continuously in the 281-4F monitor house to detect any contamination from leaks in cooling coils of process vessels that contain activity. A recorder for the monitor is in the 221-F dispatcher's office. Pumps located in pits outside each monitor house continuously pump water from the discharge headers at a flow rate of 2 to 3 gpm through the monitors.

The Gamma Monitor (9) is a swirl cell monitor in which a hollow, rotating cylinder of water surrounds, but does not touch, a 0.016-in. steel-encapsulated sodium iodide detector. The alpha monitor consists of a cell into which water is continuously pumped to maintain a disk-shaped volume of liquid against a "Mylar"-covered zinc sulfide scintillation detector.

Alarm lights and audio signals located at the dispatchers office and the 211-F control room, alarm if the activity is above a preset alarm point. Water does not divert to the retention or seepage basins automatically. If activity is detected that is greater than 1000 d/m/ml (beta-gamma), or greater than 100 d/m/ml (alpha), the process is shut down and the water is diverted to the retention basin. In addition to the monitors, samples are taken, evaporated to dryness, and counted for activity. If activity is detected up to 300 d/m/ml (beta-gamma) or up to 100 d/m/ml (alpha), special sampling procedures are undertaken to locate the source of activity.

The Segregated Cooling Water system is a once-through system that normally discharges to Four Mile Creek. It is monitored continuously for radioactivity in 281-6F monitor house. A recorder for these monitors is located in 221-F dispatchers office. The monitors for the segregated system are the same as those explained previously for the circulating system. A 500 ml sample of this water is taken once per shift for analysis by Health Protection personnel. If activity up to 10 d/m/ml (beta-gamma) or 3 d/m/ml (alpha) is determined, special sampling and surveillance is undertaken to determine the source of release. If activity from 10 to 1000 d/m/ml (beta-gamma) or 3 to 100 d/m/ml (alpha) is determined, the water is diverted to the seepage basins. If activity greater than 1000 d/m/ml (beta-gamma) or greater than 100 d/m/ml (alpha) is determined, the water is diverted to the 281-8F retention basin, and the process is shut down until the leak is isolated.

All water entering the seepage basin is sampled by a Trebler sampler. Analysis of the sample for radionuclides is used to ascertain the quantity of radioactivity discharged to the seepage basin.

Airborne Releases. Gaseous effluent streams are shown in Figure 4-5. The off-gases from the denitrators in the A-Line that include the NO_2 fumes are exhausted through scrubbers to remove particulates of UO_3 and then cooled. This stream is combined with the 221-F Canyon dissolver off-gas stream that has been processed to remove radio-iodine. The combined streams then pass to the Sorption Column (F-8) where the NO_2 -component is sorbed into water forming HNO_3 . The essentially NO_2 -free gaseous effluent from the Sorption Column passes to the 292-F stack joining the off-gases from other facilities (Figure 4-5). The combined gaseous effluent stream is discharged to the environment 200 ft above grade.

FIGURE 4-5. 200 F-Area Airborne Effluents

The effluent gas streams from each of the various facilities are sampled on a routine basis to monitor the process performance of scrubbers, sorbers, or filters and in addition can be used to estimate the contribution of each facility to the chemical and radioactive burden of the stack discharge.

A system of continuous alpha monitors, iodine monitors, and beta-gamma monitors is installed on the 291-F stack. Data generated by this system directly relates the overall conformance of the facility to the guidelines included in Chapters I and XI of DOE Order 5480.1A. Chemical analyses of daily samples of the stack gas from the stack samplers generate data from which estimates of both chemical and radioactive pollutants can be made.

The air stream which passes through the UO_3 product handling systems after passing successively through the Product and General Dust Collectors is passed through one stage of HEPA filters and discharged to the atmosphere. Daily samples of the filtered air are examined for pollutants and conformance to guideline standards for uranium content given in DOE Order 5480.1A, Chapter XI.

4.8.4 Offsite Consequences of Radionuclides Released During Normal Operations (200-Area Operations)

Listed in Tables 4-6 and 4-7 are estimated consequences from radionuclides released during normal operation. For airborne radionuclides (Table 4-6), consequences are listed in terms of dose to the maximally exposed individual located at the site boundary and in terms of annual person-rem exposure to the surrounding population.

For radionuclides released with liquid effluents, only those radionuclides released directly to surface streams are considered here. If the activity from these radionuclides exceeds a specified limit, the effluent is diverted to the retention basin for decay or clean-up. After the activity is reduced, the effluent is pumped to the seepage basin. Seepage and retention basin operations and their consequences are considered in the Seepage and Retention Basins SAR (11). Estimated consequences for releases directly to surface streams for the 200-Area operations are listed in Table 4-7 in terms of dose to the maximally exposed individual at the site boundary and in terms of annual person-rem exposure to the surrounding population.

From the calculated values in the tables, it is apparent that offsite dose consequences and radiological risk from effluent releases during normal operation primarily results from the release of gaseous effluents. The principal radionuclide contributing to offsite doses from normal releases is tritium. For the purposes of comparison with estimates of consequences and risks resulting from releases associated with accidents, the population dose of about 75 person-rem/year for normal operational releases converts to 8.2×10^{-3} person-rem/hour.

The individual and population dose estimates presented in this section were calculated with computer programs developed by the Nuclear Regulatory Commission and modified by SRL (12). The programs used were MAXIGASP for

TABLE 4-6. Annual Dose Due to Release of Radionuclides in Gaseous Effluents

Radionuclide	Dose to Offsite Maximum Individual rem/year	Offsite Population Dose person-rem/year
^3H	1.6E-3	7.4E+1
^{14}C	2.2E-5	4.3E-1
^{85}Kr	5.3E-6	3.5E-1
^{131}Xe	6.4E-10	4.6E-5
$^{89,90}\text{Sr}$	9.8E-7	2.1E-2
^{95}Zr	8.2E-9	4.8E-4
^{95}Nb	6.2E-9	3.6E-4
^{103}Ru	7.8E-9	4.3E-4
^{106}Ru	4.2E-7	1.9E-2
^{129}I	9.2E-7	8.1E-2
^{131}I	3.2E-7	1.8E-2
^{134}Cs	2.8E-9	1.5E-4
^{137}Cs	1.1E-7	1.2E-2
^{141}Ce	2.8E-11	1.7E-6
^{144}Ce	5.2E-9	2.8E-4
$^{235,238}\text{U}$	2.9E-7	2.3E-2
^{238}Pu	2.9E-7	3.2E-2
^{239}Pu	1.2E-7	1.3E-2
$^{241,243}\text{Am}$	4.0E-8	3.8E-3
$^{242,244}\text{Cm}$	5.0E-8	3.8E-3
Total		7.5E1

TABLE 4-7. Annual Dose Due to Release of Radionuclides in Liquid Effluents

Radionuclide	Dose to Offsite Maximum Individual rem/year	Offsite Population Dose person-rem/year
^3H	7.1E-7	3.1E-2
$^{134}, ^{137}\text{Cs}$	3.5E-4	8.8E-1
Gross Alpha (max)	11.9E-6	1.3E-3
Other Beta-Gamma ^a	1.4E-4	3.5E-1
Total		1.1

^aDoses assume conversion factor equivalent to ^{137}Cs .

maximum individual dose from atmospheric releases, POFGASP for population dose from atmospheric releases, and LADTAP for maximum individual and population dose from liquid releases.

4.9 SAFETY MANAGEMENT SYSTEMS

It is the policy of SRP that safety and protection of employees comes first (3), and there are two separate organizations dealing with safety. One concerns itself with the protection of the individual and his/her environment from the harmful effects of radiation, and the other deals with the industrial safety of the worker.

4.9.1 Radiation and Contamination Control

The Health Protection Department concerns itself with radiation and contamination control. The Health Protection Department (Figure 4-6) has many functions. Some of these functions are listed in Table 4-8.

It is SRP policy to limit the radiation exposure of employees to as low as reasonably achievable, i.e., the ALARA concept (4). Radiation exposure plant guides have been developed and used to help control the exposure of operating personnel in achieving this goal. These values are given Table 4-9. Typical annual exposures for personnel assigned to the F-Canyon facility are shown in Table 4-10.

The exposure of whole body to penetrating radiation is estimated by combining:

1. The radiation dose determined from thermoluminescent dosimeter (TLD) readings; these are worn by all personnel who have business in areas which are potentially subject to exposure to ionizing radiation.
2. The neutron radiation dose as determined by thermoluminescent neutron dosimeters (TLND).
3. The dose received from tritium assimilation, determined by bioassay.

Internal deposition of radionuclides is detected by routine programs for analysis of urine and by whole body or chest counting (in vivo) techniques.

Locations within the 221-F Canyon Facility are classified into three categories that depend upon expected levels of radiation or contamination. A Clean Area is an area where no radioactive materials are handled and where the radiation and contamination levels are equivalent to natural background. A Regulated Area (RA) is one in which radioactive materials are handled or where the radiation level due to beta-gamma is less than 300 mrad/hr (and the gamma component is less than 50 mrem/hr) and the contamination (surface, transferrable) is low. A Radiation Zone (RZ) is an area where the radiation or contamination levels exceed the limits for a Regulated Area.

FIGURE 4-6. Health Protection Department Organization

TABLE 4-8. Health Protection Department Organizational Responsibilities

Group	Responsibilities
F- and H-Area Survey	Radiation and Contamination Surveys Industrial Hygiene Surveys Minimize personnel radiation exposures (ALARA) Minimize skin contamination Minimize facility contamination Minimize environmental releases Prevent internal assimilations of radioactive materials Train plant personnel in Health Protection aspects of jobs Coordinate Personnel Protective Clothing and Equipment Programs Audit work and procedures
Dosimetry and Industrial Hygiene	Distribution, collection, and evaluation of personnel dosimeters Maintain bioassay records Collection and quantitative analysis of excreta for contaminants Count individuals and estimate body burdens Maintenance, calibration, and distribution of survey and monitoring instruments
Environmental Monitoring	Quantitative analysis of air, water, milk, flora, fauna Evaluation and publication of environmental samples Instrumental analysis of environmental monitoring data Investigation of releases
Occupational Health Technology	Technical support for area surveys, dosimetry, and environmental monitoring

TABLE 4-9. Plant Radiation Dose Guide Values^a

Type Exposure, Organ	Dose in Rems	
	Per Qtr	Per Yr
Occupational Exposure:		
Whole body, head and trunk, active blood forming organs, gonads, lens of eyes, red bone marrow	3	3
Skin, other organs, tissue, and organ systems (except bone)	5	15
Bone and forearms	10	30
Hands and feet	25	75
	<u>Plant Emergency^b</u>	<u>Life Saving</u>
Emergency Exposure:		
Whole Body	25	100
Hands	100 ^c	200 ^d

^aValues at or less than those given in Department of Energy, Environment, Safety and Health Manual, Chapters XI, XII.

^bInvolving protection of property or personnel.

^cIncludes whole body exposure.

^dIn addition to whole body exposure.

TABLE 4-10. Average Exposures to Employee Groups Assigned to F-Canyon Facility During 1984

Employee Group	Number in Work Group	Annual Average Exposure (rems)	Annual Total Exposure person-rem/yr
Separations	136	0.254	34.6
Maintenance	14	0.464	6.5
Electrical and Instrumentation	15	0.173	2.6
Health Protection	22	0.400	8.8
Laboratory	117	0.531	62.5
Separations Technology	48	0.018	0.9
Total			116

Source: Inter-office Memorandum to A. G. Evans from N. D. Johnson. "Average Exposures to Employee Groups Assigned to F-Canyon Facility During 1984." October 15, 1985.

All work within an RA or RZ is controlled by written, approved procedures that have been developed into the intent to complete tasks with minimum exposure to radiation. These procedures are prepared and approved prior to entry into a RZ or prior to starting non-routine work in a regulated area.

All building areas occupied by personnel are surveyed routinely (some continuously) using portable or permanently installed instruments. Air samples are taken where the potential for airborne activity exists. When air activity exceeds the values shown in Table 4-11, respiratory protection equipment is required.

Protective clothing is prescribed in the procedures for work in RAs or RZs when an actual or potential contamination hazard exists.

4.9.2 Occupational Doses from Normal Operations

Whole body exposure to penetrating radiation during normal operations is monitored by the Savannah River Health Protection Department as indicated above. Exposure includes both external radiation and exposure from inhalation or ingestion of radionuclides, principally tritium. Values obtained from Health Protection records for annual exposure lists for various occupational groups for 1984 are listed in Table 4-10.

The values in Table 4-10 indicate that exposures to various occupational groups are being controlled within the dose guidelines presented in Table 4-9.

4.9.3 Industrial Safety

The SRP safety policies are intended to help employees avoid injuries and provide a safe environment in which to work. Management directs the program through the Plant and Area Central Safety Committees. The program includes planned educational activities on a daily, weekly, and monthly basis.

The Plant Central Safety Committee is composed of top level managers, Chairmen of Area Central Safety Committees, and Safety and Fire Protection Supervision. This body establishes policy and plant-wide procedures. There are ten permanent subcommittees as shown in Figure 4-7. The primary functions of each of these subcommittees, are listed in Table 4-12.

The 200 F-Area Central Safety Committee is composed of a representative from each of the departments regularly working in the 200 F-Area. This committee is primarily a coordinating group for the departments in the area. The committee reviews all departmental reports of injuries and accidents within the area. The collective membership of the committee supplements departmental responsibility for personnel safety. This committee has three standing subcommittees: Accident Prevention, Fire Prevention, and Off-the-Job Safety.

4.9.4 Industrial Hygiene

The purpose of the industrial hygiene program is to protect plant personnel and plant environs against hazards from nonradioactive materials. This

TABLE 4-11. Airborne Radioactivity Concentrations Requiring Filter or Air-Supplied Respiratory Protective Equipment

Element	Concentration, ^a microcuries/cc	
	Filter Pac	Air-Supplied Respirator/Suit
Unidentified		
Alpha	2.0E-12	1.0E-10
Beta-gamma	1.0E-9	5.0E-8
Uranium		
Natural	6.0E-11	3.0E-9
Enriched	3.0E-11	1.5E-9
Americium	6.0E-12	3.0E-10
Neptunium	4.0E-12	2.0E-10
Curium ^b	2.0E-12	1.0E-10
Californium	2.0E-12	1.0E-10
Plutonium	2.0E-12	1.0E-10
Thorium	2.0E-12	1.0E-10
Tritium	NA	5.0E-6
Iodine	NA	9.0E-9
Mixed fission product	1.0E-9	5.0E-8

^aBased on 40-hr week.

^bThe Radioactivity Concentration Guides (DOE Environment, Safety and Health Manual, Chapters XI, XII) for the various isotopes of curium produced during the transplutonium program at SRP ranged from 9.0E-12 for ²⁴⁴Cm to 6.0E-13 for ²⁴⁸Cm. Because of the low yield of ²⁴⁸Cm (0.05 weight percent, mass abundance), the RCG for unidentified alpha isotopes will be followed for all isotopes of curium.

FIGURE 4-7. SRP Safety Committees

TABLE 4-12. Subcommittee Primary Functions

Subcommittee	Primary Functions
Policy Review	Reviews and approves changes in Safety Manual. Reviews and approves injury and unusual incident investigation reports.
Safety Activities	Devotes itself to increasing the involvement in and expanding efforts in the planning of onplant safety programs.
Process Hazards Management	Responsible for coordinating and auditing process hazards management programs.
Off-the-Job Safety	Reviews off-the-job injuries and develops programs to improve the performance.
Occupational Health	Devotes itself to special hazards involving industrial hygiene, hearing conservation, handling of radioactive and toxic materials.
Emergency Procedures	Devotes itself to the ability of plant forces to respond to emergency situations.
Traffic	Devotes itself to traffic problems, road hazards, and vehicle accidents.
Housekeeping	Reviews plant housekeeping standards and develops programs and procedures to improve housekeeping performance.
Security	Responsible for directing the security program.
Safe Distribution	Coordinates and directs the receiving, handling, and shipment of hazardous materials ("RHYTHM").

includes hazard recognition, evaluation, and other environmental control factors.

The Health Protection Department (Figure 4-6) is responsible for the program, except for noise and ventilation surveys and epidemiological studies. Noise and ventilation surveys are the responsibility of the Separations Project Department, and the Medical Department is responsible for epidemiological studies.

The Industrial Hygiene Group is part of the Dosimetry and Industrial Hygiene Group (Figure 4-6). This group provides technical expertise, procedural guidance, and conducts special hygiene surveys. The Health Protection Department F- and H-Areas Survey Group provides surveys within these areas.

Inspections are conducted in operating areas, and all newly purchased materials used are reviewed for potential hazard. Special procedures are used when handling potentially carcinogenic materials.

4.9.5 Fire Protection

A Fire Prevention Subcommittee of the 200-Area Central Safety Committee devotes itself to fire prevention, and fire fighting capabilities. This subcommittee works closely with the Fire Protection Group within the Safety Department. The Fire Protection Group maintains adequate mobile fire fighting equipment, trains personnel, and inspects protection equipment.

4.9.6 Criticality Control

There are many places in the 221-F Canyon Facility where fissionable materials are handled. Detailed procedures developed by criticality specialists are the primary control used for nuclear criticality control. These procedures are based on Technical Standard limits and on criticality calculations, which are the general responsibility of criticality safety specialists in SRL with support from the Separations Technology Department criticality specialists. These specialists perform detailed calculations of permissible masses, concentrations, and dimensions for the process equipment under the guidance and review of the SRL specialists.

In certain instances, equipment and/or facilities are designed and built to be geometrically favorable from a nuclear criticality accident, e.g., slab tanks. In some instances, instrumentation is used as a secondary control to guard against criticality. In addition to these controls, detection and alarm instrumentation, audits, and training are basic control devices.

Procedural and personnel controls as they relate to criticality control are discussed below.

Procedural Control. Technical Standards for the 221-F Canyon Facility Operations represent the highest level of control of the process. They specify limits to ensure that operation of the various process steps will be

safe to personnel and to the environment. Technical Standards on Nuclear Safety provide limits or specifications for the following:

1. Handling and storage of fissionable materials
2. Maximum permissible amounts of fissionable isotopes and mixtures of fissionable isotopes
3. Permissible methods for quantity and concentration determinations

The Technical Standards containing criticality control limits for fissionable isotopes processed in the 200 F-Area Facility are listed in Table 4-13.

Operating procedures are prepared by Separations Department personnel assigned to the various facilities. These operating procedures are reviewed by the Separations Technology Department for their nuclear criticality control adequacy. If special nuclear criticality control items are considered necessary, they are prominently included directly in the procedure.

Area Criticality Audit Committees are maintained in the 200-Area with membership from the appropriate Production Department (Chairman), Health Protection, and from both Separations Technology and Reactor and Reactor Materials Technology. These committees conduct audits of conditions, practices, and procedures to assess their adequacy for nuclear criticality control. The committees issue a report of each audit, and the Separations Department issues a followup report stating the action taken on each deficiency and recommendations.

The Nuclear Safety Review Committee (NSRC) has responsibility for audit of criticality safety for the entire SRP site. This committee reviews methods of handling and processing fissionable materials to ensure that a critical mass will never be accidentally accumulated. The NSRC is composed of the Program Manager for Reactors (Chairman), the General Superintendent of Technical (Secretary), the Program managers for Separations and Waste Management, and the Director of the SRL Nuclear Reactor Technology Section. The committee meets at least five times per year and its members receive copies of all criticality audit reports and reports of abnormal conditions or occurrences that have criticality implications. At an annual meeting the committee hears a review of the state of criticality safety by each of the SRP Production Superintendents. In quarterly meetings with the members of the criticality Audit Committees, the NSRC hears more detailed reports on the facilities and operations and inspects facilities of special concern when appropriate.

Criticality training is the responsibility of line supervision, with the assistance of appropriate safety and technical specialists. Criticality safety training is one aspect of the overall program to ensure the safety of the process and the personnel. The continuing effort is to keep the subject of criticality safety fresh and alive by presenting it in new forms and with new aids such as films and booklets. Knowledge of the subject is verified through questioning by supervision and the audit committee.

Personnel Control. Nuclear incident monitors (NIMs) are provided at strategic locations throughout 200-Area facilities (13). These monitors are provided

**TABLE 4-13. Technical Standards Containing Criticality Control Limits
for Isotopes Handled in 200 F-Area Facilities***

Facility and/or Operation	Document No.	Title
General	DPSTS-NIM	Nuclear Incident Alarm Systems
221-F, Canyon	DPSTS-0.06	Canyon Nuclear Safety Standards
221-F, B-Line	DPSTS-0.09	B-Line Nuclear Safety Standards
221-F, FB-Line	DPSTS-0.09- Supplement	Nuclear Criticality Safety Supplements

*These give general limits only. Specific limits for each unit operation are given in the standards for that operation.

wherever fissionable materials are stored, or processed in sufficient quantity for a potential critical configuration. The monitors alarm to warn personnel to exit the area along previously established well marked routes.

Each nuclear incident alarm system associated with a potential incident site consists of at least two individual NIMs. Each NIM is located within 100 ft of a potential incident location (14).

NIMs are set to alarm within 5 sec in a steady radiation field of 0.5 to 3 R/hr or if the total dose received at the monitor within one minute exceeds 50 mR. However, for service in unattended, shielded areas such as in the hot canyons, the instruments are set well above the prevailing gamma radiation background which may be as high as 100 R/hr.

Remote alarm bells are used in areas where it would otherwise be difficult to hear the alarm. Personnel are trained to respond to a NIM alarm by immediately going to a specified location by a previously prescribed route (7).

4.10 REFERENCES

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5.0 ACCIDENT ANALYSIS

The F-Canyon operations have been evaluated on the basis of three considerations: 1) potential radiological hazards, 2) potential chemical toxicity hazards, and 3) potential conditions uniquely different from normal industrial practice. Each accident sequence identified as the result of this analysis has been evaluated for its effect on personnel at the work site, at other locations on the plant site, and outside the plant boundary. Routine conditions were not considered in this section of the analysis other than for comparative purposes.

The F-Canyon facilities and operations described in Section 3 have been previously analyzed in a comprehensive Systems Analysis (1). This analysis systematically identified and evaluated the hazards utilizing a probabilistic methodology based on both actual occurrences and postulated occurrences. Most initiating events in the F-Canyon operations were identified and quantified from data in the 200-Area Fault Tree Data Bank (2). Other initiating events were identified from analyses of other similar facilities (3). Where sufficient operational data were not available, fault trees were used to quantify the frequencies of postulated initiating events (4,5). Broad categories of initiators include process occurrences, natural phenomena, and external events; other occurrences affecting support systems, and engineered safety features can impact process operations. These events are described in Section 5.1.

Once the initiating events were identified, various analytical techniques were utilized. These included developing initiator frequencies from the 200-Area Data Bank, identifying the radionuclides present at any point in the process in order to calculate dose due to transport of the radionuclides to man, evaluation of accident sequences via event trees, developing frequencies for events for which no data were available via fault trees, and analyzing on-plant and off-plant doses using various models. These techniques are described in Section 5.2.

Events subsequent to the initiating event were determined by the system characteristics and engineering data. Event trees were used to describe and quantify frequencies for a particular accident sequence from the initiating event to a final outcome. These release sequences were based on the physical confinement barriers between the initiating event and the release of activity at the points of interest. Most of the frequencies used to quantify the event trees were developed in the F-Canyon Systems Analysis (1) and are summarized in Section 5.3.

Once the initiating events and subsequent accident sequences were identified and quantified, the amount of material released outside the primary confinement was determined. This calculation required consideration of the specific nuclides present in the process, the physical form, the mechanism of release, and the performance of the engineered safety features designed to prevent their release. Most factors are specific to the accident being considered, and the supporting rationale are included with the accident analysis. Some factors were included with the calculation of the source terms and are discussed in Appendix A of the F-Canyon Systems Analysis. These source term calculations are summarized in Section 5.4.

The quantified source terms were then evaluated for their effects on personnel at the worksite, at other locations on the plant site, and outside the plant boundary. Risk in this analysis is defined as the product of the expected frequency that an event sequence will release radioactivity and the radiological consequences of that release. These consequences were compared with those normal releases due to F-Canyon operations in Section 5.5.

The final considerations in the accident analysis are the design features and physical and administrative controls which are implemented to reduce accident risks. These mitigating measures are discussed in Section 5.6.

5.1 INITIATORS AND DESCRIPTION OF ACCIDENTS

Initiating events for accidents were systematically developed in the F-Canyon Systems Analysis (1) using the SRP Incidents Data Bank (6), facility-specific considerations, and the 200-Area Fault Tree Data Bank (2,7).

The results of this systematic determination of potential accidents are grouped into three categories based on accident initiators: 1) natural phenomena, 2) external events, and 3) process related events. Other events due to chemical hazards, support system failures, and engineered safety feature failures, were also considered. The impact of support system and engineered safety feature failures on process operations was analyzed; however, these potential common mode failures were not analyzed as independent accident sequence initiators beyond the extent to which they are intrinsically a part of the 200-Area data base.

Potential process-related initiating events are defined according to four categories (8): high energetic events, medium energetic events, low energetic events, and residual release events. These categories have been selected because of the increased potential for release outside the primary confinement as energy of the event increases. Other accidents caused by natural phenomena, external events, support system failures, engineered safety feature failures, and chemical hazards are discussed separately.

5.1.1 Natural Phenomena

Wind, flood, earthquake, temperature, snow, rain, and lightning extremes may adversely affect operations. The following paragraphs discuss these effects with more detail appearing in Reference 9.

5.1.1.1 Winds

Two types of winds are considered in this section: straight winds (including hurricanes) and whirling type winds (including tornadoes).

Straight Winds. Reinforced concrete structures are not expected to be affected by straight winds even at speeds up to 175 mph. The minimum velocity of a hurricane is 73 mph. Hurricane Gracie, which passed north of the plant

site in 1959, was the only storm in the history of the plant in which hurricane intensity winds swept SRP (10).

Tornado. The F-Canyon structure is of Maximum Resistance Construction (MRC) and is designed to withstand a Design Basis Tornado (DBT) (11,12,13). The DBT is defined as a tornado having a rotational speed of 230 mph with a radius of 230 ft and a maximum translational speed of 50 mph, and a total pressure drop of 1.5 psig at a maximum rate of 0.5 psi/sec. The combination of rotational and translational wind speed is 280 mph, which corresponds to a Fujita Intensity Five tornado (13,14).

5.1.1.2 Earthquake

The Savannah River Plant is located in an area where damage from earthquakes could occur. The Design Basis Earthquake (DBE) (11,12) for MRC structures is an earthquake with a 0.2 g horizontal ground acceleration. Using the Modified Mercalli (MM) Intensity Scale, this corresponds approximately to a MM VIII Earthquake. The MM Scale is shown in Table 5-1.

5.1.1.3 Meteorite Impact

Meteors are extraterrestrial solid matter moving in a closed orbit in the solar system or are debris from comets entering the solar system. When meteors come under the gravitational pull of the earth, their courses are altered and some enter the earth's atmosphere.

The earth's atmosphere has a filtering effect on meteors. Those with an initial mass of less than 100 lb (about the size of a football) are usually vaporized while traveling through the atmosphere (15). A large meteorite striking the F-Canyon building may cause some damage; however, this event is considered hypothetical only.

5.1.1.4 Floods

The probability of release of activity from accidents initiated by large-scale flooding is negligible. The adjacent streams in broad valleys are more than 100 ft lower in elevation than Building 221-F.

5.1.1.5 Other Natural Phenomena Related Events

Extremes in temperature, snow, rain, and lightning, may adversely affect operations, but will not result in accident sequences that lead to direct releases of radioactive materials.

Cold weather has little effect on F-Canyon except through its effect on auxiliary services. Below -8°C (for the services located outside the building), lines plug, valves and pipes burst, alarms fail to operate, instrumentation may become ineffective, and equipment with movable parts may become immobilized. Icing can be a cause of common mode failure.

TABLE 5-1. An Abbreviated Modified Mercalli Scale for Earthquake Description

Modified Mercalli Intensity (MM)	Description
I	Detected only by sensitive instruments.
II	Felt by a few persons at rest, especially on upper floors; delicate suspended objects may swing.
III	Felt noticeably indoors, but not always recognized as a quake: standing autos rock slightly, vibration like passing truck.
IV	Felt indoors by any, outdoors by a few; at night some awaken; dishes, windows, doors disturbed; motor cars rock noticeably.
V	Felt by most people; some breakage of dishes, windows, and plates; disturbance of tall objects.
VI	Felt by all; many frightened and run outdoors; falling plaster and chimneys; small damage.
VII	Difficult to stand; damage to buildings varies depending on quality of construction; noticed by drivers of autos.
VIII	Panel walls thrown out of frames; fall of wall, monuments, chimneys; sand and mud ejected; drivers of autos disturbed.
IX	Buildings shifted off foundations, cracked, thrown out of plumb; ground cracked; underground pipes broken.
X	Most masonry and frame structures destroyed; ground cracked, rails bent, landslides.
XI	Few structures remain standing, bridges destroyed, fissures in ground, pipes broken, landslides, rails bent.
XII	Damage total; waves seen on ground surface; lines of sight and level distorted; objects thrown up into air.

Snow represents an aggravation that can affect normal operations. The largest recorded snowfall of 18 in fell in February 1973. The average annual snowfall over the last 45 years has been 70.9 in. Snowfall of 1 in or more has fallen in 7 of the last 45 years.

Rain can be both an aggravation to a deteriorated situation and a damage-inflicting condition in itself. Although not directly affecting canyon operations, damage to excavations, cave-in of drain lines, near overflow of seepage basins, seepage into sand filters, and spurious annular leak alarms on waste tanks have occurred as a direct result of hard rains. It is conceivable that during prolonged heavy rain, water could enter the building through the expansion joints; however, no damage or contaminated water runoff is expected to occur as a result of leakage.

The principal adverse effect of lightning is interruption of normal electrical power. Lightning can also result in spurious alarms being actuated.

5.1.2 Externally Induced Events

Most safety-related occurrences are the result of failures within the system, or as a result of some action intentionally directed toward the system. It is possible, however, for damage to be inflicted on a system as a result of some occurrence originating outside the system or in an apparently separate system.

5.1.2.1 Aircraft Crash

A large airplane crashing into the wall of the F-Canyon building will cause damage. It can be expected to completely disrupt the ventilation system and all external components such as the stack, filter, fans, etc. It is assumed to cause some damage inside the building, although the 24-in of reinforced concrete will localize the effect of the crash. The consequence of the crash will be amplified by fire after the crash. It is expected that small breaches in the outer wall would occur.

In addition to the large aircraft, DOE operates helicopters over the SRP site (16). However, a crash of these lighter aircraft (~6,000 lb) would not be expected to cause a measurable consequence within the facility and, therefore, this contribution is neglected.

5.1.2.2 Adjacent Explosion

Adjacent explosions are explosions that occur in facilities outside F-Canyon that might impact F-Canyon operations. In general, however, adjacent explosions are not expected to result in significant releases from canyon systems. Thus they are not considered credible accident initiators.

5.1.2.3 Adjacent Fire

Fires occurring in facilities adjacent to the F-Canyon building may have effects outside of the facility in which they occur. In general, however, adjacent fires are not expected to result in significant releases from canyon systems. Thus, they are not considered credible accident initiators.

5.1.3 High Energetic Events

A high energetic event is defined as one that releases sufficient energy to destroy the first confinement barrier. It will also damage the secondary confinement barrier, allowing radioactive materials to directly reach rooms occupied by personnel, or reach the environment outside the facility (8).

The initiator of such an event would need to be an explosion of a magnitude which could potentially produce severe enough damage to meet the criteria for a high energetic event.

5.1.4 Medium Energetic Events

A medium energetic event is defined as one that will cause penetration of the primary confinement barrier, and will cause materials to bypass the second confinement barrier for a short period of time (8). Initiating events which could lead to medium energetic events are discussed below.

5.1.4.1 Internal Fire

A fire occurring within the F-Canyon process area can lead to a medium energetic event. Causes of fires with F-Canyon include process heat, welding, electrical shorts, spontaneous combustion, smoking materials, friction, lightning, and lighting. Other unknown causes are also responsible for fires.

In solvent extraction, the three constituents necessary to start a fire are present, but the solvent (fuel) must be heated in air (oxygen) to the autoignition temperature, or to the fire point in the presence of an ignitor. The primary source of heat in the canyon is steam, which is capable of heating the solvent to the fire point, but not to the auto-ignition temperature (~230°C).

If a fire should occur in solvent extraction, it would probably occur outside the mixer settlers in the canyon sump. In addition, the solvent must be heated to the fire point (10 to 70°C above the flash point). Once the solvent has been heated above the fire point, a fire can occur if an ignitor is available. Examples of ignitors are adjacent fires, electrical short, friction, and static electricity. Solvent also vaporizes prior to burning, tending to concentrate nonvolatile components in the unburned residue. This phenomena has been observed in burning tests with solvents (17,18).

One case of a resin fire in ion exchange in F-Canyon has been reported. On October 1, 1964, a fire of short duration occurred in Section 10 (Hot Canyon) on top of the 10.1 tank around the single stage primary recovery column (PRC).

No resin was in the column, and the temperature within the column during the fire did not change. The fire is believed to have started in degraded resin that had gradually accumulated on the tank top from spills and perhaps from upsets within the column. The resin ignited from friction heat around the agitator shaft or from a spark from the impact wrench on the crane. The fire was not detected by the Section 10 "Fireeyes" or thermocouples in the exhaust air ducts; the thermocouples would not be expected to detect an increased air temperature with the cell cover off because the hot air would rise above the covers. The 9.3B tank and PRC agitators were shorted out by the deluge water, and the 9.3B agitator was replaced. The new three-stage column (PRC) was modified before it was installed to provide a system for spraying caustic or water on the exterior of all three columns, which should neutralize the resin and keep it damp.

5.1.4.2 Uncontrolled Reaction

In the canyons, uncontrolled reactions are the most rapid means of losing control of large volumes of highly contaminated materials. These reactions can manifest themselves in several forms including an eructation (a sudden loss of part of the contents of the vessel), foaming, boilover, gassing, or simply an undesirably high temperature resulting in degradation of materials, or evolution of excessive contaminated or toxic vapors.

Equipment is not usually damaged directly in an uncontrolled reaction, as it is in an explosion; contamination, dispersion of radioactivity, and corrosion are usual byproducts. Such occurrences pressurize the vessel, frequently forcing liquid through instrument lines into areas accessible to personnel. Contaminated materials can find their way through cracks and expansion joints into personnel corridors or into the soil beneath the building.

Loss of vessel contents from foaming, boilover, or gassing is not as spectacular as an eructation, but can cause local recovery and contamination problems as well as increasing activity in the ventilation system.

Explosions have not occurred in canyon vessels. However, in 1953, a damaging explosion in an evaporator occurred at TNX as a result of a reaction involving uranyl nitrate, TBP, and nitric acid. It is concluded that the greatest likelihood of a canyon explosion will be the result of some unforeseen reaction based on a number of lesser uncontrolled reactions that have occurred but were not anticipated.

Explosions of severe magnitude could potentially generate sufficient energy to allow radioactive materials to penetrate the secondary confinement. However, the structural integrity of the canyon vessels and secondary confinement minimizes the potential to the point that explosions are treated as medium energetic events.

Dissolver System. One of the principal safety problems encountered in the dissolver system is rapid pressurization of the dissolver which occurs because of excessive reaction rates, failure of the off-gas exhausts, or plugging of the off-gas system. Such occurrences would eject solution from the dissolver and contaminate the process cell and ventilation system.

Although none have occurred, a second mechanism for the loss of material from the dissolver via uncontrolled reaction is an explosion. One possibility is a hydrogen explosion during the coating removal step in the Purex process. The evolution of hydrogen is controlled by the addition of sodium nitrate to suppress the reaction and the controlled rate of addition of sodium hydroxide solution. A dangerous amount of hydrogen could be generated only in the event that the clad slugs were not completely covered with sodium nitrate solution and that concentrated sodium hydroxide solution came in contact with bare aluminum, or if NaNO_3 were accidentally omitted.

A second mechanism for an uncontrolled reaction is the cyclic evolution of ammonia (during coating removal) and nitric acid (during uranium metal dissolution) from the dissolver. This makes possible the formation of variable amounts of ammonium nitrate and the other solids, including small amounts of silver azide. Deposits of these solids could plug the off-gas lines with solids which can explode as a result of the inherent nature of the compounds.

Hydrogen explosions are also possible in the dissolver as discussed in Section 3.2.2.2. Silver reactor explosions in the off-gas treatment system are discussed in Section 3.2.2.3.

Head End System. Potential explosive conditions can arise in the head end system from explosive mixture of gases such as hydrogen gas generated by radiolysis of process solutions.

Rapid pressurization of the slurry tank followed by eruption of the contents can occur during cake dissolution. Nitrous acid (formed by the addition of sodium nitrate to dilute nitric acid) decomposes with the evolution of nitrogen oxides. Too rapid an addition of sodium nitrate results in excessive gas evolution and foamover of cake slurry to the canyon sump. This problem is minimized by maintaining as low a temperature as possible during dissolution to increase the solubility of nitrous acid and/or nitrogen oxides, and by adding the sodium nitrate at a low rate.

Solvent Extraction System. The only conceivable mechanism for explosion in the solvent extraction system is a rapid exothermic reaction from unknown chemicals added in error, or from the nitration of the solvent with nitric acid. The probability of this is remote since the nitration of solvent requires high concentrations of nitric acid and high temperatures.

Evaporators. The extensive use of evaporators in feed adjustment and waste handling processes entails the hazard of rapid pressurization and/or explosion. This can occur by the rapid reaction of nitric acid with excessive amounts of entrained solvent. Analysis and administrative controls limit entrained solvent in evaporator feed to 0.5 vol % by analyses, steam pressure to 25 psig, and evaporator temperature generally to 118°C (the limit in the Operational Safety Requirement is 130°C ; see Section 3.2.2.8.5). Under these conditions, the probability of excessively rapid reactions is small. Periodic

cleanout of the evaporator feed tank must be made to prevent gradual accumulation of organic material.

Ammonium nitrate, when heated with organic matter, is potentially explosive. However, large amounts of ammonium nitrate are known to enter the process, and have had no adverse effects when evaporating acidic solutions using proper administrative controls.

Solutions of hydroxylamine nitrate (HAN) are increasingly unstable at acid concentrations above 0.1M. Once initiated in boiling acidic solutions, HAN decomposition proceeds autocatalytically and completely, releasing large volumes of gas. This reaction can be prevented by destroying the HAN with sodium nitrite (NaNO_2) before evaporation. Alternatively, the decomposition of HAN can be controlled by feeding the HAN solution to an evaporator containing boiling high strength acid ($>2\text{M}$) that destroyed the HAN as it enters.

Concentrated hydrazine nitrate solutions are sensitive to impact; therefore, evaporation of solutions containing hydrazine and nitrate ion must be avoided. Hazardous conditions are avoided by oxidizing the hydrazine and hydrazoic acid to N_2O and N_2 by adding at least two moles of sodium nitrite per mole of hydrazine to the evaporator feed (19,20).

Hydrazine (N_2H_4) solutions are nonexplosive, and pure liquid hydrazine cannot be ignited by a hot wire (21). Aqueous solutions of hydrazine are not sensitive to impact, friction, or detonation. Hydrazine nitrate solutions above 50% nitrate are sensitive to impact; therefore, evaporation of solutions containing hydrazine and nitrite ion must be avoided. In F-Canyon, frame raffinates with possible hydrazine present are evaporated after adjustment with sodium nitrate. Alkaline evaporation is not considered hazardous because the hydrazine ion is unstable and decomposes to N_2 and H_2 . Nitrous acid (HNO_2) can react with hydrazine (N_2H_4) to form hydrazoic acid (HN_3) according to the equation

The lower explosive limit for aqueous solutions of hydrazoic acid is 17%. This concentration can be attained only through condensation of vapors that are formed by the distillation of solutions that contain at least 0.05M HN_3 at 30°C . Calculated estimates of the lower limit of hazardous concentrations of HN_3 show that sufficient HN_3 cannot evolve into the vapor phase at any temperature to damage a run tank, even if a spontaneous chemical reaction should occur.

Process solutions may contain mixtures of HN_3 and mercuric nitrate that yield mercuric azide. Under certain conditions mercuric azide is explosive. Laboratory tests show that spontaneous detonation can occur when the solvent is water and the concentration of mercuric azide is about 0.023M. Spontaneous detonation does not occur at concentrations less than 0.02M mercuric azide, even with water as the solvent. Precipitates formed in acidic solutions could not be detonated if they remain covered with solution. Removed and dried, the precipitates spontaneously detonate.

Ion Exchange System. Standard explosive tests have been conducted on anion exchange resins by the Carney's Point Laboratory of the Du Pont Company. The data from closed bomb tests show that the dried nitrate form of anion resins is explosive and could be detonated in the falling hammer test. However, wet samples of resin did not react even at closed bomb test temperatures up to 200°C. Because the wet resin has significantly safer properties than dry materials, every effort is made to keep the resin bed wet at all times. The latent heat consumed by evaporating the interstitial water limits the temperature, and therefore the reaction rate, to low values where the decomposition is not hazardous.

Radiolysis of anion resin in gamma fields decreases the thermal stability of the resin. After irradiation at levels of 10^8 and 10^9 R, the dried samples were ignited at furnace temperatures of 128°C and 88°C, respectively. The unirradiated material did not ignite at a furnace temperature of 140°C. Safety considerations indicate that the total radiation dose received by anion exchange resins should be limited to 10^9 R. However, the loss of exchange capacity should limit the resin irradiation level to about 3×10^8 R. Since there is no process requirement which necessitates heating the acid form of the resin above 55°C, limitation of maximum temperature to 60°C covers both process considerations and safety aspects.

The safety of the anion exchange process for the recovery of neptunium, with respect to hydrazoic acid, was determined by analyzing actual process solutions that were most likely to contain the maximum concentration of HN_3 . The most susceptible point occurs when the partitioning solution containing ^{238}Pu is heat-killed to prepare feed for the subsequent anion exchange purification of ^{238}Pu . The analysis showed that all values of HN_3 were well below the conservative safe limits.

Waste Disposal System. All uncontrolled reactions in the waste disposal system that result in loss of material to the sumps occurred during resin digestion as a result of inadequate temperature control. In these incidents, material erupted from the vessels.

Although none have been observed, other incidents that could result in uncontrolled reactions include excessive caustic addition rate during neutralization or an accumulation of hydrogen from a plugged vessel vent system.

5.1.4.3 Criticality

Dissolver System. The fissile materials handled in the canyon facilities exist in solid form up to the time they are dissolved. Before dissolving, the fissile material is diluted with depleted uranium. Therefore, the probability of obtaining a critical configuration in the dissolving equipment is remote.

Criticality in the dissolver would require either processing of a ^{238}U -free fuel or a physical separation of uranium and plutonium, such as precipitation of plutonium from the solution. A nuclear safety hazard also would require selective precipitation and accumulation of Pu from two dissolving runs (cuts) combined with failure of the sampling and analysis procedures to detect the

absence of plutonium in the dissolver solution. Accidental addition of a precipitating chemical, such as NaOH, is prevented by administrative controls including the use of nuclear safety blanks in process lines from certain cold feed tanks.

In addition, a condition can exist whereby plutonium in an acid-deficient solution forms a polymeric hydroxide and presents a problem in succeeding operations. Close control of the dissolution end point is maintained to prevent the formation of an acid-deficient solution. Also, the dissolver solution is analyzed after each out is received and acid is added if the solution analyzes less than 1% nitric acid.

Head End System. Head end solutions, with one exception, contain large amounts of depleted uranium. Thus, as discussed above for dissolving, any sequence in which fissile materials become unsafe must include both separation of plutonium from uranium and concentration of plutonium. The conceivable mechanisms are repetitive plutonium precipitations or sorption from head end solution. These sequences are initiated by a chemical addition error involving a precipitant or sorbent added inadvertently.

The one exception to the situation above occurs in the tanks (10.2 and 10.3) that receive ^{239}Pu solution recovered from scrap in FB-Line Special Recovery. The sequence for concentrating these solutions involves: 1) a release, by leak or overflow, to the canyon sump, 2) slow evaporation in the sump, and 3) accumulation during continued process operations.

Solvent Extraction System. Nuclear safety in solvent extraction is based on concentration control; that is, the solvent extraction flowsheet is designed to maintain the concentration of fissile materials well below the critical concentration. This is done by limiting the feed concentration, the extractant concentration, and the ratio of feed to extractant. Other methods used to maintain safe concentration are: 1) rigid administrative control of the operation of the process; 2) frequency monitoring, sampling, and material balance checks; and 3) the use of neutron monitors. However, there exists the potential to accumulate fissile material in unsafe concentrations through operational error or unforeseen process conditions.

All centrifugal contactors, mixer-settlers, decanters, and surge tanks are areas of potential hazard. For the unlikely conditions of carefully balanced uranium saturation of the solvent phase in the Purex 1A bank, plutonium might reflux in the extraction section and present a hazard. The 1B and 2A banks in the Purex process (see Figure 3-6) present the greatest danger of the accumulation of a critical amount of fissile material because

- The equipment is not geometrically favorable.
- The continuous nature of the solvent extraction process affords no control of batch size.
- The concentration of fissile material in the individual stages of the banks is variable and sensitive to process changes.

In the Purex 1B bank, an excessive amount of nitrous acid in the 1AP or a deficiency of ferrous sulfamate in the 1BX will prevent the reduction of Pu(IV) to Pu(III), and plutonium will concentrate in the aqueous phase by refluxing. Unless this condition is corrected, the critical concentration will be approached in one of the stages in a few hours. The controls include careful adjustment of feed to the first cycle, close surveillance of the concentration of plutonium in the 1BP, and the use of neutron monitors on the 1B bank.

In the Purex 2A bank, the refluxing of plutonium is sensitive to concentration of acid and to the rate of flow of the scrub stream (2AS). The controls include careful adjustment of the acidity of the scrub stream, close account of the plutonium discharged by the cycle, and the use of neutron monitors on the 2A bank.

Although material balance checks and close adherence to a safe flowsheet are the primary controls, neutron monitors are also used. If they show an immediate change in the level of fissile material in the banks, a signal alerts the operators to the off-standard conditions and comes in time to allow corrective measures to be taken. The primary corrective measure is to shut the process down to determine the cause of the upset. If both neutron monitors fail, continued operation is not allowed (22). In addition, end streams are routinely sampled and analyzed to determine if abnormalities exist.

In general, the technique to recover from an upset is to restore the normal flowsheet conditions. However, the extent of the upset is determined before proceeding.

5.1.5 Low Energetic Events

A low energetic event is defined as an event that will not destroy the primary confinement barrier (the primary container), but activity may penetrate it. These events by themselves do not necessarily expose operating personnel to radioactivity. Low energetic events include transfer errors, overflows, leaks, chemical addition errors, pluggage, coil and tube failures, and waste header failures.

5.1.5.1 Transfer Error

Transfer errors are defined as the movement of a material to an unintended location, premature movement, or excessive movement where potential for chemical reaction is unlikely. These errors are significant operating mistakes from a risk viewpoint because of the high frequency of occurrence and the potential consequence of primary confinement of large volumes of liquid containing concentrated radioactive materials. A secondary effect is vessel overflow as discussed in Section 5.1.5.2. The direct consequence of this is usually small; however, the vulnerability is increased for a second transfer error during attempted recovery from the spill, loss of contaminated liquid through cracks and expansion joints, increased activity to the canyon ventilation system, and contamination of the canyon cranes with subsequent personnel exposure during cleanup. Unexpected transfer to these locations

could result in radiation exposure to personnel, contamination, overflow of vessels, or uncontrolled chemical or nuclear reactions. Quantities of radioactivity involved vary widely and are discussed in sections of this report concerned with the consequences from the systems involved. The specific effects on the facility receiving the material during a transfer error are discussed in the Systems Analysis for the particular facility with the exception of a transfer error to the 211-F outside facilities which is included in this analysis.

5.1.5.2 Overflow

Overflows occur when the volumetric capacity of a vessel is exceeded. The usual result is a loss of liquid to the canyon floor, to a sump, to a vent header, or to another vessel. The consequence outside of the canyon is less than for transfer errors because radiation or contamination as a direct result is generally avoided. The secondary effects, however, can be significant as a result of a breakdown in secondary confinement, or the increased chance of a transfer error during the cleanup process because of the frequent circuitous routes to return the liquid to a vessel.

5.1.5.3 Leak

Leaks permit material to penetrate primary confinements such as vessels, pipes, and instruments. The consequences of these leaks range from the loss of a few milliliters of solution to all of the material in holdup. Leaks can also occur due to failure of valve packings, welds, or instruments.

5.1.5.4 Chemical Addition Error

Chemical addition errors usually involve the transfer of an incorrect quantity or unknown material into a known vessel, or the addition of an undesired quantity of material. The greatest hazard of the chemical addition error is the initiation of an uncontrolled chemical or nuclear reaction as discussed in Section 5.1.4.2. Other consequences are associated with product contamination or material loss.

5.1.5.5 Pluggage

The usual effect of line pluggage is simply to retard the passage for a fluid or a hydraulic signal through the system. The most common form of pluggage is that of instrument line pluggage. Liquid level and specific gravity instrumentation is particularly vulnerable. When this occurs in process vessels, it is usually necessary to shut the system down long enough to blow the pluggage from the lines using compressed air or steam.

Sampler pluggage also occurs. Aside from the delay in analysis of vessel contents that results, personnel can be exposed to significant radiation fields while the pluggage is removed. Blow-down, needle replacement, and acid dissolution are commonly used to restore sampler operation.

Jet pluggage and transfer line pluggage result in processing delays as well as personnel exposure to remove the blockage.

Other miscellaneous pluggage problems include control valves, filters, iodine reactors, air lines, weirs, and cooling water lines.

5.1.5.6 Siphoning

Siphoning involves the transfer of material from one vessel to another through primary piping due to pressure differential. The unit operations in which siphoning is likely to occur are dissolving, head end, solvent extraction, evaporation, and waste disposal.

5.1.5.7 Processing Short-Cooled Fuels

Accidental processing of short-cooled fuels results in a marked increase of release of radioiodine from the dissolving process (2). During the past 30-yr combined operation of F- and H-Canyon, one shipment of 77-day-cooled fuel was processed in F-Area. Shipment of short-cooled fuel is beyond the control of either separations area, however.

5.1.5.8 Ruthenium Volatilization

Volatile ruthenium is released primarily from solutions or dissolving processes where there is a strongly oxidizing atmosphere. Four cases in F-Canyon occurred during the past twenty years that are considered significant. Three of these were associated with dissolving operations and one during crane decontamination.

Mechanisms identified as potentially contributing to abnormal releases that have not been previously defined and analyzed are processing of short-cooled fuels and excessive temperature. Because of the effectiveness of the vessel vent system and the sand filter, these occurrences would not be expected to appear as well defined spikes of increased activity release. The activity would be confined; and significant releases that might appear later would be, in almost every case, the result of some perturbation to the ventilation system. Such releases are reported as annual routine releases.

5.1.5.9 Iodine Reactor Failure

The most common form of failure in an iodine reactor is pluggage after prolonged usage. This is reflected in reduced vacuum in the dissolver, but not in a measurable reduction in iodine removal efficiency. Failure can result in increased releases.

If the silver reactor temperature is too low iodine is inefficiently absorbed, and if the temperature is too high ruthenium, which has plated out at a lower temperature on the system piping, is re-volatilized because of the higher vapor line temperature that is caused by the high temperature in the silver reactor.

5.1.5.10 Coil and Tube Failure

Leaking coils or tubes that either heat or cool canyon vessels provide a route for radioactive material to bypass the primary liquid confinement. The consequences of these failures are mitigated by the pressure maintained in the coils and tubes during both operating and shutdown periods, by monitoring devices on steam condensate, cooling water return streams, and by diversion and retention basins.

5.1.5.11 Waste Header Failure

Material can bypass the primary liquid confinement via waste header failures. Loss of material from the headers was examined in four places: at the waste farm, between the canyon and the waste farm, inside the canyon, and inside the air tunnel.

5.1.6 Residual Release Events

A residual release event is similar to the low energetic event except that it considers only the residual activity outside the process equipment as being available for transfer through the confinement barriers. These releases are primarily due to suckback and air reversals.

5.1.6.1 Suckback

Suckbacks are caused by the cooling of condensable vapors typically present in instrument air lines associated with the gang valves. The partial vacuum that results draws highly radioactive liquids into these lines. The usual result of a suckback is a localized, but possibly intense, radiation field in one of the personnel areas. The radioactive materials are usually contained; thus, the most significant potential consequence is the exposure of the operating personnel to penetrating radiation.

5.1.6.2 Air Reversal

Protection against airborne activity releases to occupied areas is provided by the multiple-air-zone concept used in the ventilation design of many SRP facilities. Under this concept, ventilation air flows from occupied areas through areas of increasing potential for airborne activity. The barriers between zones are doors, airlocks, walls, or glove cabinets. Pressure differentials are maintained between zones by controlling the air exhaust and air supply rates. An air reversal occurs when the pressure differentials are lost so that air can flow in a reversed direction. This situation could potentially be caused by pressure surges (explosion, fire, etc.) in a process area and has been caused by malfunctions of the ventilation system. Often barrier failures or open doors between zones are a contributing factor.

5.1.7 Chemical Hazards

There are no identifiable hazards associated with cold feed preparation and storage so long as the chemicals remain within the confines of the intended vessels and piping. Unconfined, the hazards are toxicity, corrosion, fire, and carcinogenicity. Cold feed operations were analyzed for the mechanisms of chemical releases from confinement, for hazards to operating personnel, and for potential releases outside the 221-F Building.

Methodology used in this analysis and the hazards for various chemicals are discussed in detail in Reference 23.

Chemicals handled and stored in the Third Level Cold Feed Preparation (CFP) and Storage Area are listed in Table 5-2. Chemicals found only in process off-gases (e.g., H_2 and HN_3) are not analyzed; chemical fire hazards were considered earlier (Section 5.1.4.1). Although all metal nitrates (e.g., Al, Mg, and Na) are somewhat toxic and corrosive under some conditions, their potential hazard is considered to be relatively small; consequently, only uranyl nitrate was analyzed. The list of eight hazardous chemicals considered in this analysis is given in Table 5-3. Four of these chemicals are nonvolatile salts, three are potentially volatile nitrogen compounds, and one (NO_2) is the product of chemical reactions of other chemicals. They present an insidious hazard because their odor thresholds provide a poor warning of hazardous airborne concentrations and because some of them have little color.

5.1.8 Support Systems

Safety related occurrences discussed in Sections 5.1.3 - 5.1.6 also apply to ancillary operations and equipment. These occurrences are analyzed because in some cases a deteriorating condition in an ancillary facility can have a significant effect on several primary operations but will not result in a direct release of material outside the canyon. Except for the decontamination facility, the impact of the support system failures are considered with the analysis of the individual primary operations; they were not considered as separate initiating events.

5.1.8.1 Shops and Decontamination Facilities

The primary potential hazards associated with shops and decontamination facilities are loss of control of decontamination solutions, radiation exposure to personnel, and spread of airborne activity.

Loss of control of contaminated solutions occurs in the same manner as in the primary process systems, i.e., by transfer error, overflow, leaks, and coil failures. These initiators are discussed in Sections 5.1.4 and 5.1.5. Miscellaneous pathways for escape of liquids include cracking of stainless steel cell or sump liners, broken viewing windows, and leakage around service entries to the cells. Occasionally, pieces of equipment cannot be adequately drained or decontaminated prior to shipment to the burial ground resulting in transfer of activity.

TABLE 5-2. Chemicals in Cold Feed Preparation and Storage

Name	Storage Solution Concentration, %	Potential Hazard
Aluminum nitrate	34	Toxic
Ferrous sulfamate	40	Toxic NO ₂ fumes
Gelatin	1	--
Hydrazine mononitrate	30	Toxic, carcinogen
Hydroxylamine nitrate	0.53	Toxic, carcinogen
Ion exchange resin	--	Fire
n-Paraffin	100	Fire
Manganous nitrate	50	Toxic
Nitric acid	64	Toxic, corrosive
Potassium permanganate	6	Toxic, corrosive
Sodium carbonate	2.5	--
Sodium hydroxide	50	Toxic, corrosive
Sodium nitrate	40	Toxic
Sodium nitrite	30	Toxic, carcinogen**
Tributyl phosphate	100	--
Uranyl nitrate	***	Toxic, corrosive

*Lower concentrations may also be handled.

**Suspect carcinogen.

***As supplied from A-Line, normally ~425 g/l.

TABLE 5-3. Hazardous Chemicals

Name	AEI, * mg/m ³	Odor Threshold, mg/m ³
Hydrazine mononitrate	0.13	4
Hydroxylamine nitrate	0.1	--
Nitric acid	5	9
Nitrogen dioxide	9	9
Potassium permanganate	5	None
Sodium hydroxide	2	None
Sodium nitrite	0.1	None
Uranyl nitrate	0.005	None

*AEI = acceptable exposure limit, milligrams per cubic meter of air,
for 8-hr exposure.

Overheating of decontamination solutions is the primary cause of airborne activity to personnel areas in shops and decontamination facilities. When overheated, the contaminated vapors tend to travel into the railroad tunnel because the normal air flow pattern is perturbed by the thermal updraft.

5.1.8.2 Cell Covers

Cell covers have a potential for inflicting significant damage to process equipment if the covers are dropped during movement. Dropping can be caused by one of the bails in the cover breaking, the crane cable breaking, or the cover disengaging from the crane hook. In all cases of record in which the cover was dropped, the cover itself was damaged.

In other cases, cell cover damage has occurred in which the direct potential for damage to other equipment was small. Some deterioration to the covers has occurred from chemical attack but not to an extent sufficient to affect the integrity. Damage has been inflicted to the dowel pins used to center the covers over the canyon cells; and, in one case, the concrete support structure beneath the cover collapsed. Damage has also occurred from dropping equipment onto a cover. In many cases, support concrete pedestals have crumbled and have been replaced with stainless steel.

Mishandling of cell covers has caused some damage to equipment in which dropping did not occur. This is most evident when the covers are placed on top of equipment being stored on other cell covers.

Cell covers become highly contaminated during use. Thus, storage and disposal of used covers can present radiation exposure problems because of the limited storage space in the canyon and the difficulty in decontaminating the surface.

Care must be exercised in removing covers from process cells. If too many covers are removed, an air reversal will result as discussed in Section 5.1.6.2.

5.1.8.3 Gang Valve Corridors

Difficulties in the gang valve corridors were analyzed with regard to equipment malfunction and to potential hazards to operating personnel. Specific items include: gang valve failures, other valve failures, corrosion, fire, leaks, surface contamination, airborne radioactivity, and radiation.

Gang valves fail in a number of ways, including: bellows and seat failures, electrical motor failure, loss of remote operation function, timer failure, valve lever sticks in one position, valve lever will not travel into a position, steam trap failure, limit switch failure, pluggage, and valve corrosion.

5.1.8.4 Cranes

Functioning in the canyon cranes is essential for operation of the canyons. Failure or impairment of a crane or its equipment essentially stops all cask

unloading, fuel charging, equipment changes, and maintenance. Effectiveness of the cranes was analyzed, therefore, from three points of view: impairment of crane functions, damage to other equipment, and hazards to operating personnel.

Items that can impair operability of the crane include: optics, hoists, limit switches, failure of wheels, bearings, brakes, cables, communications, remote tools, electrical systems, and lights.

Damage may be inflicted on other equipment by crane misoperations such as dropping of loads, handling errors, and other operating errors.

Damage to crane ancillaries includes spalling of concrete beneath the crane rails; acid fumes causing corrosion to runway emergency doors, shielding doors, and removable rails; blown fuses; sticking door; lost keys; timer switches; and bent lifting yokes.

Hazards to crane operating personnel are radiation, contamination, and personal injury. Radiation exposure primarily occurs during maintenance to crane components. This exposure is controlled in part by decontamination prior to maintenance, shielding, and limiting work intervals.

Airborne contamination in the crane cabs is caused by leaking equipment in the canyons, inadequate ventilation flow through the canyons, failure of the crane air conditioner, holes in air conditioner ducts, improperly seated HEPA filters, and fire in the canyon. The warm crane has experienced a number of problems in recent years with improperly seated HEPA filters. These deficiencies are detected by DOP testing after installation and corrections are made prior to crane use. Surface contamination within the crane cabs is caused primarily by leaks in the air conditioner, gear box, and hydraulic system.

Personal injury has occurred about once per year according to data bank records. Generally, these occur during maintenance work.

5.1.8.5 Sample Aisles

Difficulties in the sample aisles were analyzed with regard to equipment malfunction and to potential hazards to operating personnel. Specific items in the first category include: inability to sample (especially pluggage), hoist failures, elevator outages, and instrument malfunction. Items in the second category are: surface and airborne contamination, radiation, fire, fumes, and injury.

5.1.8.6 Steam Distribution

The steam distribution system was analyzed to determine the mechanisms and frequencies of disabling all or portions of the steam supply, and for radiation or contamination potential associated with steam. Four sections of the steam system were examined; the powerhouse, the 325 psi header, the building supply header, and the condensate headers.

5.1.8.7 Compressed Air

Air compressor failure can occur in any of four independent systems within the canyons: plant air, instrument air, process air, and breathing air.

5.1.8.8 Electrical Distribution

The electrical distribution system was analyzed to determine the expected frequency of failure to supply power to a piece of vital equipment and the more vulnerable parts of the system.

5.1.8.9 Cooling Water Distribution

The cooling water distribution system was analyzed to determine the expected frequency of failure to supply cooling water to canyon vessels and the more vulnerable parts of the system. Nine subsystems or events were identified as being significant with regard to failure of the cooling water system. Other highly unlikely occurrences, such as a plane crash or earthquake, could disable the water supply; however, these are tacitly considered in the subsystem analyses, and are not considered independently.

5.1.9 Engineered Safety Features

Engineered safety features provide the secondary confinement for hazardous materials and include the ventilation systems, coil and tube pressure regulators, neutron monitors, diversion systems for water return, seepage basins, and secondary liquid confinement. Safety related occurrences discussed in Sections 5.1.3 to 5.1.6 also apply to engineered safety features. These occurrences are analyzed because failures within these systems can result in failures of secondary confinement which lead to materials being released outside the canyon.

5.1.9.1 Ventilation System

The ventilation system includes the sand filter (Building 294-F), the fan house (Building 292-F), and the 200-ft tall stack (Building 291-F). It was analyzed to determine expected frequencies and consequences of releases of radioactive materials to the atmosphere via the stack and to the operating areas as a result of upset to the ventilation system. The main contributors from canyon facilities to the activity in the stack effluent are the canyon air and the process vessel vent system that collects off-gases from process solutions. The radioactive content of the input air to the sand filter comes about equally from these two ventilation systems. Releases to operating areas result primarily when the normal direction of air flow within the building reverses. Air reversals were discussed in Section 5.1.6.2.

A definitive study of activity releases from process systems is difficult due to the multiplicity of contributors and the simultaneous operation of process

equipment. Activity released from the stacks has varied widely as methods of operation have changed.

A number of operational practices are known to elevate the airborne radioactivity release rate from certain process vessels in the canyons. These releases to the process vessel vent system are filtered prior to release to the environment. Some releases are scrubbed prior to filtering. However, none of these treatment systems removes all the radioactivity. When large curie quantities of radioactivity are involved, the amount of activity released can be significant. The radioactivity normally in the input air to the various filters results from entrainment of process solutions while sparging, neutralizing acidic solutions, transferring by steam jet, or evaporation. Additionally, ruthenium and iodine may be evolved in varying amounts as chemical conditions change, e.g., ruthenium tetroxide may be evolved in large quantities if excess permanganate is added to give strongly oxidizing conditions. Process deviation and equipment failures also have given above-normal releases. In the early years of operation, fuel cooling times after irradiation were short, and radioactive iodine and other short-lived isotopes were released in substantial quantities during chemical processing. Over the years, releases have been significantly reduced by requiring longer fuel cooling times.

During 1975, large quantities (~2000 lb) of ammonium nitrate were detected in the vessel vent systems of the canyon buildings. The presence of ammonium nitrate is a potential explosive hazard in the vent system. The hazard of ammonium nitrate arises from its exothermic decomposition to produce gaseous products, which under certain circumstances can lead to an explosion.

Explosions of ammonium nitrate can be generated by impact or fires or other exothermic reactions that generate into an explosion. Ammonium nitrate is not very sensitive to impact; a large shock wave of 30 kilobars is required to initiate a detonation. It is unlikely that any impact occurring in the canyon building (such as dropping a cell cover, etc.) could ever cause ammonium nitrate to explode. A fire could turn into an explosion if large amounts (>>100 lb) of ammonium nitrate are present in confined places. Impure ammonium nitrate is more reactive because of the presence of small amounts of chloride ion, nitric acid, and heavy metals that catalyze the decomposition. In addition, if organic matter is present, considerably more energy is evolved during the decomposition.

In view of the high sensitivity of ammonium nitrate containing impurities and the fact that impurities are probably present in the vessel vent filter, it seems likely that fire involving 1000 lb or more of ammonium nitrate would turn into an explosion. Furthermore, the material in the filter is confined, increasingly the chance of an explosion. However, under normal operating conditions, air at 50 to 80°C passes through the filter at about 2900 cfm and, as it traverses the filter, it picks up heat from the decomposing ammonium nitrate and removes it from the filter. These temperatures are considerably lower than those (>170°C) that would lead to thermal excursions with large quantities of ammonium nitrate present. If the air flow to the filter were shut off, the ammonium nitrate would begin to self-heat; but with the amounts of ammonium nitrate that would likely be present in the dry form, wall temperatures in excess of 70°C would be required to lead to an excursion. Therefore, the system is considered to be safe against self-heating during

normal operating conditions and during times of no air flow, provided the canyon ambient temperature does not exceed 70°C.

5.1.9.2 Water Return and Diversion

The water return and diversion systems were analyzed to determine the mechanisms for movement of radioactive materials into streams leading offplant and to undesired intermediate locations within the separations areas. Assuming that activity enters the water streams, seven mechanisms have been identified for loss of this activity into the plant stream. Each is discussed below.

The primary cause of activity in the segregated water is coil or reboiler failure during operation of a canyon vessel. (Note that less than 1/3 of the coil failures described in Section 5.1.5.10 result in releases to the segregated water system.) Other potential causes include residual activity becoming dislodged, the Cash pressure system being valved off during shutdown of a vessel with a failed coil, heat exchanger leaks, and unspecified causes. Multiple detections and diversions, however, may result from a single failure.

Activity can reach either Upper Three Runs or Four Mile Creek (see Figure 2-1) from a combination of the following events: activity in either the segregated or circulated water systems, a small breach in the effluent piping, and sufficient rain to result in runoff.

The third mechanism is similar to the second except it is assumed that a major breach to the effluent piping occurs and that rain is unnecessary. Real or potential causes include minor pipe break coupled with ground settling, manhole deterioration, overflow of manhole, bypass of delaying basin, delaying basin overflow, valving error in Building 211-F, or pipe crushed by heavy equipment.

Activity reaching the cooling tower at the same time the cooling tower excess is overflowing to Four Mile Creek is another mechanism for release. Activity can reach the cooling towers by three mechanisms:

- Activity in circulated cooling water and failure to divert the stream to the retention basin
- Activity being pumped from the waste farm because of a significant heat exchanger leak
- A leak in the isolation gate valve coupled with activity in the circulated cooling water system.

The fifth mechanism is the improper removal of contaminated sludge from the bottom of a delay basin. This fault has occurred once in 200-Area operations with a consequence of about one curie of activity being released to the effluent ditch.

If activity sorbs in the soil of the effluent ditch and is not removed, it may eventually erode to the creek. One such occurrence is on record.

The final mechanism, although ~~not~~ directly related to canyon operations, can be the result of malfunctions within the canyon. Overflow of lined basin, seepage because of damage to the liner from solar radiation, draining valve error, or pump-out error can occur allowing activity to reach the system.

5.1.9.3 Seepage Basins

The F-Area seepage basin system consists of three earthen basins with a total capacity of 14 million gal. These basins provide a means for reducing offsite releases by filtration, soil sorption, and radioactive decay.

The condensate from the various evaporators is a major contributor to the volume of water going to the seepage basins. The largest volumes come from the nitric acid recovery unit and the waste storage system evaporator. Lesser contributors are sumps and drains from many operating areas that contain only low levels of activity.

There is a continuous sampler on the composite water stream flowing to the seepage basin. Samples are analyzed daily, and more complete analyses are obtained on weekly composite samples. Selected annual releases were published in Reference 24, and a complete analysis of the seepage basins is published in Reference 25.

5.1.9.4 Secondary Confinement for Liquids

Secondary confinement for liquid materials within the canyon building is provided by the walls, floor, and sumps of the canyon. Secondary confinement for airborne materials is provided by the engineered safety features of the ventilation system. An analysis of the ventilation system is presented in Section 5.1.9.1. This analysis is concerned with the escape of liquid materials from the canyons into areas normally occupied by personnel, or outside of the building.

Historically, the most frequent path penetrating the secondary confinement is through expansion joints between each canyon section. Under most circumstances, liquids lost from the primary piping system simply drain into a sump and are returned to the process. Sumps are equipped with liquid-level detectors. If the sump floods to a level higher than the curbing separating the canyon sections, liquid seeps into the joints and frequently reappears in a personnel area. Liquid may also get into the joints as a result of a gasket or pipe leak directly over the joint, liquid being spilled onto the joint, or liquid splattering off equipment or walls onto the joint. The condition is aggravated by deteriorated packing, much of which cannot be replaced beyond a few inches from the surface.

The second most frequent path for activity penetration is around or through embedded piping. The frequency of the original pipe failing or of liners failing increased to a high of about 40 in 1976, but has decreased dramatically.

Other causes of secondary confinement penetration to personnel areas have included seepage through cracks other than expansion joints, drainage through test holes in concrete, and drilling through concrete into process piping.

No significant damage to the secondary confinement as a result of an earthquake would be expected at less than MM VIII (24). Should damage to piping and vessels occur alone with shifting at expansion joints, process liquids initially would be expected to drain to the lower levels of the building. No mechanisms have been identified for transporting process liquids outside of the building in such a way that surface runoff would occur.

5.1.9.5 Pressure Regulators

Coil and tube leaks in canyon vessels are mitigated by 1) the pressure maintained in the coils and tubes during both operating and shutdown periods, 2) by monitoring devices on the segregated and circulated water systems, and 3) by use of delaying and retention basins. The engineered safety feature of principal interest is the coil and tube pressure regulating system, known as the Cash Regulator System.

In most cases, these leaks release no detectable activity to these water systems because the coils or tubes are pressurized at all times so that the direction of any leakage is into the vessel. Such leaks are detected by increasing liquid levels in the vessel or by decreasing specific gravity in the solution; the potential for release can then be reduced by moving the solution to another vessel.

5.1.9.6 Neutron Monitors

Neutron monitors are positioned in the canyon to provide early warning of increasing concentrations of fissile materials in the mixer-settlers used in solvent extraction operations. These monitors (and other process instrumentation) alert the control room operators to the reduced margin for nuclear safety so that prompt corrective measures can be taken.

5.2 ANALYSIS METHODOLOGY

The purpose of the accident analysis is to determine and evaluate the risk of operating the F-Canyon facilities. The overall approach was to review the facility and equipment design and the methods for operating and controlling the process. The knowledge obtained from this review was then systematically analyzed to predict the exposed frequency of events. Consequences of events were calculated from available data or estimated where necessary.

Risk in this analysis is defined as the product of the expected frequency of an event that will release radioactivity and the radiological consequence. The characterization of the radioactivity releases were defined by dividing the types of initiating events leading to these releases according to their energetics and the impact of the released energy on the confinement barriers. Event trees were developed to define the sequences of events subsequent to the initiating event. These event trees provided a basis for analyzing the

physical processes occurring during operations and for determining confinement failure modes. These sequence definitions then permitted estimates of the amount of radioactivity that would be released from the occurrence.

To obtain the expected frequency for a given event consequence, it was necessary to determine the expected frequency of the initiating event and the failure rate per demand for the sequenced conditions. The determination of frequencies and consequences was based largely on the extensive data base of incidents that have occurred in 200-Area operations. For those events where data were not available, event tree analysis was used supplemented by fault tree analysis.

5.2.1 Data Base

5.2.1.1 Information Sources

Initiators were systematically identified by a search of the Incidents Data Bank (6), the 200-Area Fault Tree Data Bank (2,7), and by judgment of experienced analysts. The 200-Area Data Bank includes data from all 200-Area operations. The data entries range from minor equipment malfunctions to incidents with potential for injury or contamination of personnel. Several types of information have been extracted from these reports, including actual incidents, potential incidents, consequences, and event frequencies.

5.2.1.2 Data Storage and Retrieval

Failure data for Savannah River operations are stored in several data banks in a manner suitable for sorting and retrieval of information for risk assessment. The data banks include a generic incident data bank (6) that contains known potential incidents that could occur in each of the unit operations associated with fuel reprocessing.

The 200-Area Fault Tree Data Bank (2) contains listings for actual deviations from normal operation, including dates of occurrence. These incidents are coded by site location, facility, unit operation, and keyword so that they can be retrieved by a variety of specifications. A computer code (7) provides the capability of selecting data by any of nine separate specifications. These specifications are: area, facility, unit operation, key word, "and" logic, "not logic", source document, date and text words. From these data, component failure frequencies and repair times, are calculated. A review of these data also provided information on initiating events.

5.2.2 Radionuclide Distribution

Radionuclide releases are reported in terms of curies of activity released to the environment. To calculate dose to man resulting from transport of this material, the individual radionuclides present and their amount must be known. Table 5-4 shows the radionuclide distributions for the major process systems. These distributions are the basis for calculating distributions for materials released to the environment.

TABLE 5-4. Mix of Radionuclides for Various Process Solutions

Nuclide	Isotopic Curie Fraction						
	Raw Metal Solution	Head End	First Cycle	Second U Cycle	Second Pu Cycle	Waste	Ion Exchange
89Sr	0.777E-01	0.7800E-01	0.777E-01	0.335E-01	0.103E-02	0.784E-01	0.784E-01
90Sr	0.744E-02	0.7400E-02	0.744E-02	0.321E-02	0.985E-04	0.751E-02	0.751E-02
90Y	0.744E-02	0.7400E-02	0.744E-02	0.321E-02	0.985E-04	0.751E-02	0.751E-02
91Y	0.119E-00	0.1200E+00	0.119E-00	0.516E-01	0.158E-02	0.121E-00	0.121E-00
95Zr	0.119E-00	0.1200E+00	0.119E-00	0.257E-00	0.158E-01	0.120E-00	0.120E-00
95Nb	0.319E-01	0.3200E-01	0.319E-01	0.689E-01	0.105E-02	0.322E-01	0.322E-01
103Ru	0.586E-01	0.5900E-01	0.586E-01	0.759E-01	0.856E-03	0.592E-01	0.592E-01
106Ru	0.600E-01	0.6000E-01	0.600E-01	0.777E-01	0.868E-03	0.606E-01	0.606E-01
106Rh	0.600E-01	0.6000E-01	0.600E-01	0.259E-01	0.794E-03	0.606E-01	0.606E-01
110Ag	0.758E-03	0.7600E-03	0.758E-03	0.327E-03	0.100E-04	0.765E-03	0.765E-03
123Sn	0.102E-02	0.1000E-02	0.102E-02	0.442E-03	0.135E-04	0.103E-02	0.103E-02
125Sb	0.128E-02	0.1300E-02	0.128E-02	0.554E-03	0.170E-04	0.130E-02	0.130E-02
127Te	0.194E-02	0.1900E-02	0.194E-02	0.839E-03	0.257E-04	0.196E-02	0.196E-02
129Te	0.122E-02	0.1200E-02	0.122E-02	0.526E-03	0.161E-04	0.123E-02	0.123E-02
134Cs	0.130E-02	0.1300E-02	0.130E-02	0.560E-03	0.172E-04	0.131E-02	0.131E-02
137Cs	0.995E-02	0.9900E-02	0.995E-02	0.430E-02	0.132E-03	0.100E-01	0.100E-01
141Ce	0.385E-01	0.3800E-01	0.385E-01	0.166E-01	0.509E-03	0.388E-01	0.388E-01
144Ce	0.179E-00	0.1800E+00	0.179E-00	0.773E-01	0.237E-02	0.181E-00	0.181E-00
144Pr	0.179E-00	0.1800E+00	0.179E-00	0.773E-01	0.237E-02	0.181E-00	0.181E-00
147Pm	0.321E-01	0.3200E-01	0.321E-01	0.124E-01	0.425E-03	0.324E-01	0.324E-01
148Pm	0.374E-03	0.3700E-03	0.374E-03	0.162E-03	0.542E-05	0.378E-03	0.378E-03
155Eu	0.563E-03	0.5600E-03	0.563E-03	0.243E-03	0.745E-05	0.568E-03	0.568E-03
234U	0.148E-08	0.1500E-08	0.148E-08	0.639E-05	0.196E-11	0.141E-12	0.141E-12
235U	0.237E-07	0.2400E-07	0.237E-07	0.102E-02	0.314E-10	0.235E-11	0.235E-11
236U	0.214E-07	0.2100E-07	0.214E-07	0.104E-02	0.314E-10	0.235E-11	0.235E-11
238U	0.219E-05	0.2200E-05	0.219E-05	0.946E-00	0.290E-08	0.235E-09	0.235E-09
237Np	0.315E-08	0.3100E-08	0.315E-08	0.136E-08	0.417E-10	0.640E-10	0.318E-08
238Pu	0.149E-05	0.1500E-05	0.149E-05	0.193E-04	0.197E-03	0.940E-10	0.415E-08
239Pu	0.448E-03	0.4500E-03	0.448E-03	0.580E-02	0.594E-01	0.270E-07	0.136E-05
240Pu	0.102E-03	0.1000E-03	0.102E-03	0.132E-02	0.135E-01	0.620E-08	0.308E-06
241Pu	0.679E-02	0.6800E-02	0.679E-02	0.880E-01	0.899E-00	0.410E-06	0.206E-04
242Pu	0.981E-08	0.9900E-08	0.991E-08	0.128E-06	0.131E-05	0.600E-12	0.300E-10

5.2.3 Event Trees

The principal means for identifying significant accident sequences uses event tree methodology. An event tree is a logic diagram for identifying the possible outcomes of a given initiating event. The number of possible final outcomes depends upon the release pathways that are applicable following an initiating event. In this analysis, the initiating event for each tree was chosen based on its energetics and physical form (i.e., liquid or airborne). Events subsequent to the initiating event are determined by system characteristics and engineering data. In these event trees, a particular sequence from the initiating event to final outcome is termed an accident sequence.

Release sequences were based on the physical confinement equipment (sometimes referred to as confinement barriers). When dealing with airborne materials, the ventilation equipment became the significant confinement barriers. Event trees provide combined accident sequences from the initiating event to the release of activity. Release points for liquids are releases to the water systems that may ultimately reach the outside ground or may be a source of an airborne release. Release points for airborne materials include releases to the canyon, ventilation system, and stack.

The starting point for the development of an event tree is the event that initiates a potential accident situation. These initiating events were derived in Section 5.1. Confinement barriers are discussed in detail in Section 3.3.3.

In the preparation of an event tree, the first step is to determine which systems are designed to mitigate the event. As an example (Figure 5-1), given the initiating low energetic event, the systems that affect the subsequent course of events are the vessel, the ventilation system, and the sand filter. Each of these barriers is ordered in its sequence across the top of Figure 5-1. The upper branch of the tree represents failure of the system to fulfill its confinement function. In the absence of other constraints, there are $2^{(n-1)}$ accident sequences, where n is the number of headings (functions or systems) included on the tree. However, there are known relationships (constraints) between system functions. For example, if the vessel contains the solution, then no consequence will occur. Once these relationships are determined, some of the sequences can be eliminated because they represent success paths. These reduced event trees are used in the analysis.

If the event sequences are independent, then the expected frequency of occurrence of a given sequence is the product of the initiating event frequency and the individual demand probabilities of the individual systems in that sequence. Because the failure demand probabilities are almost always 0.1 or less, it is common practice to approximate success $(1-p)$ as 1. It should be noted that, as indicated in Figure 5-1, the study developed event trees in which each branch point provides only two options: system failure or success. No consideration is given to the fact that partial system success may occur within an accident sequence. Thus, an accident sequence is conservatively assumed to lead to the total release. The effects of partial system failure are accounted for by adjusting the release (in curies) to compensate for partial failure.

FIGURE 5-1. Example Event Tree for Airborne Releases

5.2.4 Fault Trees

The fault tree method was used to estimate expected frequencies for events for which no data was available. The method uses a logic that is essentially the reverse of that used in event trees. Given a particular failure, the fault tree method is used to identify the various combinations and sequences of other failures that lead to the given failure.

The fault tree method, illustrated in Figure 5-2, shows a fault tree for a low energetic liquid and/or finely divided solid release event. The fault trees developed in the F-Canyon Systems Analysis (1) and utilized in this study were developed downward only to the point where either data bank failure numbers (Section 5.2.2) or engineering estimates could be made. In developing the trees, consideration was given to intrinsic component failures, human factors, testing, and maintenance. The expected frequencies of the failures were assigned to the appropriate elements of the tree, and the expected frequencies of the top event were calculated.

5.2.5 Dose Models

Significant onplant and offplant doses from the operation of F-Canyon are considered in this analysis. A description of the dose models used is included in the following sections.

5.2.5.1 Models for Calculating Doses from Atmospheric Releases for Accidents

The radiological consequences of radionuclides released from F-Canyon during an accident were evaluated for three population groups: 1) population dose to offsite people residing within 50 miles, 2) population dose to onsite personnel, and 3) dose to the offsite maximum individual at the plant boundary.

For each accident scenario postulated, the radiological consequences were analyzed using the AXAIR code. The AXAIR code performs both environmental transport and radiation dosimetry calculations for the postulated accidents involving airborne releases. The environmental transfer models used are based on the U.S. Nuclear Regulatory Commission Regulatory Guide 1.145 (26). The exposure pathways considered in the AXAIR code include inhalation of radionuclides and gamma-irradiation from the radioactive plume (27).

Radiation doses from inhalation of radionuclides in air depends on the amount of radionuclides released, the gaseous dispersion factor, the physical, chemical, and radiological nature of the radionuclides, and various biological parameters such as breathing. Standard sets of breathing rates for different age groups are used in the AXAIR code (27).

Radiation doses from gamma shine result from radiation emanating from the traveling plume and depends on the spatial distribution of the radionuclides, the energy of the radiation, and the extent of shielding. In the AXAIR code, no shielding was assumed in calculating population doses. The gamma-shine

FIGURE 5-2. Example Fault Tree for Low Energetic Event Release of Liquid

doses were calculated using a non-uniform Gaussian model because of its more realistic modeling compared to the conventional uniform semi-infinite plume model (28).

In addition to the use of the worst sector, 99.5 percentile meteorology, and conservative breathing rates and shielding factors, the AXAIR code also provides no credit for the probable plume rise from stack releases. Therefore, the doses calculated by the AXAIR code should be considered as an upper bound of radiological consequences for the atmospheric releases postulated. (For population dose calculations, meteorological conditions are weighted by sector, wind speed and dispersion class based on actual Savannah River site meteorological data.)

5.2.5.2 Models for Calculating Doses from Liquid Releases

The environmental effects of radioactive materials released in liquid effluents are evaluated for a variety of pathways: 1) drinking water, 2) aquatic foods consumption, 3) recreational uses of bodies of water, and 4) irrigation of food crops.

A review of water utilization from the Savannah River downstream of the Savannah River Plant site revealed that there is no known use of river water for irrigation; therefore, the irrigation of food crops was not considered as a pathway of radiation exposure. The relative importance of other pathways generally depends on the radionuclides of concern. However, for SRP liquid releases to the Savannah River, the exposure pathways of consumption of drinking water and aquatic foods are deemed the most significant. There are no available models that will characterize the uptake and retention of radionuclides by aquatic foods (fish, invertebrates, etc.) under transient conditions or over short-term periods as experienced during accidental releases. However, rather than omit the aquatic foods pathway, doses are calculated for steady-state conditions, recognizing that such doses are conservative and overestimated.

The method of calculating dose-to-man from liquid effluent pathways is that recommended in the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.109 (29). The LADTAP II, developed by the NRC and Oak Ridge National Laboratory (ORNL), is a computer code to perform dose calculations using the dose models specified in Regulatory Guide 1.109. Although LADTAP was specifically developed for routine releases, its use for postulated accidental releases is appropriate because it provides conservative overestimates. The 50-yr age-specific dose commitment factors specified in Regulatory Guide 1.109, with minor correction by the original author, were used (30).

5.3 ACCIDENT FREQUENCIES

The expected frequencies for accidents leading to release of radioactivity were developed in the F-Canyon System Analysis (1) and are summarized in this section. The event trees for postulated release sequences are developed in Appendix B. These release sequences include the expected frequency of the release initiator, along with failure of the filtration system and secondary confinement. Frequencies for exhaust system failure and criticality were

calculated by use of fault tree analysis. These fault trees are shown in Appendices B and C of the F-Canyon Systems Analysis (1).

5.3.1 Natural Phenomena

The frequency of occurrence for each of the natural phenomena considered in the analysis is discussed in the following sections.

5.3.1.1 Winds

Straight Winds. Straight winds with speeds approaching 175 mph are estimated to occur onplant at a frequency of less than $1 \times 10^{-6}/\text{yr}$ ($1.1 \times 10^{-10}/\text{hr}$; Reference 13). Straight winds greater than 175 mph have not been observed.

Tornado. The frequency of a tornado strike on a given location (building) depends on the size of the building and its orientation relative to prevailing tornado direction as well as by the tornado damage area (13). For F-Canyon, the probability of a damaging tornado strike is $<10^{-8}$ year ($<1.2 \times 10^{-12}/\text{hour}$) (31).

5.3.1.2 Earthquake

Earthquake frequencies for SRP are shown in Figure 5-3 (32). The expected frequency of an MM VIII is $2 \times 10^{-4}/\text{yr}$. This would result in some damage to the structure. Total destruction is not expected unless a level MM XII occurs (8). These data are consistent with those of Reference 33. The release sequence for an earthquake is loss of the contents of the vessels to the secondary confinement with subsequent seepage through cracks in the building and airborne releases at ground level. Failure of the sand filters would not be expected to occur for an MM VIII earthquake (34).

5.3.1.3 Meteorite Impact

A meteorite impact is considered to be a hypothetical event due to the small expected frequency of occurrence. The expected frequency of the hypothetical accident of a meteorite impact at SRP is calculated from studies of worldwide meteorite impact by assuming that meteorites are randomly and uniformly dispersed over the earth's surface. Resultant estimates of the frequency of this hypothetical event range from 10^{-9} to 10^{-14} impacts per year (35). Based on the impact of a one ton meteorite at SRP, the estimated frequency of impact from a meteorite for F-Canyon is $1.5 \times 10^{-15}/\text{hr}$.

5.3.1.4 Floods

Flooding was not considered a credible initiator (see Section 5.1.1.4).

FIGURE 5-3. Earthquake Frequency at the Savannah River Plant

5.3.1.5 Other Natural Phenomena Related Events

Extremes in temperature, snow, rain, and lightning may adversely affect operations, but will not, in general, result in release of radioactive materials. The cumulative frequency from all other natural phenomena related events is $5 \times 10^{-4}/\text{hr}$.

Cold weather has little effect except through its effect on auxiliary services. Below -8°C (for the services located outside the building), lines plug, valves and pipes burst, alarms fail to operate, instrumentation may become inoperable, and equipment with moveable parts may become immobilized. Thus, icing is a prime cause of common mode failure. The frequency of significant ice damage is $2 \times 10^{-4}/\text{hr}$.

The average annual snowfall over the last 45 years has been about 0.9 in. Trace amounts or greater can be expected three out of four years; snowfall of 0.1 in or more occurs slightly less than every other year. One inch or more has fallen in 7 of the past 45 years. The expected frequency of sufficient snow to aggravate canyon operations is estimated to be $8 \times 10^{-6}/\text{hr}$.

The expected frequency of sufficient rain falling in a 24-hr period to aggravate a deteriorated situation is estimated to be $8 \times 10^{-3}/\text{hr}$, and to cause internal building damage, $1 \times 10^{-4}/\text{hr}$. No contaminated water runoff is expected, however.

The principal adverse affect of lightning is interruption of electrical power. In 1973-74, an improved multidisk ionization-type lightning-arrester system was installed in the 13.8 kV distribution lines in place of the pellet-type arresters. These new arresters have reduced the frequency of power failure due to lightning to an estimated $2 \times 10^{-4}/\text{hr}$, an improvement by a factor of about 10.

5.3.2 Externally Induced Failures

Three significant sources of externally induced failure mechanisms were identified as applicable to canyon operations: adjacent explosions, adjacent fires, and impact. The overall frequency of occurrence of externally induced failure is estimated to be $1 \times 10^{-5}/\text{hr}$, primarily due to impact. The release sequence for an externally induced failure is shown in Figure B-1.

5.3.2.1 Aircraft Crash

Studies related to power reactors based on U.S. Civil Aviation accident data indicate that the expected frequency of aircraft overflights becomes essentially constant at distances greater than five miles from an airport runway. The expected frequency is about $3 \times 10^{-9}/(\text{flight})(\text{mi}^2)$ for commercial aviation and about $7 \times 10^{-9}/(\text{flight})(\text{mi}^2)$ for general aviation (15).

Based on a total estimated frequency of 4000 flights over SRP per year and the $4 \times 10^{-3} \text{ mi}^2$ "target" area of the F-Canyon building, the overall aircraft crash frequency is calculated to be $1 \times 10^{-11}/\text{hr}$. Since only 30% of the

crashes would be due to aircraft large enough to cause damage, the frequency for damage due to aircraft crash is $3 \times 10^{-12}/\text{hr}$.

5.3.2.2 Adjacent Explosion

Explosions are estimated to occur in the canyon with a frequency of $1 \times 10^{-6}/\text{hr}$ (1). However, an adjacent explosion is not considered a credible initiator (Section 5.1.2.2). Adjacent explosions are not expected to occur at a frequency greater than that experienced in the canyon.

5.3.2.3 Adjacent Fire

Of the 352 fires reported in F-Area, 294 have occurred outside the canyon and its ventilation system (Table 5-5). The expected frequency of occurrence of fires in F-Area outside the canyon building and its ventilation system is $1.3 \times 10^{-3}/\text{hr}$. However, an adjacent fire is not considered a credible accident initiator (Section 5.1.2.3).

5.3.3 High Energetic Events

Explosions of severe magnitude could potentially produce this event. However, the structural integrity of the canyon vessels and secondary confinement minimizes the potential to the point that no high energetic events were identified. Therefore, explosions are treated in the category of a medium energetic event and are discussed in Section 5.3.4.

There were no other accident scenarios that meet the descriptions of a high energetic event for F-Canyon.

5.3.4 Medium Energetic Events

A medium energetic event is defined as an event that will cause penetration of the primary confinement and will cause materials to bypass the second confinement barrier for a short period of time. The initiator frequencies are discussed below. The event tree for medium energetic accident sequences is shown in Figure B-2.

5.3.4.1 Fire

There have been 58 fires within the canyon and its ventilation system (excluding B-Line) (Table 5-5). Of the 58, only two involved process materials: one resin fire on top of a vessel within the canyon and one resin fire in cold feed tank 8B-1. Assuming the frequency is similar to the historical average, the expected frequency for a fire in F-Canyon from all causes is $2.5 \times 10^{-4}/\text{hr}$. The expected frequencies for process related fire initiators are developed in the event trees in Figures B-3 and B-4 ($7 \times 10^{-8}/\text{hr}$ for each cycle of solvent extraction and $9 \times 10^{-7}/\text{hr}$ for ion exchange).

TABLE 5-5. Causes of Ignition of 200 F-Area and 221 F-Canyon Fires*

Cause of Ignition	Total Number for F-Area	Frequency, Occurrences/hr	Number Only for F-Canyon	Frequency, Occurrences/hr
Process heat	77	3.2×10^{-4}	0	--
Welding	60	2.5×10^{-4}	17	7×10^{-5}
Electrical short	56	2.4×10^{-4}	20	8×10^{-5}
Spontaneous combustion	56	2.4×10^{-4}	6	3×10^{-5}
Miscellaneous or unspecified	53	2.2×10^{-4}	2	8×10^{-6}
Smoking materials	18	8×10^{-5}	4	2×10^{-5}
Sparks (nonwelding)	9	4×10^{-5}	1	4×10^{-6}
Explosion	8	3×10^{-5}	2	8×10^{-6}
Friction	7	3×10^{-5}	2	8×10^{-6}
Lightning	7	3×10^{-5}	4	2×10^{-5}
Lighting	1	4×10^{-6}	0	--
Total	352	1.5×10^{-3}	58	2.5×10^{-4}

*Fires in facilities adjacent to F-Canyon:

Number = 352 total - 58 (only in F-Canyon)

= 294 fires in adjacent facilities

Frequency = 1.5×10^{-3} /hr - 2.5×10^{-4} /hr (only in F-Canyon)

= 1.3×10^{-3} /hr for fires in adjacent facilities

5.3.4.2 Uncontrolled Reaction

Uncontrolled reactions occur with an overall frequency of about 2.4×10^{-4} /hr. A typical eruption discharges 1000 lb of materials, but up to 7000 lb have spilled to the canyon floor in a single release. The mean release for all uncontrolled reactions is 2580 lb.

Explosions have not occurred in canyon vessels. However, in 1953, a damaging explosion in an evaporator occurred at TNX as a result of a reaction involving uranyl nitrate, TBP, and nitric acid. It is concluded that the greatest likelihood of a canyon explosion will be the result of some unforeseen reaction based on a number of lesser uncontrolled reactions that have occurred but were not anticipated. An expected frequency of 1×10^{-6} /hr for explosions in F-Canyon was estimated.

The systems in which uncontrolled reactions have occurred are shown in Table 5-6. Evaporation and ion exchange operations account for 62% of the occurrences. About 16% occurred in the third-level tank gallery and involved chemical hazards to personnel, but no radioactive material release.

5.3.4.3 Criticality

There have been no criticality accidents experienced either at F-Canyon or at any other facility at SRP. Frequencies for criticality in various F-Canyon systems are therefore calculated by fault tree analysis (36,37,38). Results are summarized in Table 5-7. The event tree for a release due to criticality is shown in Figure B-5.

5.3.5 Low Energetic Events

A low energetic event is defined as an event that will not destroy the primary confinement barrier (the primary container), but activity may penetrate it. Because this event does not affect the secondary confinement, it will not in itself expose operating personnel to radioactivity. Event trees for low energetic accident sequences are shown in Figures B-6 and B-7 (airborne releases at ground level and airborne releases from the stack).

5.3.5.1 Transfer Error

Within F-Canyon, the frequency of occurrence of transfer errors is 5.1×10^{-4} /hr. About 15% of these transfers result in liquid being discharged directly to the canyon floor and sumps. A secondary effect is vessel overflow as discussed in Section 5.3.5.2. About 12% of the transfer errors result in liquid movement outside of the confines of the shielded canyon, such as to Building 211-F (4%), 241-F (2%) or a seepage basin (6%). Unexpected transfer to these locations could result in radiation exposure to personnel, contamination, overflow of vessels, or uncontrolled chemical or nuclear reactions. Quantities of radioactivity involved vary widely and are discussed in sections of this report concerned with the systems involved. The specific

TABLE 5-6. Frequency and Canyon Systems Affected by Uncontrolled Reactions

System	Number of Occurrences	Frequency, Occurrences/hr
Evaporation	17	9.2×10^{-5}
Ion exchange	11	6.0×10^{-5}
Tank gallery	7	3.8×10^{-5}
Dissolving	5	2.7×10^{-5}
Solvent extraction	3	1.6×10^{-5}
Waste disposal	1	5×10^{-6}
Rerun	1	5×10^{-6}
Head end	0	2×10^{-6} (est)
	45	2.4×10^{-4}

TABLE 5-7. F-Canyon Criticality Potential

Unit Operation	Frequency, Occurrences/hr
Dissolving	1.2×10^{-10}
Head End	1.7×10^{-9}
First Cycle	1.8×10^{-7}
2nd Pu Cycle	1.8×10^{-7}
Total	3.6×10^{-7}

effects on the facility receiving the material during a transfer error are discussed in SARs for the particular facility.

As shown in Tables 5-8 and 5-9 virtually every system in the canyons has been affected by transfer errors. Table 5-8 shows those systems in which the error was initiated, and Table 5-9 shows those systems that were insulted by the move. About 40% of all transfer errors are initiated in the cold chemical and water additions systems, followed by solvent extraction (14%) and dissolving (12%). Dissolving and canyon sumps are most likely to be affected by the errors (14%), followed by ion exchange (12%). Waste disposal, evaporation, and solvent extraction follow, each at 8%.

5.3.5.2 Overflow

The overall frequency of occurrence of overflows (4.8×10^{-4} /hr) is less than that for transfer errors.

As shown in Table 5-10, the chemical feed tanks in the gallery on the third level are involved in the greatest number of overflows at a frequency of 1.4×10^{-4} /hr. Solvent extraction follows with a frequency of about 1.3×10^{-4} /hr. Ion exchange, evaporation, sumps, and waste disposal are in the range of 3×10^{-5} /hr to 5×10^{-5} /hr.

5.3.5.3 Leak

Leaks in any unit operation fall into three principal categories: cooling water or steam leaks, cold chemical leaks, and product solution leaks during transfer to another tank. Only the last category involves significant radioactive material. Coil leaks are analyzed in Section 5.3.5.10. Leak frequencies are evaluated separately for each applicable ancillary and primary operation in Table 5-11 and summarized in Table 5-12.

Leaks in the process operations fall into three principal categories: water leaks in the condenser cooling system, cold chemical leaks into the vessel, and product solution leaks during transfer to another tank. The first two of these categories do not involve significant radioactive material. Two additional types of leaks are coil leaks and vessel leaks.

5.3.5.4 Chemical Addition Error

The frequency of occurrence of chemical addition errors is 4.4×10^{-4} /hr. As shown in Table 5-13, about 75% of all chemical addition errors are initiated in the cold chemical or water addition facilities. Ten percent are from ion exchange operations, while the remainder occur from miscellaneous operations.

Table 5-14 shows those systems that are insulted by the error. Solvent extraction and ion exchange errors are usually associated with valence adjustment. Dissolving operations are impacted mainly by unwanted water additions or by erroneous fuel charging.

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TABLE 5-8. Canyon System in Which Transfer Error was Initiated

System	Number of Occurrences	Frequency, Occurrences/hr
Cold chemicals and water additions	38	2×10^{-4}
Solvent extraction	13	7×10^{-5}
Dissolving	11	6×10^{-5}
Evaporation	10	5×10^{-5}
Ion exchange	8	4×10^{-5}
B-Line	3	2×10^{-5}
Sumps	3	2×10^{-5}
Solvent recovery	2	1×10^{-5}
Waste disposal	1	5×10^{-6}
Head end	1	5×10^{-6}
Cell flush	1	5×10^{-6}
Decontamination cells	1	5×10^{-6}
Not identified	1	5×10^{-6}
	93	5.1×10^{-4}

TABLE 5-9. Canyon System into Which Erroneous Transfer was Made

System	Number of Occurrences	Frequency, Occurrences/hr
Dissolving	13	7×10^{-5}
Sumps	13	7×10^{-5}
Ion exchange	11	6×10^{-5}
Waste disposal	7	4×10^{-5}
Evaporation	7	4×10^{-5}
Solvent extraction	7	4×10^{-5}
Seepage basin	5	3×10^{-5}
Cold chemicals	5	3×10^{-5}
Rerun	5	3×10^{-5}
Solvent recovery	4	2×10^{-5}
211-F	4	2×10^{-5}
Drain headers	4	2×10^{-5}
241-F	3	2×10^{-5}
Head end	1	5×10^{-6}
Decontamination cell	1	5×10^{-6}
Dehumidifier condensate tank	1	5×10^{-6}
Not identified	2	1×10^{-5}
	93	5.1×10^{-4}

TABLE 5-10. Canyon Systems Affected by Overflows

System	Number of Occurrences	Frequency, Occurrences/hr
Tank gallery	26	1.4×10^{-4}
Solvent extraction	24	1.3×10^{-4}
Evaporation	10	5×10^{-5}
Ion exchange	8	4×10^{-5}
Waste disposal	8	4×10^{-5}
Sumps	6	3×10^{-5}
Sample aisle	5	3×10^{-5}
Head end	1	5×10^{-6}
Dissolving	1	5×10^{-6}
Rerun	0	--
	89	4.8×10^{-4}

TABLE 5-11. Canyon Systems in Which Leaks Occur

Unit Operation	Type of Leak	Number of Occurrences	Frequency, Occurrences/hr
Decontamination Facilities	All leaks	--	5×10^{-4}
	Contaminated solution	--	2.5×10^{-4}
	Uncontaminated solution	--	2.5×10^{-4}
Dissolver	Cooling water	4	8×10^{-5}
	Cold chemicals	2	4×10^{-5}
	Coating waste	13	2.5×10^{-4}
	Product transfer	7	1.3×10^{-4}
Head end	Cooling water/steam	21	4×10^{-4}
	Process solution	40	8×10^{-4}
	Cold chemicals	1	2×10^{-5}
Solvent extraction	All leaks	--	5.4×10^{-3}
	Contaminated waste solution transfer	--	2.7×10^{-3}
	Uncontaminated waste solution transfer	--	2.7×10^{-3}
Evaporation	All leaks	--	7.3×10^{-3}
	Contaminated solution	--	3.7×10^{-3}
	Uncontaminated solution	--	3.7×10^{-3}
Ion exchange	Cooling water/steam	36	7×10^{-4}
	Process solutions	47	9×10^{-4}
	Cold chemicals	5	1×10^{-4}
Waste disposal	Cooling water/steam	29	6×10^{-4}
	Process solutions	69	1×10^{-3}
	Cold chemicals	1	5×10^{-5}

TABLE 5-12. Frequencies of Contaminated Process Solution Leaks

Unit Operation	Frequency, Occurrences/hr
Ancillary Operations and Equipment	
Shops and decontamination facilities	3×10^{-4}
Gang valve corridors	6×10^{-3}
Steam distribution	
Steam header	2×10^{-4}
Condensate header	6×10^{-4}
Cold feed preparation and storage	2×10^{-3}
Primary Operations and Equipment	
Dissolving	1×10^{-4}
Head end	8×10^{-4}
Solvent extraction	3×10^{-3}
Evaporation	7×10^{-3}
Ion exchange	9×10^{-4}
Waste disposal	1×10^{-3}

TABLE 5-13. Canyon System from Which Chemical Addition Error was Initiated

System	Number of Occurrences	Frequency, Occurrences/hr
Cold chemicals and water addition	63	3.4×10^{-4}
Ion exchange	9	5×10^{-5}
Fuel storage	4	2×10^{-5}
Solvent extraction	2	1×10^{-5}
Evaporation	1	5×10^{-6}
Dissolving	1	5×10^{-6}
Head end	1	5×10^{-6}
	81	4.4×10^{-4}

TABLE 5-14. Canyon System into Which Chemical Addition Error was Made

System	Number of Occurrences	Frequency, Occurrences/hr
Solvent extraction	30	1.6×10^{-4}
Ion exchange	21	1.1×10^{-4}
Dissolving	12	7×10^{-5}
Cold chemicals	7	4×10^{-5}
Waste disposal	4	2×10^{-5}
Solvent recovery	3	2×10^{-5}
Head end	1	5×10^{-6}
Evaporation	1	5×10^{-6}
B-Line	1	5×10^{-6}
Drain header	1	5×10^{-6}
	<hr/> 81	<hr/> 4.4×10^{-4}

5.3.5.5 Pluggage

The most common form of pluggage experienced in F-Canyon is instrument line pluggage that occurs at a rate of one to two per day (5×10^{-2} /hr). Sampler pluggage occurs about once per week (7×10^{-3} /hr). Jet pluggage and transfer air pluggage each occurs about once per month (1×10^{-3} /hr).

5.3.5.6 Siphoning

Siphoning has not been reported as having occurred in the dissolving, head end, solvent extraction, or waste disposal operations. It is estimated that it could occur at an expected frequency of 1×10^{-6} /hr.

Siphoning has been reported as having occurred in the evaporator system at a frequency of 2×10^{-5} /hr.

5.3.5.7 Processing Short-Cooled Fuels

Because shipment of the short-cooled fuel is beyond the control of either reprocessing area, the historical frequency of once per thirty years is assumed to be applicable to either area. This is equivalent to 4×10^{-6} /hr.

5.3.5.8 Ruthenium Volatilization

No specific occurrences have been reported for either head end or solvent extraction operations.

About 20 Ci of ^{106}Ru were released from the stack when the 6.1D dissolver was operated at boiling accompanied by sparging while the upper steam coils were partially exposed to the vapor phase. The acid scrubbers were bypassed. This single event leads to a frequency of 6×10^{-6} /hr based on operating history.

5.3.5.9 Iodine Reactor Failure

No significant releases of iodine have been reported as a result of failures in the iodine reactors.

5.3.5.10 Coil and Tube Failure

As shown in Table 5-15, over 60% of the coil and tube failures occur in waste evaporation. Failures also occur in dissolver heating coils and in heating or cooling coils in various canyon hold tanks. These failures are caused mainly by corrosion.

During the period studied a total of 54 coil failures occurred in both F- and H-Canyons for an overall frequency of 2.9×10^{-4} /hr. The frequency with which activity is detected in the segregated water is 1×10^{-3} /hr. The primary cause is coil or reboiler failure during operation of a canyon vessel (1.2×10^{-4} /hr). Less than 1/3 of the coil failures result in releases to the

TABLE 5-15. Canyon Systems Affected by Coil and Tube Failures

System	Number of Occurrences	Frequency, Occurrences/hr
Evaporators		
LAW	17	
HAW	14	
Product/intercycle	3	
Rerun	1	
Subtotal	35	1.9×10^{-4}
Dissolvers	3	1.6×10^{-5}
Tanks		
Head end	4	
Product/intercycle	6	
Waste	5	
Solvent recovery	1	
Subtotal	16	8.7×10^{-5}
	54	2.9×10^{-4}

segregated water system. Other potential causes include residual activity becoming dislodged, the Cash pressure system being valved off during shutdown of a vessel with a failed coil, heat exchanger leaks, and unspecified causes. Multiple detections and diversions, however, may result from a single failure.

5.3.5.11 Waste Header Failure

No material has been recorded as having been lost at the waste farm as a result of waste header failure. On one occurrence, however, waste was released from the canyon because of a transfer error and sent to the waste farm without the knowledge of personnel at the receiving end. An increase in tank level was observed before overflow occurred, and the transfer was stopped. The estimated frequency for this event is 4×10^{-6} /hr.

Double encasement of the transfer line from the canyon to 241-F has prevented any recorded losses of activity to the soil between the facilities. It is estimated that a sudden loss could occur from internally induced failures such as an explosion or flaw or from externally induced failure such as crushing or ground shifting at a frequency of 1×10^{-7} /hr. In addition, slower losses could occur from corrosion. The frequency is estimated at 1×10^{-6} /hr.

Loss of waste from header failure inside the canyon has occurred at a frequency of 6×10^{-5} /hr, primarily because of leaks. In one case, a blank had been removed from a waste line.

On two occasions, waste material backed up into an air tunnel through vents due to pluggage. Partial pluggage of waste headers occurs about once per month, but it is normally cleared before escape of material occurs. Based on the record, failure to remove a plug before activity escapes occurs at a frequency of 1×10^{-2} /demand. The frequency of activity release is therefore 8×10^{-6} /hr.

5.3.6 Residual Activity Release Events

A residual activity release event is one that allows the residual activity that accumulates in the secondary confinement to reach personnel areas or the outside. These events include suckback and air reversals.

5.3.6.1 Suckback

Seventeen cases of suckback from the hot canyon of gamma intensity at varying distances were reported. The data from both F- and H-Area were normalized to a distance of three inches to yield a calculated mean of 175 R/hr. Intensities up to 500 R/hr at five inches have been reported.

5.3.6.2 Air Reversals

Air reversals usually result from multiple causes that vary somewhat according to the system affected. Open barriers (e.g., doors, cell covers) contributed to 29 of the 37 occurrences. In 14 cases, the air balance between the supply

and exhaust systems was lost by inadequate exhaust and/or excessive air supply. Maintenance operations were being performed in seven cases. Other contributing causes are ventilation air leaks past barriers, damper malfunctions, duct pluggage, and local power failure.

The overall frequency of air reversals somewhere in the canyon is 2.0×10^{-4} /hr.

5.3.7 Chemical Hazards

Historically, the principal initiators for chemical releases are leaks, overflows, transfer errors, and uncontrolled reactions (Table 5-16). Only one (uncontrolled reaction) is classed as a medium-energy event; the others are low-energy events.

Leaks are normally not a major concern. Typically, leaked material is confined by a curbing around the vessel apron and flows into a drain leading to a collection sump on first level. Some volatilization may occur, but the building ventilation system is adequate to protect against high concentrations. More significant hazards to personnel are the spraying leak and leaks that travel to clean areas of the building through expansion joints in the building floor. In a few cases, workers have experienced clothing, skin, or hair contamination by leaking or spraying chemicals. In other cases, fumes (from leaked chemicals that caused a chemical reaction) have exceeded the Acceptable Exposure Limit (AEL) for short periods, but in no case has the Immediate Danger to Life and Health (IDLH) level been reached. The mean mass of leaked solutions cannot be calculated accurately; data are not available.

Overflows behave similarly to leaks, except that splashing is more likely. Most of the material, however, is confined by the curbing, drain, and sump. The mean mass of overflowed material in the cold feed preparation (CFP) area was 3570 lb.

Transfer errors in CFP are generally of economic rather than safety concern, because of the undesired mixing with other chemicals. Erroneous movement of chemicals from third level to the canyon usually is more significant, as discussed in the individual process analyses. The mean mass of material involved in a transfer error in CFP was 14,620 lb.

Uncontrolled reactions are the least frequent, but potentially the most hazardous of the mechanisms for removal of storage vessel contents. Because of the energetic manner in which the material is expelled, the consequences vary widely. Some will, of course, drain to the collection sump, but much of the material often sprays into the access aisle, creating high airborne concentrations. Large ventilation openings provide ample passageway for airborne movement to the second level where unprotected equipment and personnel are subject to exposure. The mean mass involved in uncontrolled reactions in CFP was 2580 lb. However, this value is dominated by uncontrolled reactions of cold chemicals in canyon processing, rather than in CFP itself.

TABLE 5-16. Frequencies of Chemical Releases from Cold Feed Preparation Area

Initiating Event	Frequency,* Occurrences/hr	Outside Release Probability**	Outside Release Frequency, Occurrences/hr	Release Point
Overflow	1.4×10^{-4}	0.023	3.2×10^{-6}	Seepage basin
Leak	2.2×10^{-3}	0.008	1.8×10^{-5}	Truck dock
Transfer error	2.4×10^{-4}	0.04	1.0×10^{-5}	211-F
Uncontrolled reaction	3.8×10^{-5}	0.026	1.0×10^{-6}	Stack
Total			3.2×10^{-5}	

*Data from 200-Area Data Bank.

**Data from 200-Area Data Bank. Some data include H-Area data also.

5.3.7.1 Frequency of Releases to Outside Areas

The frequency of each initiating event was obtained from Table 5-16. Source data from the data bank were used to determine the probability that an event resulted in a release either outside the building or inside the building. Examples of nonrelease events are overflows or leaks from one tank to another. The intent in calculating release probabilities is to be sufficiently conservative so as to allow for those releases of unknown origin that are detected outside the building, but which may originate in the facility being analyzed.

The product of the initiating event frequency and the subsequent probability of outside release is the frequency of release from CFP to a location outside 221-F (Table 5-16). The overall frequency is 3.2×10^{-5} /hr. Release sites outside the building were identified in the data bank and are indicated in Table 5-16.

5.3.7.2 Frequency of Releases to Occupied Areas

The frequency of releases inside the CFP area was calculated in a similar manner to that for releases to outside areas above. Results are shown in Table 5-17. Figure B-8 shows the logic and the frequency values used in this calculation.

5.3.8 Support Systems

The frequencies of occurrence of failures of ancillary operations and equipment are discussed in the following sections.

5.3.8.1 Shops and Decontamination Facilities

Frequencies of loss of control of contaminated solutions were discussed in Sections 5.3.4 and 5.3.5.

Exposure of personnel is under administrative control for radiation and contamination protection, which are only as effective as the personnel at the work site. Failure to follow the procedures occurs with a frequency of 10^{-3} to 10^{-4} /demand. Serious contamination and exposure cases in the canyons indicate that failure to comply is on the low end of the spectrum, or about 10^{-4} /demand.

Release of airborne activity to personnel areas is estimated to occur at a frequency of 1×10^{-4} /hr. Very limited quantitative data indicate activity levels in the order of $1 \times 10^3 \times \text{RCG}$.

5.3.8.2 Cell Covers

The historical frequency of dropping a cell cover has been 1×10^{-5} /hr; however, none have caused significant damage to equipment.

TABLE 5-17. Frequencies of Chemical Releases to Occupied Areas

Type of Incident	Frequency*, Occurrences/hr
Hazardous liquids	
Overflow	1.7×10^{-5}
Leak	8.0×10^{-5}
Transfer error	9.4×10^{-6}
Uncontrolled reaction	1.7×10^{-6}
Total	1.1×10^{-4}
Hazardous vapors	
Overflow	1.5×10^{-6}
Leak	3.2×10^{-5}
Transfer error	8.3×10^{-7}
Uncontrolled reaction	2.6×10^{-6}
Total	3.7×10^{-5}

*Frequencies were calculated from data in Table 5-16 and Figure B-8.

5.3.8.3 Gang Valve Corridors

Frequency data for gang valve failures indicate a mean time between occurrences of 7 days ($6 \times 10^{-3}/\text{hr}$). Leaks are reported about once per week ($6 \times 10^{-3}/\text{hr}$). Significant cases of surface contamination are reported about once per month.

5.3.8.4 Cranes

Failure or impairment of a crane or its equipment occurs at a frequency of once per day. Dominant causes and frequencies are shown in Table 5-18.

The average rate of dropping of loads is about three per year; however, no major damage has resulted from the dropping incidents.

Damage to crane ancillaries has also occurred. Spalling of concrete beneath the crane rails occurred in 1979 and required 18 days to complete repairs. Failures with shielding doors and removable rails occur about three times per month, primarily with the rail. Thirty to fifty lifting yokes must be straightened each year.

Hazards to personnel are radiation, contamination, and personal injury. Radiation exposure occurs primarily during maintenance. Airborne contamination in the crane cabs, primarily from air conditioner failure, occurs about four times per year on the hot crane and twice per year on the warm crane. Surface contamination also occurs within the crane cabs, primarily due to leaks in the air conditioner, gear box, and hydraulic system. These occur about four times per year in the hot crane and twice a year in the warm crane. Personnel injuries occur about once per year.

5.3.8.5 Sample Aisles

Causes and frequencies of equipment malfunctions in the sample aisles are shown in Table 5-19. Significant surface or airborne contamination cases occur about once per week. Radiation fields at the open door to samples occur at a rate of about once every three weeks. Two fires have been recorded for the aisle. Fumes have migrated into the sample aisle from the cold chemical storage area. Three injuries have been recorded.

5.3.8.6 Steam Distribution

Complete area-wide steam outages occur at a rate of $2.2 \times 10^{-4}/\text{hr}$ and last an average of about one day. Half of these outages are due to failures in the powerhouse, and half are due to failures in the main high-pressure (325 psi) header system.

Reduced steam supply, affecting one or more unit operations for short periods, occur with a frequency of $9 \times 10^{-4}/\text{hr}$ and is usually due to problems at the powerhouse ($8 \times 10^{-4}/\text{hr}$). The remainder is caused by failures in the 325 psi distribution system ($1 \times 10^{-4}/\text{hr}$).

TABLE 5-18. Cause and Frequencies of Crane Malfunctions

Cause	Hot Crane Frequency	Warm Crane Frequency
Wheel and bearing failure	~6 months	4 years
Major brake repair or adjustment	Every other month	Once every 6 months
Cable replacement	Once per month	Once per quarter
Breakdown of communications	Once per year	Once per month
Impact wrenches	Once per week	Once per week
Electrical failure	1-2 per month	1-2 per month
Relamping	Every other month	Every other month

**TABLE 5-19. Causes and Frequencies of Equipment Malfunctions
in Sample Aisles**

Cause	Frequency
Failure to sample (all causes)	$1 \times 10^{-2}/\text{hr}$
Failure to sample due to plugged sample jet	$7 \times 10^{-3}/\text{hr}$
Hoist failure	$\sim 1/\text{month}$
Elevator failure	$\sim 2/\text{year}$
Radiation and contamination detection instruments	$\sim 1/\text{month}$

Severe condensate header leaks occur at a frequency of 6×10^{-4} /hr. In general these leaks are a nuisance rather than a hazard, and are tolerated until a sufficient number occur to justify isolation of the headers. Usually, they are repaired during scheduled process shutdowns.

Contamination spread due to leakage or maintenance occurs at a frequency of 2×10^{-5} /hr.

5.3.8.7 Compressed Air

The expected frequency of compressor failure in the plant air, instrument, and process air systems is 2×10^{-5} /hr, and 8×10^{-6} /hr for the breathing air system. Coupled with the probability of electrical failure external to the compressor motors, the overall expected frequencies of compressed air failure are 3×10^{-5} /hr and 2×10^{-5} /hr, respectively.

5.3.8.8 Electrical Distribution

The hierarchy and failure frequencies of the electrical distribution system were analyzed as part of the ventilation system fault tree in Appendix B of the F-Canyon Systems Analysis (1). The expected frequency for failure of a given piece of vital equipment for reason of electrical malfunction is 2×10^{-5} /hr.

5.3.8.9 Cooling Water Distribution

Based on the F-Canyon Systems Analysis (1) the expected frequency for failure to supply cooling water in a normal manner to equipment is calculated to be 4×10^{-5} /hr.

5.3.9 Engineered Safety Features

The frequencies of occurrence of failures of engineered safety features are discussed in the following sections.

5.3.9.1 Ventilation System

Performance of the sand filter is of prime importance in limiting the amount of particulate activity released to the atmosphere from operations in the canyons. No single number can be given for filtration efficiency; but under routine operating conditions, efficiencies of 99.8 to 99.99% for beta-gamma, and 99.5 to 99.9% for alpha activity are normal. However, analysis of stack emissions, as a function of input activity to the sand filter, shows that at high input (burst release) the sand filter efficiency is 99.98% (39). The sand filter units have served well to retain activity and reduce atmospheric releases except in two specific cases when failure of a support tile in the bottom air distribution system let the gravel and sand fall into the bottom supply air tunnel. In general, increased releases are experienced even during

minor disruptions of the sand bed. The bed becomes less effective and releases sorbed radioactive material such as ^{106}Ru and ^{95}Zr . The sand bed becomes less effective if the air flow through the bed is perturbed. These perturbations may occur in either the process vessel vent system or in the canyon exhaust air.

For analysis purposes, the efficiency of the sand filter for all particulates is assumed to be 99.51%, which corresponds to the worst 97th percentile of the measured efficiencies (40).

Complete failure of either of these exhaust fan systems could cause an air reversal. Only one such incident has occurred in 221-F; both canyon and center section ventilation air were interrupted for about 40 min in June 1970 when a fire occurred in a 440 volt breaker. Since then, the system has been modified to provide emergency power directly to canyon exhaust. No further complete failures have occurred. Thus, the frequency of such failure ($1 \times 10^{-7}/\text{hr}$) was calculated by fault tree analysis as shown in Appendix B of the F-Canyon Systems Analysis (1).

The dominant contributors to the ventilation system failure are the one component cutsets which include a major earthquake, exhaust air tunnel collapse, and stack collapse. Expected frequencies of major earthquakes were calculated using the "conservative" value of an MM-VIII earthquake. Stack collapse was based on the frequency of being struck by a tornado (41). Tunnel collapse from causes other than earthquake was based on judgement. The failure frequency of the canyon fans would be most affected by uncertainties in the above values. The expected frequency for failure of the entire system was calculated to be $1 \times 10^{-7}/\text{hr}$.

Frequencies of local air reversals are summarized in Table 5-20.

History indicates that the frequency of losing all stack sampling and monitoring is about $1 \times 10^{-5}/\text{hr}$. There is only one recorded instance.

Other ventilation system single-component failure rates are shown in Table 5-21 (42).

5.3.9.2 Water Return and Diversion

The frequencies for the seven mechanisms identified for loss of activity into plant streams are shown in Table 5-22.

5.3.9.3 Seepage Basins

A complete analysis of the seepage basins is provided in Reference 25.

5.3.9.4 Secondary Confinement for Liquids

The frequency of reported leakage from expansion joints is slightly over once per month.

TABLE 5-20. Frequencies of Local Air Reversals

Location	Frequency, Occurrences/hr
Railroad tunnel	8.8×10^{-5}
Center section	3.2×10^{-5}
Crane maintenance	8.0×10^{-6}

TABLE 5-21. Ventilation System Single Blower Failure Rates

Ventilation System	<u>Failure Rate, Occurrences/hr/fan</u>	
	Air Supply	Air Exhaust
Canyon air	4.0×10^{-5}	2.0×10^{-4}
Central air	5.5×10^{-5}	3.7×10^{-5}
Process vessel vent	--	1.9×10^{-4}
Recycle vent	--	1.0×10^{-4}

TABLE 5-22. Frequencies for Release of Activity into Plant Streams

Mechanism	Causes	Frequency
Activity in segregated water		$1 \times 10^{-3}/\text{hr}$
	Diversion valve failure	$1.5 \times 10^{-4}/\text{demand}$
	Monitor failure	$1 \times 10^{-2}/\text{demand}$
	Personnel error	$3 \times 10^{-2}/\text{demand}$
Activity in either Upper Three Runs or Four Mile Creek		$6 \times 10^{-6}/\text{hr}$
	Activity in segregated or circulated water systems	$3 \times 10^{-5}/\text{hr}$
	Small break in effluent piping	$3 \times 10^{-2}/\text{demand}$
	Sufficient rain to result in runoff	$0.2/\text{demand}$
Major break in effluent piping		$3 \times 10^{-4}/\text{demand}$
Activity reaching cooling tower at same time cooling tower excess is overflowing to Four Mile Creek		$4 \times 10^{-6}/\text{hr}$
	Activity in circulated cooling water and failure to divert to retention basin	$4 \times 10^{-6}/\text{hr}$
	Activity from waste tank due to heat exchanger leak	$4 \times 10^{-7}/\text{hr}$
	Leak in isolation gate valve and activity in circulated cooling water	$1 \times 10^{-7}/\text{hr}$
Improper removal of contaminated sludge from delaying basin		$1 \times 10^{-6}/\text{hr}$
Activity sorbs in soil of effluent ditch		$2 \times 10^{-6}/\text{hr}$
Malfunctions within canyon		$4 \times 10^{-6}/\text{hr}$

Liquids in canyon sumps are monitored by sump liquid-level detectors. Sump liquid level failures have occurred at an overall frequency of $2.1 \times 10^{-4}/\text{hr}$.

5.3.9.5 Pressure Regulators

A study of the maintenance and repair of Cash regulators shows that some failures have occurred. However, these failures are infrequent because of preventive maintenance on these regulators. Some regulators have been routinely replaced at 6-month intervals, thereby reducing their failure rate in operation. Cash regulator failures are mechanical in nature (43). The mean failure rate for each Cash regulator is about one failure every two years ($6 \times 10^{-5}/\text{hr}$).

5.3.9.6 Neutron Monitors

The overall frequency of neutron monitor failures is $4.0 \times 10^{-4}/\text{hr}$ for each monitor unit.

5.4 ACCIDENT CONSEQUENCES

The source term calculations were performed using the methodology documented in Appendix A of the F-Canyon Systems Analysis (1). The batch sizes and isotopic content used as the basis for consequence calculation varied for each unit operation. These source term calculations are included in Appendix C of this document.

The techniques used to estimate the amount of activity released to various locations are described in detail below. In general, the release mechanism is a liquid spill to the canyon floor by whatever initiator is postulated followed by transport to the stack via evaporation. Assumptions used in these calculations are documented in Appendix A of the F-Canyon Systems Analysis (1).

5.4.1 Natural Phenomena

Extremes in winds, earthquakes, and other natural phenomena may adversely affect operations. The following paragraphs discuss these events.

5.4.1.1 Winds

Two types of winds are considered in this section: straight winds (including hurricanes) and whirling type winds (including tornadoes). No radionuclide releases are postulated as the result of winds.

Straight Winds Reinforced concrete buildings are not expected to be affected by straight winds, even at speeds up to 175 mph.

Tornado. The F-Canyon building is designed to withstand a Design Basis Tornado (DBT). No release of activity in liquid or finely divided solid form is expected due to this event.

The H-Canyon building has been analyzed with regard to tornado resistance and the results extrapolated to F-Canyon. It is concluded that the outer walls would remain intact during a DBT (34). Uncontained equipment such as ventilation ducts, electrical services, and exposed pipelines are, however, vulnerable to missiles. The resistance of truckwell and railroad airlock doors is questionable, but loss of these doors would not be significant unless the inner shielding doors were open.

5.4.1.2 Earthquake

The canyon buildings were designed to resist a horizontal force equal to 10% of the total dead load (including the operating weight of equipment) plus 50% of the live load. The canyon building is expected to remain standing and the outer walls would not be significantly damaged by an earthquake of less than intensity MM VIII (34,44). The results of the H-Canyon analyses extrapolated to F-Canyon indicate, however, that some of the internal sections of the buildings marginally meet the no overall collapse criteria, but local overstressing is possible. Types of possible local failures include: 1) local crushing failure under combined bending and an axial compression; 2) local failure under combined bending and axial tension; 3) formation of the three hinge failure mechanism in a floor slab; and 4) lateral instability of a local frame. The building could have possible local failures as follows:

Building 221-F Section 2 (Typical)

- Floor slabs at levels 2, 3, and 4 along wall A, and at level 4 along column E and wall F - local failure under combined bending and tension.
- Base of column E at level 1 - local crushing under combined axial compression and bending with ductility ratio of 1.46 and ratio of axial load to capacity of 0.44.
- Frame between column E and wall F above level 3 - lateral instability due to formation of failure mechanisms.

Closer examination of the above possible failures indicate the following:

- Floors with the three hinge mechanism could exhibit appreciable sag, but total collapse is not probable.
- Collapse of columns is not probable due to the redistribution of lateral load to nearby members and the extra capacity provided by the dowels for a portion of the height above the base.

The expected frequency of an MM VIII earthquake is used as the basis for assuming loss of one-half of the contents of a system to the secondary confinement, with subsequent seepage through cracks into the soil beneath the

canyon. Based on the source term calculations in Appendix A of the F-Canyon Systems Analysis (1), one-half the contents of a system would be lost to the secondary confinement with subsequent seepage through cracks to the ground. The consequences for this event would be 0.005% released to the ground and $5 \times 10^{-6}\%$ to the atmosphere (8) at ground level. The source terms for airborne releases at ground level are shown in detail in Table C-1.

5.4.1.3 Meteorite Impact

For this study, it was assumed that no significant release of activity would be caused by small meteorites. Damage due to large meteorites is not included in this analysis due to the low probability of occurrence. This event is considered to be hypothetical.

5.4.1.4 Other Natural Phenomena Related Events

Cold weather has little effect within F-Canyon except through its effect on auxiliary services. This results in pluggage of lines and valves that are located outside of the building. Pipes may burst and instrumentation may become inoperative. Icing is a prime cause of common mode failure.

In addition to canyon services being impaired, activity release can occur through broken lines. Release to controlled storm sewers of 2.2×10^{-12} Ci/ml and to the ground of 4.5×10^{-9} Ci have been measured. Releases to the storm sewer are included in the overall risk due to releases to the seepage and retention basins. Treblier monitors in lines to the seepage basin have become inoperative due to freezing, thus reducing the effectiveness for detecting contaminated line breaks.

There have been no known adverse effects on equipment due to hot weather.

Snow represents an aggravation that can affect normal operations and maintenance. The largest snowfall recorded (18 in in February 1973) resulted in shutdown of canyon operations, primarily due to immobilization of transportation equipment. Although some damage resulted, it was attributed to the 14°F minimum temperature following the snowfall.

Occasionally, rain has been both an aggravation to a deteriorated situation, and a damage-inflicting condition in itself. Five cases are recorded of contamination being spread as a consequence of rain, four of these to outside the area perimeter. Although not directly affecting canyon operations, damage to excavations, cave-in of drain lines, near overflow of seepage basins, seepage into sand filters, and spurious annular leak alarms on waste tanks have occurred as a direct result of hard rains.

Leaks within the canyon building have also caused some spread of contamination and electrical shorting. The parapet walls on the roof have been penetrated to permit better drainage, but some leakage still occurs.

As in the case of very cold weather, the effectiveness of personnel is reduced during rain. Routine inspection and maintenance may be limited to essential equipment. Personnel are less likely to follow detailed procedural steps

during rain than when it is not raining, as indicated by recorded general untoward occurrences.

The principal adverse effect of lightning is interruption of electrical power. Lightning has resulted in several spurious alarms being actuated. Alarm modifications have reduced the incidence but have not completely eliminated false alarms. Personnel confidence in, and consequence response to, alarms that have a history of spuriously sounding may be severely reduced.

One electrical power pole fire and one grass fire were attributed to lightning. Also, some minor damage has occurred to outside equipment on 440 V systems. These lower voltage systems generally are not protected by lightning arresters. No radiological releases are attributed to lightning.

5.4.2 Externally Induced Failures

Three significant sources of externally induced failure mechanisms were identified as possibly significant for canyon operations: adjacent explosions, adjacent fire, and impact which includes incidents such as a cell cover being dropped on a vessel. The overall expected frequency of occurrence is calculated to be 1×10^{-5} /hr; however, it is estimated that damage sufficient to cause large losses of vessel contents would occur only 10% of the time to any specific system.

The contents of a vessel in the canyon could be lost to the canyon sumps as a result of system failure due to external causes. It is assumed that half of the vessel contents would be deposited in a canyon sump. The event tree for releases from the stack due to externally induced failures is shown in Figure B-1. The calculation of radionuclide release from the stack is documented in Appendix A of the F-Canyon Systems Analysis (1). The results are shown in detail in Table C-2 of this document.

5.4.2.1 Aircraft Crash

It is expected that an airplane crashing into the wall of a reinforced concrete, blast resistant structure will cause some damage inside the building. The damage, however, is expected to be slight for the size of aircraft normally flown in this area, and the crash is not expected to lead to release of activity. Large aircraft with heavy fuel loads that might penetrate the roof and the cell covers and then explode and burn are assumed to release 1% of the total amount of activity being processed at the time of the accident. This event is assumed to be a ground-level release. The overall frequency of occurrence is 1.4×10^{-11} /hr, however, and thus the consequences of this event were not analyzed separately.

5.4.2.2 Adjacent Explosion

Damage is not expected in F-Canyon systems as the result of explosions in an adjacent facility (Section 5.1.2.2).

5.4.2.3 Adjacent Fire

Damage is not expected in F-Canyon systems as the result of fires in an adjacent facility (Section 5.1.2.3).

5.4.3 High Energetic Events

The consequences associated with an event of this energy could severely contaminate personnel areas. Only an explosion of severe magnitude could potentially produce this result. Potential explosions in F-Canyon are not expected to produce consequences this severe. Explosions, therefore, are treated as medium energetic events.

5.4.4 Medium Energetic Events

Medium energetic events can contaminate the canyon operating area and some personnel areas. Examples of medium energetic events include fires, criticality, and uncontrolled reactions.

The event tree for medium energetic event releases from the stack is shown in Figure B-2. Assumptions used in the source term calculations are discussed in the following sections.

5.4.4.1 Fire

Only two fires have occurred involving process materials: one resin fire on top of a vessel within the canyons and one resin fire in cold feed tank 8B-1 (Section 5.3.4.1). Thus the only fires that were analyzed in detail were fires in solvent extraction and ion exchange. For the postulated fire in solvent extraction; the solvent must be available outside the contactors.

Solvent vaporizes prior to burning, tending to concentrate nonvolatile components in the unburned residue. Less than 1% of the radionuclides are expected to become airborne. For analytical purposes a partition factor of 1×10^{-2} was used. For the postulated fire in ion exchange, resin must accumulate outside the vessel. The event trees for the two postulated fires in the canyon are shown in Figures B-3 and B-4. The initiator frequencies for these fires were based on the frequency that materials are available outside the canyon vessel. The radionuclide release from the stack for a fire involving 263 lb from a process leak are shown in detail in Table C-3.

5.4.4.2 Uncontrolled Reaction

The consequences of uncontrolled reactions that resulted in releases of liquids were quantified in the same manner as that for transfer errors, described in Section 5.4.5.1. Of the incidents studied for both F- and H-Canyons, 45% did not result in any liquid release, 42% were spills to the floor or sump, and 13% sent process material to another vessel. Quantitative data were not available for 17% of the incidents. The mean mass of material released from uncontrolled reactions was 2580 lb. No distinction was made

between ion exchange materials and canyon process materials because only 7% of the incidents involved ion exchange materials.

Where no liquid release occurred, the consequences were typically a pressurized vessel, high temperatures, and foaming solution. One incident resulted in a resin fire in an ion exchange column. Another incident resulted in a release of airborne ruthenium from the stack.

An uncontrolled reaction that expels liquid from openings in the system or an explosion of sufficient intensity to rupture a vessel would create an aerosol of the tank contents, which, in the worst case, would fill the entire cell (45). The median size is assumed to be 30 microns (about the maximum size that could be suspended in the ventilation air). The upper-limit concentrations would be 100 mg/m^3 for aerosols of liquids. The upper-limit concentration of 100 mg/m^3 for liquids appears reasonable, as it approximates the concentration of water in a rain or drizzle (46).

Consequences of uncontrolled reactions are based on a mean value of 2580 lb ejected from the vessel. Sequence frequencies are based on the frequency that the uncontrolled reaction results in material released to the canyon floor from the individual process operations and is released via the ventilation system. Source term calculations are shown in detail in Table C-4.

5.4.4.3 Criticality

Prior to 1967, no less than 37 events were experienced throughout the world in which the power level of fissile systems became uncontrollable because of unplanned or unexpected changes in system reactivity (47-49). No excursions of this type have occurred at the Savannah River Plant. The consequences of the criticality events have varied widely, ranging from minor contamination to death and significant localized property damage.

Radiological significance of criticality can generally be related to the total number of fissions occurring during the accidents, coupled with a knowledge of the physical layout of the facility. Although it is possible to calculate the maximum number of fissions occurring during a particular accident in a given system in which the accident is terminated by either scram or disassembly (49), the cumulative distribution functions for the spectrum of possible conditions is infeasible to determine. A generic estimate was made, however, by using the information from recorded accidents.

Data from 22 events were selected as a basis for quantification of the consequences of a possible excursion within the canyons (47). Three statistical functions were calculated for both criticality accidents in solutions, and for criticality accidents in solid systems. The mean number of fissions for accidents in solutions is 2×10^{18} , and in solid systems is 5×10^{17} . The density function and cumulative distribution function are shown in

*U.S. Nuclear Regulatory Commission Regulatory Guide 3.34 specifies 1×10^{19} fissions which is in the upper tenth of incident experience. SRP analyses are based on mean values, to the extent possible, for all incidents.

Table 5-23. Data for consequence calculations used in this analysis are given in Table 5-24. The event tree for airborne releases from the stack due to criticality and the detailed radionuclide source terms are shown in Figure B-5 and Table C-5.

The cumulative distribution functions are plotted on a log probability graph as shown in Figure 5-4. From this graph, given that a criticality accident occurs, the probability of the accident involving a total number of fissions equal to or greater than the value shown may be determined. Application of this information is made for each operation within the canyons in which a criticality accident could occur.

Results of radiation dose calculations from a postulated nuclear incident of 10^{18} fissions show that the most exposed areas are the warm canyon gang valve corridor, the warm canyon crane, and the maintenance bridge. The radiation dose in the gang valve corridor directly opposite the mixer-settlers is 75 rem, but less than 0.1 rem if the radiation beam reaching the receptor would subtend a 30° angle with the 2-ft concrete wall. Radiation doses in the warm canyon crane cab and the maintenance bridge are 4.5 rem directly above the incident and only 0.02 rem at 60° from the vertical. These doses are significantly lower than doses that result in early fatalities.

5.4.5 Low Energetic Events

A low energetic event will not destroy the primary confinement barrier (the canyon vessel) but activity may be released from it. These events by themselves do not necessarily expose personnel to radioactivity. Examples of low energetic events are transfer errors, overflows, leaks, chemical addition errors, and pluggage.

Release sequences and source terms were developed for airborne releases at ground level due to transfer errors to 211-F (Figure B-6 and Table C-6) and for airborne releases to the stack due to transfer errors, overflows, and leaks to the sump (Figure B-7 and Tables C-7 to C-9). The fault tree for releases to Four Mile Creek from coil failures is shown in Figure B-9; the source terms are developed in Tables C-10 and C-11. Assumptions and data used in developing the event trees and fault tree are documented in the F-Canyon Systems Analysis (1). In addition, liquid releases to 211-F due to transfer errors are also discussed.

5.4.5.1 Transfer Error

The course of a transfer error is virtually impossible to predict. Of 93 cases examined, 89 unique errors occurred. Only four occurred twice. Amounts of material range from a few pounds to more than 140 tons, and activity levels range from nil to that contained in HAW concentrate. Usually, the larger masses are nonradioactive chemicals or water.

Direct consequences also vary widely. In most cases, the material is simply recycled with resulting production delays. In other cases, nonradioactive chemicals are discharged irrecoverably to seepage basins where some must be

TABLE 5-23. Density Functions for Criticality in Canyon Operations

Solution Systems			Solid Systems		
Total Fissions	Frequency	Cumulative Distribution	Total Fissions	Frequency	Cumulative Distribution
1×10^{19}	0.06	0.06	5×10^{18}	0.06	0.06
2×10^{18}	0.12	0.18	1×10^{18}	0.12	0.18
5×10^{17}	0.19	0.37	4×10^{17}	0.19	0.37
1×10^{17}	0.25	0.62	1×10^{17}	0.25	0.62
5×10^{16}	0.19	0.81	3×10^{16}	0.19	0.81
2×10^{16}	0.12	0.93	1×10^{16}	0.12	0.93
1×10^{16}	0.07	1.00	3×10^{15}	0.07	1.00

TABLE 5-24. Curies of volatile Fission Products Available
Following 2×10^{18} Fissions in Solution

Nuclide	Curies Produced*
^{83}Br	9.2×10^0
^{84}Br	7.0×10^1
^{85}Br	2.6×10^2
^{131}I	1.7×10^0
^{132}I	1.4×10^1
^{133}I	3.0×10^1
^{134}I	3.8×10^2
^{135}I	1.0×10^2
^{136}I	2.3×10^4
$^{83\text{m}}\text{Kr}$	5.1×10^0
$^{85\text{m}}\text{Kr}$	1.2×10^1
^{87}Kr	7.6×10^1
^{88}Kr	2.2×10^2
^{89}Kr	2.0×10^3
^{90}Kr	1.5×10^4
^{133}Xe	1.8×10^0
$^{135\text{m}}\text{Xe}$	1.1×10^2
^{135}Xe	3.2×10^1
^{137}Xe	4.9×10^3
^{138}Xe	1.4×10^3
Total	4.8×10^4

*Includes maximum quantity of significant isotopes (>1 Ci produced) which occur between 1 minute and 8 hrs after fission.

FIGURE 5-4. Cumulative Distribution Functions for Criticality in the Canyons

neutralized. Corrosion of equipment, contamination, radiation, and product degradation are the most significant consequences, however.

In order to quantify the risk of a failure or error, it is necessary to define not only the probability of the occurrence of such a failure or error, but the consequence. The prima facie consequence of a transfer error is the loss of control of a liquid containing radioactive materials. In general, transfer errors are reported as involving a certain mass of material rather than a certain number of curies of radioactivity. Therefore, it was necessary to quantify the error in terms of mass, and then superimpose radiological considerations by selecting an appropriate process stream.

Masses of materials involved in specific transfer errors were determined from the data bank and applied categorically to the error as a whole rather than by attempting to apply the data to the particular operation in which the error occurred. About 45% of the transfer error reports were explicit with regard to the mass of materials. The values were grouped primarily according to activity level; that is, canyon process material, cold chemicals, and water. A fourth group, ion exchange materials, was also used because the masses involved were generally an order of magnitude less than other canyon transfer errors.

Three statistical functions were calculated for each group of transfer errors; the mean and median mass of material transferred, and the 90% bounds. The results are shown in Table 5-25. Consequence data for both F- and H-Canyon have been combined because of the small difference between measured consequences for the two areas.

Airborne Releases. Detailed breakdowns of the quantities of radionuclides that would be released to the atmosphere due to transfer errors to outside facilities (211-F) and to the sump are presented in Tables C-6 and C-7. Details of the source term calculations are documented in Appendix A of the F-Canyon Systems Analysis (1).

In estimating release due to the airborne risk for a transfer error to the outside facilities, it was assumed that the probability is 10^{-2} that the receiving vessel will fail to contain the material that is erroneously transferred. Loss of confinement of liquids in the outside facilities, either by an overflow or spill, will result in release of liquid materials to the environment. Potential liquid-borne releases from the outside facilities are analyzed in Reference 50, based on 634 curies available.

Liquid Releases. Only two plausible mechanisms exist for the release of liquid materials from the canyon to the environment: a transfer error and a coil failure. In each case, additional failures of confinement outside the canyon would have to occur before such a release from the canyon would result in a release to the environment. Consequences of coil failures are discussed in Section 5.4.5.10.

For a liquid release, the mean loss due to transfer errors is 3170 lb of process materials containing 0.26 Ci/lb of actinides and 35.5 Ci/lb of fission products. The radionuclides transferred to 211-F are obtained from the

TABLE 5-25. Masses of Materials Involved in Transfer Errors

System	Mean, lb	Median, lb	90% Bounds, lb
Canyon process material	3,170	2,460	434 - 9,000
Chemicals	14,620	2,865	133 - 72,250
Water	36,290	2,920	252 - 290,000
Ion exchange material	280	94	16 - 2,720

isotopic fraction as described in Appendix A of the F-Canyon Systems Analysis (1). These consequences are summarized in Table 5-26 for various unit operations.

5.4.5.2 Overflow

Quantification of the consequences of an overflow was done in the same manner as that for transfer errors, as described in the previous section. The results are given in Table 5-27 for combined data from F- and H-Canyons.

A detailed breakdown of the radionuclides that would be released to the atmosphere due to process solution overflows to the sump is presented in Table C-8. Details of the source term calculations are documented in Appendix A of the F-Canyon Systems Analysis (1).

5.4.5.3 Leak

Leak data are seldom reported in a manner that allows the mass of material involved to be determined. For instance, leak rates are often reported, but not the duration or volume. Therefore, it is assumed for the purpose of conservative analysis that the mean consequence of a leak in ancillary operations is the release of one sump-full, or 623 lb of solution, calculated from standard canyon sump dimensions (10 ft³) and a solution density of 8.33 lb/gal (1). For process operations, when sump liquid levels are monitored and controlled more closely, the mean consequence is 263 lb of process material. The basis is 40% of the sump volume (the difference between the alarm point for the sump liquid-level detector and the normal sump heel level) and a solution specific gravity of 1.05.

A detailed breakdown of the radionuclides that would be released to the atmosphere due to process solution leaks to the sump is presented in Table C-9. Details of the source term calculations are documented in Appendix A of the F-Canyon Systems Analysis (1).

5.4.5.4 Chemical Addition Error

The greatest hazard of the chemical addition error is the initiation of an uncontrolled chemical or nuclear reaction as discussed in Section 5.4.4.2 or 5.4.4.3. Other consequences are associated with product contamination or material loss.

5.4.5.5 Pluggage

When pluggage occurs, the usual consequence is process delay. However, in some cases, the pluggage will result in a backup of materials to the extent that an overflow results. Loss of materials in the liquid-bearing systems that could result from pluggage are accounted for in the section on Overflow (Section 5.4.5.2). Pluggage in the off gas system is discussed in Section 5.1.5.9.

TABLE 5-26. Liquid Releases to 211-F due to Transfer Errors

Unit Operation	Frequency, Occurrences/hr	lb	Ci/lb Actinides	Ci/lb Fission Products	Ci	Release Rate, Ci/hr
Dissolving	1×10^{-6}	3170	0.26	35.5	1.1×10^5	1.1×10^{-1}
Head end	2×10^{-7}	3170	0.26	35.5	1.1×10^5	2.3×10^{-2}
Solvent Extraction						
First Cycle	4×10^{-7}	3170	0.26	35.5	1.1×10^5	4.4×10^{-1}
Second U Cycle	2×10^{-7}	3170	--	2.9×10^{-4}	9.2×10^{-1}	1.8×10^{-7}
Second Pu Cycle	8×10^{-7}	3170	--	2.7×10^{-1}	8.6×10^2	6.9×10^{-4}
Total	1×10^{-6}	--	--	--	--	4.5×10^{-2}
Evaporation						
HAW	5×10^{-7}	3170	1.3×10^{-3}	58.4	1.9×10^5	9.3×10^{-2}
LAW	5×10^{-7}	3170	1.6×10^{-5}	1.5×10^{-3}	4.8×10^0	2.4×10^{-6}
Ion exchange	8×10^{-7}	280	1.3×10^{-3}	58.4	1.6×10^4	1.3×10^{-2}
Waste disposal	2×10^{-6}	3170	4.0×10^{-5}	58.4	1.9×10^5	3.7×10^{-1}

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TABLE 5-27. Masses of Materials Involved in Overflows

System	Mean, lb	Median, lb	90% Bounds, lb
Canyon process material	1,860	575	43 - 7,770
Chemicals	3,570	1,050	50 - 14,800
Water	2,280	2,700	300 - 5,000
Ion exchange material	550	300	15 - 2,000

A more significant consequence is radiation exposure to personnel during removal of the pluggage. Forty-eight reports of high radiation resulting from pluggage of hot canyon samplers were statistically analyzed.

Mean open door exposure rate	1700/276 mrad/mr/hr
Median open door exposure rate	1000/100 mrad/mr/hr
90% range	5000/1000 to 136/15 mrad/mr/hr

The listed numbers indicate open window (beta-gamma)/closed window (gamma) readings. Much lower values are recorded for the warm canyon sample aisle for a mean value of 16/6 mrad/mr/hr.

5.4.5.6 Siphoning

The likely consequence of siphoning would be to transfer liquid from one vessel to another with no loss of radioactive material outside the primary piping and vessel system. The risk is concluded to be nil.

5.4.5.7 Processing Short-Cooled Fuels

Accidental processing of short-cooled fuels will result in a release of about 0.5% of the iodine charged to the dissolver. This resulted in a release of 1.65 Ci of the 530 Ci of ^{131}I that was charged. The 1.65 Ci that was released from F-Area is assumed to be applicable to any potential errors; thus the release rate is 7×10^{-6} Ci/hr.

5.4.5.8 Ruthenium Volatilization

About 20 Ci of ^{106}Ru were released in a single event from the stack from the dissolver. The average release is assumed to be 7 Ci. Such releases are generally reported as annual routine releases due to the effectiveness of the vessel vent system and the sand filter.

5.4.5.9 Iodine Reactor Failure

No significant releases of iodine have been reported as a result of failures in the iodine reactors. During changeout of the Berl saddles from the cartridges, a radiation field of 4 rad/0.4 rem/hr at 30 cm is typical.

5.4.5.10 Coil and Tube Failure

In most cases, these failures release no detectable activity to the water return systems, because the leaking tubes or coils were pressurized so that water (or steam) leaked into the vessel. The failures were detected by increasing liquid levels in the vessels or by decreasing specific gravities in the solutions.

Sometimes, activity is detected by a monitor, and the water is diverted to a retention basin with no detectable release to Upper Three Runs Creek or Four Mile Creek. Diversions of segregated water containing activity have occurred at an overall frequency of 3.5×10^{-4} /hr. Circulated water has been diverted with a frequency of 6.8×10^{-5} /hr. The median activity (Table 5-28) is about 0.2 Ci per event and the mean value is 17 curies (24,43,50,51).

In 1968, about 100 mCi of fission product activity was released to Four Mile Creek before the recirculated cooling water was diverted following a leak in the HAW evaporator. About 2 Ci were sent to the retention basin from this leak. Through 1980, this type of release to the creek was a rare event. However, in April, July, and August of 1981, the monthly release guides were exceeded as a result of F-Canyon evaporator leaks to the segregated water system. Thus, the consequences of coil and tube failures (i.e., releases to Four Mile Creek) appear to be increasing, while the frequency is decreasing slightly (1).

Releases in excess of 40 Ci to a water system are expected to occur at a frequency no more than 3.4×10^{-6} /hr (24,51).

As discussed in Section 5.1.5.10, coil failures can allow process materials to escape the confines of the canyon via the segregated water system or the circulated cooling water system. The mean frequency for all releases from the canyon process operations due to coil failures is 1.2×10^{-4} /hr; the mean release is 17 Ci. Radionuclide releases to the retention basin from coil failures in various unit operations are shown in Table C-10.

Activity in either of these water systems does not constitute a release to the environment. The circulated cooling water system is normally monitored and returned to the cooling tower for reuse. The segregated water system is normally monitored and sent via a discharge ditch to Four Mile Creek. Radioactive contamination in either system is diverted to the retention basin, either automatically or manually.

As discussed above, failure of the monitoring or diversion system is necessary to release radioactive material to the cooling tower or to the environment via Four Mile Creek. The overall release rate is 2.7×10^{-4} Ci/hr, practically all of which is associated with Four Mile Creek (1). The fault tree for release of activity to Four Mile Creek is shown in Figure B-9. Radionuclide releases to Four Mile Creek from coil failures in various unit operations are based on the release to the retention basin and the probability that the water is not diverted (0.04) as shown in Table C-11.

Liquid releases to the retention basin and Four Mile Creek are summarized in Table 5-29.

5.4.5.11 Waste Header Failure

Analysis of the consequences of a waste header failure in the tank farm is presented in Reference 52.

For a sudden release between the canyon and the waste farm, radionuclide release is assumed to be 150 gal of fresh waste at a concentration of 140

TABLE 5-28. Total Activity Release per Coil Leak (F- and H-Canyons)

Activity Released, Ci	Frequency	Cumulative Distribution
300	0.02	0.02
2	0.04	0.06
1.2	0.06	0.12
0.8	0.08	0.20
0.5	0.10	0.31
0.3	0.12	0.43
0.2	0.14	0.57
0.16	0.12	0.69
0.13	0.10	0.80
0.10	0.08	0.88
0.08	0.06	0.94
0.07	0.04	0.98
0.007	0.02	1.00

TABLE 5-29. Liquid Releases To Retention Basin and Four Mile Creek due to Coil and Tube Failure

Operation	% Coil and Tube Failures*	Frequency/hr	Release, Ci	Release Rate, Ci/hr	
				Basin	Four Mile Creek**
Decontamination	--	1.5×10^{-6}	0.2	3.0×10^{-7}	1.2×10^{-8}
Dissolving	6	7.2×10^{-6}	16.9	1.2×10^{-4}	4.9×10^{-6}
Head end	7	8.4×10^{-6}	16.9	1.4×10^{-4}	5.7×10^{-6}
Solvent extraction	13	1.6×10^{-5}	16.9	2.6×10^{-4}	1.1×10^{-5}
Evaporation	65	7.8×10^{-5}	17.2	1.3×10^{-3}	5.4×10^{-5}
Ion exchange	2	2.4×10^{-6}	17.2	4.1×10^{-5}	1.7×10^{-6}
Waste disposal	7	8.4×10^{-6}	17.2	1.4×10^{-4}	5.8×10^{-6}

*See Table 5-15. Mean frequency for all canyon releases is 1.2×10^{-4} /hr.

**Based on the frequency of release to the retention basin times the probability that the water is not diverted (0.04) and distribution of coil and tube failures due to various unit operations.

Ci/gal, or 21,000 Ci. For a slow release, the mean leak release is assumed to be 7 Ci (51). The release to Four Mile Creek is 3×10^{-5} Ci/hr.

The consequences of failures of waste headers inside the canyon are included in Section 5.4.5.3.

The consequences of partial pluggage of waste headers are included in Section 5.4.5.5.

5.4.6 Residual Release Events

Residual release events may release the residual radioactivity outside the process equipment through the confinement barriers. The most significant consequence is the exposure of operating personnel.

5.4.6.1 Suckback

Seventeen cases from the F-Area hot canyon of gamma intensity at varying distances were reported. The data from both F- and H-Area were normalized to a distance of three inches to yield a calculated mean of 175 R/hr. Intensities up to 500 R/hr at 5 in have been reported.

Most suckbacks have been controlled by the installation of an automatically timed air blow sequence on gang valves; however, at least two have occurred in F-Canyon since 1975.

5.4.6.2 Air Reversal

Air reversals represent a loss of a confinement barrier. Where previous incidents have left contaminated particulates in a zone, air reversals can result in airborne activity migration, potentially to occupied areas. The most undesirable consequence is an uptake by a worker. Uptakes were experienced in two of the 37 cases studied (5%). Skin contaminations or contamination of a work area are less severe consequences, usually resulting only in lost work time during cleanup. Many air reversals result in no detectable airborne activity or contamination spread. Air reversal data are given in Table 5-30.

5.4.7 Chemical Hazards

Leaks are the largest contributor to outside releases of hazardous chemicals (3.2×10^{-5} /hr) and of hazardous liquid (1.1×10^{-4} /hr) or vapor (3.7×10^{-5} /hr) releases to occupied areas (see Tables 5-16 and 5-17). Because of uncertainty in quantifying the consequences of leaks, it is conservatively assumed that each release to an occupied area constitutes a potential exposure of a worker to a hazardous material at or above the acceptable limit:

- The frequency for potential chemical contamination of personnel due to liquids is 1.1×10^{-4} /hr. These incidents may include effects extending from loss of outer clothing only, to burns to

TABLE 5-30. Consequences of Air Reversals

Airborne activity data reported, cases	12
No measureable airborne activity, cases	21
Inadequate data reported, cases	<u>4</u>
Number of air reversals, total cases	37
Mean airborne activity, RCG	31
Median airborne activity, RCG	<0.1
90% bounds, RCG	<0.1-132

RCG = Radiation Control Guide:

1×10^{-9} $\mu\text{Ci}\beta\text{-}\gamma/\text{cc air}$

2×10^{-12} $\mu\text{Ci } \alpha/\text{cc air}$

the skin, or to absorption through the skin with medical effects depending on the protective clothing in use, the duration of the exposure, and the mitigating actions taken afterwards.

The SRP minor injury and unusual incident records were compared to the CFP data from the 200-Area data bank. The probability that (given a potential exposure to CFP hazardous liquids) an injury occurs is calculated to be 0.103 for leaks and 0.16 for the other three initiating events (overflow, transfer error, and uncontrolled reaction). The frequency for hazardous liquid injury is therefore:

$$(8.0 \times 10^{-5}/\text{hr})(0.103) = 8.2 \times 10^{-6}/\text{hr}$$

$$(3.0 \times 10^{-5}/\text{hr})(0.16) = 4.8 \times 10^{-6}/\text{hr}$$

$$\text{Total} = 1.3 \times 10^{-5}/\text{hr}$$

The most likely injury is an acid burn to the skin.

- The frequency for potential exposure to hazardous vapors in an occupied area is $3.7 \times 10^{-5}/\text{hr}$ at or above the AEL for that vapor. The potential for chemical uptakes and for illness depends on the safety measures taken before the exposure, the duration of the exposure, and the mitigating actions taken afterwards.

Neither the SRP records nor the data bank contain a case of injury from hazardous vapor uptake from any initiating event in CFP. Therefore, given a potential exposure to hazardous vapors, the probability that a hazardous vapor uptake occurs in CFP is assumed to be less than 0.15, and the frequency for hazardous vapor injury in CFP is $<5.6 \times 10^{-6}/\text{hr}$. The most likely event is inhalation of hazardous vapor above the AEL released either by an uncontrolled reaction or by leaks during maintenance operations.

5.4.8 Support Systems

Consequences of failure of ancillary operations and equipment are discussed in the following sections. Support system failures are considered with the analysis of the process operations and are not analyzed as separate individual accident initiators.

5.4.8.1 Shop and Decontamination Facilities

Consequences associated with operation of shop and decontamination facilities are included in Sections 5.4.4, 5.4.5, and 5.4.7.

5.4.8.2 Cell Covers

The consequences of dropping a cell cover include damage to the cover itself and damage to other equipment. Cell covers can also become highly contaminated during use and present radiation exposure problems. Air reversals can also result if too many covers are removed.

5.4.8.3 Gang Valve Corridors

The consequences of gang valve failure are failure to transfer process materials, excessive transfer, local contamination, and suckback.

Valves other than gang valves that have failed in the corridors include steam block valves, condensate header valves, relief valves, instrument air valves, and deluge valves. The most common are the steam valves. Inability to isolate sections of lines or leaks in the corridor are the usual consequence.

Corrosion of concrete around embedded pipe has occurred to some extent as a result of leaks in the piping. Some embedded piping has been abandoned as a result of loss of the structural integrity.

Fires are relatively rare in the gang valve corridors. Only three are recorded and these occurred prior to 1963. Two involved welding operations and the other was an electrical short that ignited cloth wipes.

Because of the abundance of thermally hot lines in the corridors, several minor personnel burns have occurred. None of these were serious.

Leaks include steam, condensate, and cooling water from the gang valves, ceiling nozzles, steam headers, and condensate headers. Many of these are multiple leaks in a single system, especially those involving the steam and condensate headers. Consequences are contamination, airborne activity, corrosion, and personnel injury potential.

Significant cases of surface contamination to 20×10^6 dis/min alpha and 80,000 counts/min beta-gamma have been measured. The usual sources include seepage through ceiling cracks, pipe leaks, nozzle leaks, maintenance operations, and embedded pipe liner installations.

Normally, admission to the gang valve corridors does not require the use of respiratory protection. Occasionally, however, airborne activity does occur following leaks or maintenance work. Levels to about 1500 RCG have been recorded, but the usual level is much lower.

Significant radiation exposure can occur as a result of spills, suckback, seepage through conduit in the abandoned "fireye" system, or from coil leaks into the condensate header. The calculated mean gamma radiation intensity as a result of suckback is 175 R/hr at 3 in. Seepage through the "fireye" conduit produces the same order of magnitude field; however, coil failure radiation intensities are about three orders of magnitude less.

5.4.8.4 Cranes

Consequences of crane failures include impairment of crane functions, damage to other equipment, and hazards to operating personnel.

Impairment of crane functions essentially stops all cask unloading, fuel charging, equipment changes, and maintenance. Wheel bearing failures result in repair times of one to 21 days for wheel replacement. Misalignment of wheels results in excessive wear to the rails, rail supports, and wheels. Brake malfunction results in the inability to control drift in the hoists or the potential for not being able to control crane travel. Cables are routinely inspected for fraying or kinking; however, breaking under load occasionally occurs. Breakdown of communications problems with the hot cranes are not significant; warm crane communications continue to be a problem. Impact wrenches, which are the most used remote tool, are high maintenance items. Electrical equipment has experienced one fire in the equipment and considerable decontamination damage as a result of a fire within the canyons.

Damage may be inflicted on other equipment by crane misoperations. No major damage has resulted from dropping incidents. Handling errors have resulted in damage to "fireeyes," limit switches, jumpers, yokes, air conditioners, and radiation monitors. Other operating errors which can result in damage include incorrect piping or jumper installation, failure to check clearances, failure to apply brakes, improperly positioned equipment, and bumping fixed equipment.

Damage to crane ancillaries has also occurred on several occasions. Spalling of concrete beneath the crane rails occurred in 1979. Acid fumes have caused corrosion to runway emergency doors to the extent that they had to be pried open. Shielding doors and removable rails, especially the rails, require frequent maintenance. Other failures include blown fuses, sticking door, lost keys, timer switches, and bent lifting yokes.

Hazards to personnel include radiation, contamination, and personal injury. Radiation exposure and personal injury occur primarily during maintenance. Airborne and surface contamination in the crane cabs are caused primarily by leaks. Typical contamination levels are 1000 counts/min.

5.4.8.5 Sample Aisles

The most significant of the causes of failure to sample is pluggage. Normally, pluggage is removed by attaching compressed air fittings from the blow-down cart or by changing the sampler needles. If these measures fail to relieve the pluggage, nitric acid is poured through the system. Occasionally, however, the sampler and/or the jet must be removed from the system and rodged.

About 10% of the failures of the sample aisle hoists result in all or part of the hoist falling to the aisle floor which could result in personnel injury. The sample aisle elevator also fails. Although this does not present any hazard to operating personnel, it does disable the normal route for removing samples from the aisle.

Significant surface or airborne contamination cases also occur. Normally respirator protection is not required to enter the aisle, but is required during sampling, maintenance, or when contamination is present. Nasal contamination cases, however, occur about every other year.

Radiation fields at the open door to samplers in excess of 1 R/hr occur at a rate of about once every three weeks. These usually are the result of sampler pluggage, leaks, or broken sample vials.

Two fires have been recorded for the aisle. Both were the result of welding operations and both during the early years of canyon operation.

Fumes from leaks in the cold chemical storage area can migrate into the sample aisle. Two recent cases resulted in 10 and 25 ppm of nitric acid fumes. The TLV for nitric acid is 2 ppm.

Three injuries as a result of a whipping compressed air line, a cut while sampling, and a nitric acid burn are recorded. No contamination resulted.

5.4.8.6 Steam Distribution

Malfunctions in the powerhouse do not necessarily reduce the steam supply to less than the demand because peak requirements are not continuous. Failure of the 325 psi header has resulted in steam outages or reduced steam supply for short periods. No failures in the 221-F supply header were severe enough to hamper steam supply significantly, nor to present a safety hazard to personnel. Severe condensate header leaks are considered a nuisance rather than a hazard.

It is possible to cause severe radiation or contamination through the steam system. Suckbacks and coil leaks are the most common. Isolated cases of process vessel overflow have occurred as a result of an accumulation of steam condensate: one as a result of a valving error, and another due to a valve leak. A severe steam leak through a gang valve caused liquid in one process vessel to be accidentally jetted to another vessel.

5.4.8.7 Compressed Air

Several significant difficulties could arise from compressor failure. Should a canyon vessel coil leak develop, the likelihood of getting activity into the segregated cooling water is high if the plant air system fails, because the Cash air controller is on this system. Instrument air failure negates the effectiveness of many instruments but does not in itself cause a release of activity, nor does it prevent emergency shutdown of operations. Failure of the process air system can result in suckback as discussed in Section 5.4.6.1.

Failure of the breathing air compressors does not present a direct hazard to personnel because a back-up system is available from cylinders connected to a manifold. Contents of breathing air cylinders are analyzed before disbursement from Stores, and samples from the manifold system at the 400-Area are analyzed weekly; however, it is possible to introduce several undesirable gases into the system prior to analysis. Compressed Gas Association (CGA)

connections for nitrous oxide, helium, carbon dioxide, argon with 20% oxygen, and nitrogen with 18.65% oxygen will mate with the breathing air connection, even though some leakage would be likely. It is also possible for an incorrect gas to be charged to a breathing air cylinder. No probabilities have been determined for these occurrences.

5.4.8.8 Electrical Distribution

Failure of the electrical distribution system does not present a direct hazard to personnel. Consequences of failure to supply power to a piece of equipment are considered with each unit operation.

5.4.8.9 Cooling Water Distribution

Failure of the cooling water system should not imply that the contents of process vessels will immediately boil away. Under most circumstances, adequate time is available to make necessary repairs. Consequences of failure to supply cooling water are considered with the appropriate unit operation.

5.4.9 Engineered Safety Features

Consequences of failures of engineered safety features are discussed in the following sections.

5.4.9.1 Ventilation System

Analyses of stack releases were made on the basis of perturbations in canyon processes. It is concluded that the loss of radioactive material because of nonenergetic occurrences, such as leaks, do not significantly affect stack releases. The medium energetic occurrences, such as boiloff, boil over, and eruption, release less than one-half of one percent of the vessel contents to the sand filter intake. Because safety-related occurrences in canyon processes do not contribute significantly to above-normal releases, no attempt is made to correlate events with stack releases. Recognized process events that can cause small stack releases are criticality, processing short-cooled materials, and abnormally high temperatures.

Significant measurable releases to the atmosphere from the stack are caused by perturbations in the process vessel vent system that ventilates the process vessels, and the canyon exhaust system which exhausts air from regions over and around equipment in the warm and hot canyons. These two streams combine before passing through the sand filter. Each of these systems contributes about 50% of the radioactivity in the input air to the sand filter.

Perturbations in the canyon ventilation system or in the control air system can result in air reversals. Consequences of air reversals were discussed in Section 5.4.6.2. Median airborne activity levels due to local air reversals are less than 0.1 RCG in the railroad tunnel, center section, or crane maintenance area.

5.4.9.2 Water Return and Diversion

Consequences associated with the seven mechanisms identified for loss of liquid borne activity into plant streams are included in Figure B-9.

Overall, the rate of having activity escape to plant streams through the cooling water system is 2.7×10^{-4} Ci/hr, about 95% of which is associated with release to Four Mile Creek.

5.4.9.3 Seepage Basins

A complete analysis of the seepage basins is provided in Reference 25.

5.4.9.4 Secondary Confinement for Liquids

Gamma radiation intensities in personnel areas due to escape of liquids through expansion joints generally are about 10% of that caused by suckbacks in the hot gang valve corridor and second level. However, liquid emanating from the joints is unconfined except in specific locations known to be especially vulnerable, such as in the Section 9 personnel tunnel. Radiation levels from negligible to 500 R/hr at 3 in have been recorded. Alpha contamination to 6×10^5 dis/min has been reported.

Limited information is available regarding radioactivity in the soil near F-Canyon. Water samples from shallow wells near the periphery of the facility have shown low levels of beta-gamma activity (3 to 5 times normal background levels) from a well near the joint between Sections 8 and 9. Similar levels of activity have occasionally been detected in the well near the north end of F-Canyon. Deep wells near the periphery and in the groundwater below the facility have not shown levels above normal background indicating little if any transport of beta-gamma activity to the nearby surface streams. One deep well near the south end of F-Canyon has shown some contamination; however, analysis has shown that the contamination is largely depleted uranium with the near-by A-Line facility the likely source (53).

Activity migration through failed embedded instrument conduit running beneath the cell floors is causing radiation levels in the first level and in the gang valve corridor in the same range as for expansion joints. Lead shielding has been required in some cases.

5.4.9.5 Pressure Regulators

The scenario for a large (>40 Ci) release to the segregated water system requires that a coil or tube leak occurs at the same time as the Cash regulator failure (24). That is, the release is initiated by the leak, and the regulator failure enables this incident to result in measurable consequences.

5.4.9.6 Neutron Monitors

Consequences of neutron monitor failures are included in the fault tree analysis of the solvent extraction operations (Section 5.3.4.3).

5.5 RADIOLOGICAL RISKS

The radiological consequences and risk due to accidents for F-Canyon operations have been separated into three categories:

- Natural phenomena (Section 5.5.1)
- Externally induced failures (Section 5.5.2)
- Process related occurrences (Section 5.5.3)

Risk is defined as the product of the radiological consequence of an event and the expected frequency of the event. The event frequencies were derived in Section 5.3 for the various accidents. Radiological consequences are calculated for three types of effects: 1) overall offsite 50-year dose commitment for people within 50 miles, 2) onsite 50-year dose commitment to exposed employees, and 3) dose to the maximally exposed individual at the site boundary (a child). These radiological consequences were calculated utilizing the methods described in Section 5.2.5.1 (atmospheric releases) and Section 5.2.5.2 (liquid releases).

Doses to various body organs for the offsite population, onsite population, and maximally exposed individual as a result of radionuclide inhalation are presented in Appendix D. The organs considered besides the total body dose are the G.I. tract, bone, liver, kidney, thyroid, and lung. These dose calculations were based on the source terms developed in Section 5.4 and presented in detail in Appendix C.

5.5.1 Risks Due to Natural Phenomena

For natural phenomena, the type of event leading to significant release is a seismic event (earthquake). Earthquake frequency and radionuclide release to the environment are summarized in Table 5-31. As discussed in Section 5.4.1.2, the release path is via leakage through cracks (probably at joints) to the ground and subsequent release to the atmosphere. With respect to radionuclide distribution for consequence calculations, a distribution equivalent to that for externally induced failures was used. Calculated risks for the various unit operations via ground airborne release shown in Table 5-31. Detailed dose calculations are shown in Tables D-1 to D-3.

5.5.2 Risks Due to Externally Induced Failures

For externally induced failures, the type of event leading to significant release is an impact. Impact frequency and radionuclide release to the environment are summarized in Table 5-32. As discussed in Section 5.4.2.1, the release path is via liquid release to the sump and subsequent airborne

TABLE 5-31. Risks Due to Natural Phenomena for F-Canyon Unit Operations*

Unit Operation	Frequency, /yr	Release, Ci	Risk		
			Offsite Population, person-rem/ yr	Onsite Population, person-rem/ yr	Offsite Maximum Individual, rem/yr
Dissolver	2.0E-04	2.5E-01	5.9E-06	1.9E-06	2.3E-09
Head End	2.0E-04	2.3E-01	5.4E-06	1.8E-06	2.1E-09
First Cycle	2.0E-04	2.3E-01	5.4E-06	1.8E-06	2.1E-09
2nd U Cycle	2.0E-04	5.4E-11	1.1E-14	3.7E-15	3.9E-18
2nd Pu Cycle	2.0E-04	1.2E-06	3.0E-08	1.0E-08	9.7E-12
HAW Evaporator	2.0E-04	4.1E-01	2.4E-06	6.6E-07	1.4E-09
LAW Evaporator	2.0E-04	1.4E-08	6.4E-13	2.2E-13	2.3E-16
Ion Exchange	2.0E-04	3.8E-01	2.3E-06	6.1E-07	1.3E-09
High Heat Waste	2.0E-04	4.1E-01	2.4E-06	6.6E-07	1.4E-09
Total for all Unit Operations			2.4E-05	7.4E-06	1.0E-11

*Event leading to significant release is an MM VIII earthquake.

TABLE 5-32. Risks Due to Externally Induced Failures for F-Canyon Unit Operations*

Unit Operation	Frequency, /yr	Release, Ci	Risk		
			Offsite Population, person-rem/ yr	Onsite Population, person-rem/ yr	Offsite Maximum Individual, rem/yr
Dissolver	8.8E-03	2.5E+00	1.9E-03	2.1E-04	2.8E-07
Head End	8.8E-03	2.3E+00	1.7E-03	1.9E-04	2.6E-07
First Cycle	8.8E-03	2.3E+00	1.7E-03	1.9E-04	2.6E-07
2nd U Cycle	8.8E-03	5.3E-10	3.4E-12	3.9E-13	4.8E-16
2nd Pu Cycle	8.8E-03	1.2E-05	9.5E-06	1.1E-06	1.2E-09
HAW Evaporator	8.8E-03	4.0E+00	7.9E-04	7.3E-05	1.7E-07
LAW Evaporator	8.8E-03	1.4E-07	2.0E-10	2.3E-11	2.8E-14
Ion Exchange	8.8E-03	3.7E+00	7.2E-04	6.7E-05	1.6E-07
High Heat Waste	8.8E-03	4.0E+00	7.9E-04	7.3E-05	1.7E-07
Total for all Unit Operations			7.6E-03	8.0E-04	1.3E-06

*Event leading to significant release is impact.

release from the 200-ft stack. Calculated risks for the various unit operations are shown in Table 5-32. Detailed dose calculations are shown in Tables D-4 to D-6.

5.5.3 Risks Due to Process-Related Occurrences

For process-related occurrences, various events can lead to airborne releases at ground level, airborne releases from the stack, and liquid releases to Four Mile Creek. Accident frequency and radionuclide release to the environment are summarized in Table 5-33. As discussed in Sections 5.4.3 to 5.4.6, the releases can be grouped into categories that relate the release energy and the potential for release outside the confinement barrier. The categories which include significant releases are medium and low energetic events. Calculated risks for the various unit operations are shown in Table 5-33. Detailed results for dose calculations are shown in Tables D-7 to D-15 for medium energetic events and Tables D-16 to D-30 for low energetic events.

5.5.4 Summary of Risks from Accidents

Risks are summarized in Table 5-34 for all accidents and in Table 5-35 for all unit operations. Table 5-34 shows the total risk for each accident from all unit operations. Table 5-35 shows the total risk for each unit operation from all accidents. The risk in terms of whole body dose is 1.7×10^{-1} person-rem/yr for the offsite population and is 2.0×10^{-2} person-rem/yr for the onsite population. The maximum calculated risk to an offsite individual at the site boundary is 3.9×10^{-5} rem/yr. Radiological risks from accidents are summarized in Table 5-36.

5.5.5 Risks to Operating Personnel

Operating personnel in F-Canyon are normally exposed to certain radiological and nonradiological risks both from normal operations and as a result of potential accidents. A concerted effort is made to maintain these risks to as low as reasonably achievable as discussed in Section 4.9. Radiological risks include direct exposures to penetrating radiation and assimilation of radioactive materials. Exposure of operating personnel to radiological risks due to potential conditions uniquely different from normal industrial practice also occurs. Nonradiological risks include chemical exposure and toxicity from normal industrial hazards such as improper handling of tools and equipment. The hazards and risks associated with normal operation are discussed in Section 4.9. There are several safety-related occurrences which can result in the potential radiological exposure of operating personnel in addition to the normal exposures. These occurrences are discussed in detail in Section 5.4 and Reference 1, and summarized below.

Radiation dose calculations from a postulated nuclear incident of 10^{18} fissions show that the most exposed areas are the warm canyon gang valve corridor, the warm canyon crane, and the maintenance bridge. The radiation dose in the gang valve corridor directly opposite the mixer-settlers is 75 rem, but less than 0.1 rem if the radiation beam reaching the receptor would subtend a 30° angle with a 2-ft concrete wall. Radiation doses in the warm

TABLE 5-33. Risks Due to Process Related Occurrences for F-Canyon Unit Operations

Unit Operation	Accident Category	Accident	Risk			
			Frequency, /yr	Release, Ci	Offsite Population, person-rem/ yr	Offsite Population, Maximum person-rem/ Individual, rem/yr
Decontamination	Low Energetic (Airborne)	Transfer Error to 211-F	2.6E-05	9.9E-05	4.4E-10	1.2E-10
		Transfer Error to Sump	3.3E-03	4.8E-07	2.0E-10	1.9E-11
		Overflow to Sump	4.4E-03	2.8E-07	1.6E-10	1.5E-11
		Leak to Sump	2.2E+00	8.9E-08	2.5E-08	2.4E-09
		Total (Airborne)			2.6E-08	2.5E-09
Dissolver	Medium Energetic (Airborne)	Coil Failure (Liquid)	5.3E-04	2.0E-01	5.1E-06	4.6E-10
		Total (Airborne)			2.6E-08	2.5E-09
		Uncontrolled Reaction	2.6E-01	2.3E-01	9.2E-03	1.0E-03
		Criticality	1.1E-06	4.8E-04	1.8E-06	1.6E-06
		Total (Airborne)			9.2E-03	1.0E-03
Head End	Low Energetic (Airborne)	Transfer Error to 211-F	1.1E-04	5.4E-01	6.6E-04	2.2E-04
		Transfer Error to Sump	3.9E-02	2.6E-01	8.9E-04	9.9E-05
		Overflow to Sump	4.4E-02	1.5E-01	5.8E-04	6.4E-05
		Leak to Sump	1.1E+00	2.2E-02	2.1E-03	2.4E-04
		Processing Short-Cooled Fuel Ruthenium Volatilization	3.5E-02	1.7E+00	2.7E-03	2.7E-04
Head End	Medium Energetic (Airborne)	Ruthenium Volatilization	5.3E-02	7.0E+00	8.5E-03	7.5E-04
		Total (Airborne)			1.6E-02	1.6E-03
		Coil Failure (Liquid)	2.5E-03	1.7E+01	2.1E-03	1.9E-07
		Total (Airborne)			2.5E-02	2.7E-03
		Uncontrolled Reaction	4.4E-02	2.1E-01	1.6E-03	1.7E-04
Head End	Low Energetic (Airborne)	Criticality	1.8E-05	4.8E+04	3.0E-05	2.7E-05
		Total (Airborne)			1.6E-03	2.0E-04
		Transfer Error to 211-F	1.8E-05	4.8E+01	9.8E-05	3.2E-05
		Transfer Error to Sump	6.6E-03	2.4E-01	1.3E-04	1.5E-05
		Overflow to Sump	4.4E-02	1.4E-01	5.2E-04	5.8E-05
Head End	Medium Energetic (Airborne)	Leak to Sump	7.0E+00	1.9E-02	1.2E-02	1.3E-03
		Total (Airborne)			1.2E-02	1.4E-03
		Coil Failure (Liquid)	2.9E-03	1.7E+01	2.4E-03	2.2E-07
		Total (Airborne)			1.4E-02	1.6E-03
		Uncontrolled Reaction				

TABLE 5-33. Risks Due to Process Related Occurrences for F-Canyon Unit Operations (Continued)

Unit Operation	Accident Category	Accident	Frequency, /yr	Release, Ci	Risk	
					Offsite Population, person-rem/ yr	Offsite Maximum Individual, rem/yr
First Cycle	Medium Energetic (Airborne)	Fire	6.1E-04	4.6E-01	2.7E-04	3.1E-05
		Uncontrolled Reaction	2.9E-03	3.0E-01	4.9E-05	5.2E-06
		Criticality	1.6E-03	4.8E-04	2.7E-03	2.4E-03
		Total (Airborne)			3.0E-03	2.5E-03
						4.4E-06
	Low Energetic (Airborne)	Transfer Error to 211-F	3.1E-05	4.8E-01	1.7E-04	5.7E-05
		Transfer Error to Sump	1.1E-02	2.4E-01	2.3E-04	2.6E-05
		Overflow to Sump	2.0E-01	1.4E-01	2.4E-03	2.7E-04
		Leak to Sump	1.1E-01	1.9E-02	1.9E-02	2.1E-03
		Total (Airborne)			2.2E-02	2.5E-03
	Coil Failure (Liquid)		1.9E-03	1.7E-01	1.5E-03	1.4E-07
		Total (Airborne)			2.5E-02	4.9E-03
						7.7E-06
2nd U Cycle	Medium Energetic (Airborne)	Fire	6.1E-04	3.6E-06	2.7E-08	3.1E-09
		Uncontrolled Reaction	2.9E-03	1.4E-07	4.8E-09	5.5E-10
		Total (Airborne)			3.2E-08	3.6E-09
						4.4E-12
	Low Energetic (Airborne)	Transfer Error to 211-F	1.8E-05	3.9E-09	7.2E-14	2.4E-14
		Transfer Error to Sump	6.9E-03	1.9E-11	9.9E-14	1.1E-14
		Overflow to Sump	7.7E-02	1.1E-11	8.1E-18	9.6E-19
		Leak to Sump	3.1E-00	1.6E-12	3.6E-12	4.0E-13
		Total (Airborne)			3.7E-12	4.4E-13
	Coil Failure (Liquid)		1.9E-03	1.4E-09	1.3E-13	1.1E-17
		Total (Airborne)			3.2E-08	3.6E-09
						4.4E-12
2nd Pu Cycle	Medium Energetic (Airborne)	Fire	6.1E-04	3.5E-03	2.3E-04	2.7E-05
		Uncontrolled Reaction	2.9E-03	1.3E-04	4.1E-05	4.9E-06
		Criticality	1.6E-03	4.8E-04	2.7E-03	2.4E-03
		Total (Airborne)			3.0E-03	2.5E-03
						4.4E-06
	Low Energetic (Airborne)	Transfer Error to 211-F	7.4E-05	2.5E-04	2.3E-06	8.0E-07
		Transfer Error to Sump	2.7E-02	1.2E-06	3.1E-06	3.6E-07
		Overflow to Sump	2.3E-01	7.2E-07	1.5E-05	1.8E-06
		Leak to Sump	9.6E-00	1.0E-07	8.9E-05	1.0E-05
		Total (Airborne)			1.1E-04	1.3E-05
	Coil Failure (Liquid)		1.9E-03	8.8E-05	8.1E-09	7.2E-13
		Total (Airborne)				

TABLE 5-33. Risks Due to Process Related Occurrences for F-Canyon Unit Operations (Continued)

Unit Operation	Accident Category	Accident	Frequency, /yr	Release, Ci	Risk		
					Offsite Population, person-rem/ yr	Onsite Population, person-rem/ yr	Offsite Maximum Individual, rem/yr
HAM Evaporator	Medium Energetic (Airborne)	Uncontrolled Reaction	6.3E-01	6.0E-01	1.0E-02	9.6E-04	2.3E-06
	Low Energetic (Airborne)	Transfer Error to 211-F	8.8E-05	1.4E-02	3.7E-04	1.0E-04	2.1E-07
		Transfer Error to Sump	3.3E-02	7.0E-01	5.2E-04	4.8E-05	1.2E-07
		Overflow to Sump	2.2E-01	4.1E-01	2.0E-03	1.9E-04	4.5E-07
		Leak to Sump	1.6E+01	5.8E-02	2.0E-02	1.9E-03	4.5E-06
	Total (Airborne)				2.3E-02	2.2E-03	5.3E-06
	Coil Failure (Liquid)		1.4E-02	1.7E+01	1.1E-02		1.0E-06
	Total (Airborne)				3.4E-02	3.2E-03	7.6E-06
	LAM Evaporator	Medium Energetic (Airborne)	Uncontrolled Reaction	1.6E-01	6.8E-06	1.3E-06	1.6E-07
Low Energetic (Airborne)		Transfer Error to 211-F	8.8E-05	4.8E-06	9.7E-11	3.3E-11	3.5E-14
		Transfer Error to Sump	3.3E-02	2.4E-08	1.3E-10	1.5E-11	1.9E-14
		Overflow to Sump	2.2E-01	1.4E-08	5.2E-10	5.9E-11	7.2E-14
		Leak to Sump	1.6E+01	2.0E-09	5.3E-09	6.0E-10	7.3E-13
Total (Airborne)				6.0E-09	7.1E-10	8.6E-13	
Coil Failure (Liquid)			1.4E-02	5.8E-07	3.9E-10		3.5E-14
Total (Airborne)				1.4E-06	1.6E-07	1.8E-10	
Ion Exchange		Medium Energetic (Airborne)	Fire Uncontrolled Reaction	7.9E-03 5.3E-01	9.1E-01 7.9E-01	1.9E-04 1.1E-02	1.8E-05 1.0E-03
	Total (Airborne)				1.1E-02	1.1E-03	2.5E-06
	Low Energetic (Airborne)	Transfer Error to 211-F	7.0E-05	1.3E+01	2.8E-05	7.4E-06	1.5E-08
		Transfer Error to Sump	2.6E-02	6.4E-02	3.8E-05	3.5E-06	8.4E-09
		Overflow to Sump	3.5E-01	2.2E-01	1.7E-03	1.6E-04	3.8E-07
		Leak to Sump	7.9E+00	7.6E-02	1.3E-02	1.2E-03	3.0E-06
	Total (Airborne)				1.5E-02	1.4E-03	3.4E-06
	Coil Failure (Liquid)		8.4E-04	1.7E+01	7.1E-04		6.3E-08
	Total (Airborne)				2.7E-02	2.5E-03	5.9E-06

TABLE 5-33. Risks Due to Process Related Occurrences for F-Canyon Unit Operations (Continued)

Unit Operation	Accident Category	Accident	Frequency, /yr	Release, Ci	Risk		
					Offsite Population, person-rem/ yr	Onsite Population, person-rem/ yr	Offsite Maximum Individual, rem/yr
High Heat Waste	Medium Energetic (Airborne)	Uncontrolled Reaction	4.4E-02	6.0E-01	7.2E-04	6.7E-05	1.6E-07
	Low Energetic (Airborne)	Transfer Error to 211-F	1.8E-05	1.4E+02	7.5E-05	2.0E-05	4.2E-08
		Transfer Error to Sump	6.6E-03	7.0E-01	1.0E-04	9.5E-06	2.3E-08
		Overflow to Sump	3.5E-01	4.1E-01	3.2E-03	3.0E-04	7.1E-07
		Leak to Sump	8.8E+00	5.8E-02	1.1E-02	1.1E-03	2.5E-06
	Total (Airborne)				1.5E-02	1.4E-03	3.3E-06
	Coil Failure (Liquid)		2.9E-03	1.7E+01	2.5E-03		2.2E-07
	Total (Airborne)				1.6E-02	1.4E-03	3.5E-06
	Medium Energetic (Airborne)	Fire			6.9E-04	7.6E-05	1.1E-07
		Uncontrolled Reaction			3.3E-02	3.3E-03	6.5E-06
		Criticality			5.4E-03	4.9E-03	8.7E-06
Total for all Unit Operations	Total (Airborne)				3.9E-02	8.3E-03	1.5E-05
	Low Energetic (Airborne)	Transfer Error to 211-F			1.4E-03	4.3E-04	6.2E-07
		Transfer Error to Sump			1.9E-03	2.0E-04	3.4E-07
		Overflow to Sump			1.0E-02	1.0E-03	2.1E-06
		Leak to Sump			7.8E-02	7.8E-03	1.5E-05
		Processing Short-Cooled Fuels			2.7E-03	2.7E-04	5.1E-07
		Ruthenium Volatilization			8.5E-03	7.5E-04	1.9E-06
	Total (Airborne)				1.0E-01	1.1E-02	2.1E-05
	Coil Failure (Liquid)				2.1E-02		1.8E-06
	Total (Airborne)				1.4E-01	1.9E-02	3.6E-05
	Total				1.6E-01	1.9E-02	3.8E-05

TABLE 5-34. Summary of Risks for F-Canyon Accidents for All Unit Operations

Accident		Risk		
		Offsite Population, person-rem/ yr	Onsite Population, person-rem/ yr	Offsite Maximum Individual, rem/yr
Natural Phenomena		2.4E-05	7.4E-06	1.0E-08
Externally Induced Failures		7.6E-03	8.0E-04	1.3E-06
Process Related Occurrences				
Medium Energetic (Airborne)	Fire	6.9E-04	7.6E-05	1.1E-07
	Uncontrolled Reaction	3.3E-02	3.3E-03	6.5E-06
	Criticality	5.4E-03	4.9E-03	8.7E-06
	Total (Airborne)	3.9E-02	8.2E-03	1.5E-05
Low Energetic (Airborne)	Transfer Error to 211-F	1.4E-03	4.3E-04	6.2E-07
	Transfer Error to Sump	1.9E-03	2.0E-04	3.4E-07
	Overflow to Sump	1.0E-02	1.0E-03	2.1E-06
	Leak to Sump	7.8E-02	7.8E-03	1.5E-05
	Processing Short-Cooled Fuels	2.7E-03	2.7E-04	5.1E-07
	Ruthenium Volatilization	8.5E-03	7.5E-04	1.9E-06
	Total (Airborne)	1.0E-01	1.1E-02	2.1E-05
Coil Failure (Liquid)		2.1E-02		1.8E-06
Total (Airborne)		1.4E-01	1.9E-02	3.6E-05
Total (Process Occurrences)		1.6E-01	1.9E-02	3.8E-05
Total for all accidents	Airborne	1.5E-01	2.0E-02	3.7E-05
	Liquid	2.1E-02		1.8E-06
	Total	1.7E-01	2.0E-02	3.9E-05

TABLE 5-35. Summary of Risks for F-Canyon Unit Operations for All Accidents

Unit Operation	Accident Category	Risk		
		Offsite Population, person-rem/ yr	Onsite Population, person-rem/ yr	Offsite Maximum Individual, mrem/yr
Decontamination	Process Occurrences (Airborne)	2.6E-08	2.6E-09	5.5E-12
	Process Occurrences (Coil Failures)	5.1E-06		4.6E-10
	Total (Airborne)	2.6E-08	2.6E-09	5.5E-12
	Total (Liquid)	5.1E-06		4.6E-10
	Total	5.1E-06	2.6E-09	4.6E-10
Dissolver	Natural Phenomena	5.9E-06	2.0E-06	2.3E-09
	Externally Induced Failures	1.9E-03	2.1E-04	2.9E-07
	Process Occurrences (Airborne)	2.5E-02	2.7E-03	4.6E-06
	Process Occurrences (Coil Failures)	2.1E-03		1.9E-07
	Total (Airborne)	2.7E-02	2.9E-03	4.9E-06
	Total (Liquid)	2.1E-03		1.9E-07
	Total	2.9E-02	2.9E-03	5.1E-06
Head End	Natural Phenomena	5.4E-06	1.8E-06	2.1E-09
	Externally Induced Failures	1.7E-03	1.9E-04	2.6E-07
	Process Occurrences (Airborne)	1.4E-02	1.6E-03	2.2E-06
	Process Occurrences (Coil Failures)	2.4E-03		2.2E-07
	Total (Airborne)	1.6E-02	1.8E-03	2.5E-06
	Total (Liquid)	2.4E-03		2.2E-07
	Total	1.8E-02	1.8E-03	2.7E-06
First Cycle	Natural Phenomena	5.4E-06	1.8E-06	2.1E-09
	Externally Induced Failures	1.7E-03	1.9E-04	2.6E-07
	Process Occurrences (Airborne)	2.5E-02	4.9E-03	7.7E-06
	Process Occurrences (Coil Failures)	1.6E-03		1.4E-07
	Total (Airborne)	2.7E-02	5.1E-03	8.0E-06
	Total (Liquid)	1.6E-03		1.4E-07
	Total	2.8E-02	5.1E-03	8.1E-06
2nd U Cycle	Natural Phenomena	1.1E-14	3.7E-15	3.9E-18
	Externally Induced Failures	3.4E-12	3.9E-13	4.8E-16
	Process Occurrences (Airborne)	3.2E-08	3.6E-09	4.4E-12
	Process Occurrences (Coil Failures)	1.3E-13		1.1E-17
	Total (Airborne)	3.2E-08	3.6E-09	4.4E-12
	Total (Liquid)	1.3E-13		1.1E-17
	Total	3.2E-08	3.6E-09	4.4E-12

TABLE 5-35. Summary of Risks for F-Canyon Unit Operations for All Accidents
(Continued)

2nd Pu Cycle	Natural Phenomena	3.0E-08	1.0E-08	9.7E-12
	Externally Induced Failures	9.5E-06	1.1E-06	1.2E-09
	Process Occurrences (Airborne)	3.1E-03	2.5E-03	4.4E-06
	Process Occurrences (Coil Failures)	8.1E-09		7.0E-13
	Total (Airborne)	3.1E-03	2.5E-03	4.4E-06
	Total (Liquid)	8.1E-09		7.0E-13
	Total	3.1E-03	2.5E-03	4.4E-06
HAW Evaporator	Natural Phenomena	2.4E-06	6.6E-07	1.4E-09
	Externally Induced Failures	7.9E-04	7.3E-05	1.7E-07
	Process Occurrences (Airborne)	3.4E-02	3.2E-03	7.6E-06
	Process Occurrences (Coil Failures)	1.2E-02		1.0E-06
	Total (Airborne)	3.4E-02	3.3E-03	7.8E-06
	Total (Liquid)	1.2E-02		1.0E-06
	Total	4.6E-02	3.3E-03	8.8E-06
LAW Evaporator	Natural Phenomena	6.4E-13	2.2E-13	2.3E-16
	Externally Induced Failures	2.0E-10	2.3E-11	2.8E-14
	Process Occurrences (Airborne)	1.4E-06	1.6E-07	1.8E-10
	Process Occurrences (Coil Failures)	3.9E-10		3.5E-14
	Total (Airborne)	1.4E-06	1.6E-07	1.8E-10
	Total (Liquid)	3.9E-10		3.5E-14
	Total	1.4E-06	1.6E-07	1.8E-10
Ion Exchange	Natural Phenomena	2.3E-06	6.1E-07	1.3E-09
	Externally Induced Failures	7.2E-04	6.7E-05	1.6E-07
	Process Occurrences (Airborne)	2.7E-02	2.5E-03	5.9E-06
	Process Occurrences (Coil Failures)	7.1E-04		6.3E-08
	Total (Airborne)	2.7E-02	2.5E-03	6.1E-06
	Total (Liquid)	7.1E-04		6.3E-08
	Total	2.8E-02	2.5E-03	6.1E-06
High Heat Waste	Natural Phenomena	2.4E-06	6.6E-07	1.4E-09
	Externally Induced Failures	7.9E-04	7.3E-05	1.7E-07
	Process Occurrences (Airborne)	1.6E-02	1.4E-03	3.5E-06
	Process Occurrences (Coil Failures)	2.5E-03		2.2E-07
	Total (Airborne)	1.6E-02	1.5E-03	3.6E-06
	Total (Liquid)	2.5E-03		2.2E-07
	Total	1.9E-02	1.5E-03	3.9E-06
Total for all Unit Operations	Airborne	1.5E-01	2.0E-02	3.7E-05
	Liquid	2.1E-02		1.8E-06
	Total	1.7E-01	2.0E-02	3.9E-05

TABLE 5-36. Summary of Risk for Accidents

Pathway	Risks Due to Accidents			
	Natural Phenomena	Externally Induced Failures	Process Operations	Total
Airborne Release to Offsite Population, person-rem/yr	2.4×10^{-5}	7.6×10^{-3}	1.4×10^{-1}	1.5×10^{-1}
Airborne Release to Onsite Population, person-rem/yr	7.4×10^{-6}	8.0×10^{-4}	1.9×10^{-2}	2.0×10^{-2}
Airborne Release to Offsite Maximum Individual, rem/yr	1.0×10^{-8}	1.3×10^{-6}	3.6×10^{-5}	3.7×10^{-5}
Liquid Release to Offsite Population, person-rem/yr*	--	--	2.1×10^{-2}	2.1×10^{-2}
Liquid Release to Offsite Maximum Individual, rem/yr*	--	--	1.8×10^{-6}	1.8×10^{-6}

*Direct release to Four Mile Creek.

canyon crane cab and the maintenance bridge are 4.5 rem if the occurrence is directly below the crane or bridge and 0.02 rem at 60° from the vertical.

A significant consequence of pluggage in sample lines is radiation exposure to personnel during removal of the pluggage. Radiation resulting from pluggage of hot canyon samplers has been estimated to be:

Mean open door exposure rate	3590/235 mrad/mr/hr
Median open door exposure rate	2000/15 mrad/mr/hr
90% range	20,500/1265 to 16/4 mrad/mr/hr

The first of the listed dose values is the open window or beta-gamma dose while the second is the closed window or gamma dose. Other exposures in the sample aisles include radiation fields at the open door to samplers in excess of 1 R/hr which occur at a rate of about once every two weeks. These usually are the result of sampler pluggage, leaks, or broken sample vials.

No significant releases of iodine have been reported as a result of failures in the iodine reactors. However, during changeout of the Berl saddles from the cartridges, a radiation field of 4000 mrad/400 mr/hr at 1 ft is typical.

Air reversals represent a loss of ventilation system function and its capability to control the spread of airborne radioactive material. While previous incidents have left contaminated particulates in a zone, air reversals can result in airborne activity migration, potentially to occupied areas. The most undesirable consequence is an uptake by a worker. Uptakes were experienced in 5% of the air-reversal cases. Skin contaminations or contamination of a work area are less severe, usually resulting only in lost work time during cleanup. Many air reversals result in no detectable airborne activity or contamination spread. Air reversal data were given in Table 5-30.

Significant cases of surface contamination to 1×10^6 dis/min alpha and 80,000 counts/min beta-gamma have been measured in the gang valve corridors. The usual sources include seepage through ceiling cracks, pipe leaks, nozzle leaks, maintenance operations, and leakage during embedded pipe liner installations.

Significant radiation exposure can occur as a result of spills, suckback, seepage through conduit in the abandoned "fireye" system, or from coil leaks into the condensate header. The calculated mean gamma radiation intensity as a result of suckback is 175 R/hr at 3 in. Seepage through the "fireye" conduit produces the same order of magnitude field; however, coil failure radiation intensities are about the three orders of magnitude less.

Gamma radiation intensities in personnel areas due to escape of liquids through expansion joints generally are about 10% of that caused by suckbacks in the hot gang valve corridor and second level. However, liquid emanating from the joints is uncontained except in specific locations known to be especially vulnerable, such as in the Section 9 personnel tunnel. Intensities range from a few thousand counts per minute to 500 R/hr at 3 in. Alpha contamination to 6×10^5 counts/min has been reported.

Activity migration through failed embedded instrument conduit running beneath the cell floors is causing radiation intensities in the first level and in the gang valve corridor in the same range as for expansion joints. Lead shielding has been required in some cases.

Personnel may be exposed to high radiation fields during equipment handling in the railroad tunnel, either as a result of some action on the part of the recipient, or as a result of some situation beyond his control. Generally, about 5% of the cases of radiation exposure were the result of carelessness on the part of the recipient. Contrarily, situations in which the recipient could not have exercised reasonable control included high radiation from equipment being handled, shielding door being opened with personnel in the airlock, and radiation from fuel that was dropped earlier and not immediately recovered from atop cell covers. Radiation fields of up to 25 R/hr at 10 ft have been experienced in the railroad tunnel.

5.6 ACCIDENT MITIGATION

This section summarizes the design features and the physical and administrative controls which can be implemented to reduce accident risks that have been identified. Section 6 further describes the safety related systems and administrative controls that are available to reduce the occurrence of safety-related incidents.

5.6.1 Natural Phenomena

5.6.1.1 Winds and Tornadoes

Building 221-F is designed as a "blast resistant structure" to withstand 1000 lb/sq ft pressure. The design basis tornado corresponds to an Intensity 5 on the Fujita scale (261-318 mph) with a probability of striking the SRP site of 8×10^{-8} /yr. Even through a tornado of this intensity might cause damage to outside ductwork (e.g. the air supply), the truckwell or the tunnel airlock, the radiological hazard would not be affected beyond a possible air reversal of a few seconds duration in the event the eye of the tornado moved directly over the building. The low frequency of occurrence makes further consideration unproductive. Although winds up to 175 mph are more frequent (1×10^{-5} /yr) the expected velocity is less than in the design basis tornado and further consideration of the consequences was not made.

5.6.1.2 Earthquake

The canyon building is designed to resist an MM VIII earthquake that has an expected frequency of 2×10^{-4} /yr.

5.6.1.3 Meteorite

The estimated frequency of a meteorite impact is so low (1×10^{-15} /hr) that this event is considered hypothetical.

5.6.1.4 Floods

The F-Canyon Facility's elevation of 100 ft above the floor of any nearby valley precludes a flood engulfing the area.

5.6.1.5 Other Natural Phenomena

Extremes in temperatures, rain, snow, or lightning are not expected to result in increased releases of radioactive material from F-Canyon, even though some operations may be halted. The effect of heavy rain and of lightning that frequently accompanies it aggravates operations (false alarms, loss of normal power) but does not result in an abnormal release of radioactive materials from the systems.

The occurrence of frozen lines and degraded performance of instruments and valve-operation during periods of low temperature snow and ice retards the progress of operations but has no effect on radiological releases from the facility.

5.6.2 Externally Induced Failures

The consequences of fires and/or explosions in adjacent facilities are included in the events considered under impact. The releases assumed to occur from the stack are low. Emergency evacuation procedures would limit the exposure of operating personnel. The remote location of the facilities from the public at large limits the number of individuals involved in any evacuation. In addition, site emergency plans include procedures for the evacuation of potentially exposed populations located in the down wind sector.

The frequency of an aircraft impacting the F-Canyon building is estimated at 1.4×10^{-11} /hr. When considered with the low damage associated with the small planes normally flown in the area, the consequences of the event fall to a very low level. Only the consequences of a large aircraft with a full fuel load is considered. The expected frequency of occurrence is so low, however, that this event was not analyzed separately.

5.6.3 High Energetic Events

Only an explosion of severe magnitude could potentially cause damage sufficient to qualify as a high energy event. None were found in this analysis.

5.6.4 Medium Energetic Events

5.6.4.1 Fire

The likelihood of fires occurring in the F-Canyon is limited by rigid procedural control of ignition sources present in the systems and control of the inventory of combustibles. Similarly, resin fires are minimized by the

design of the columns that ensures that the resin beds remain submerged at all times. Fire detection and control equipment in the canyon reduce consequences of any fire that might occur. Similarly, the canyon ventilation air is filtered to remove airborne particulates before being released to the environment from the stack.

5.6.4.2 Uncontrolled Reaction

The likelihood of an uncontrolled reaction occurring in a canyon system is limited through rigid administrative controls. These ensure continuous surveillance and controls the rate of reagent additions to a reactive system where an accumulation of excess reagent may occur.

The confinement system of the F-Canyon is designed to limit the release of any material that might be forced from process vessels to the canyon floor by a runaway reaction and remove any airborne particulate matter from the canyon ventilation air that occurred before discharging it to the environment via the 200-ft stack.

Regularly programmed reviews of the individual unit processes and operational procedures by the Separations Technology Group for performance and safety ensures that potential sources of uncontrolled reactions are identified. Appropriate measures are developed and included in the operational procedures.

5.6.4.3 Criticality

The likelihood of a criticality event occurring in the F-Canyon processing system (excluding the FB-Line) is reduced through a system of rigid administrative and procedural controls which act to limit and monitor the quantities and the concentrations of fissile materials in process solutions to safe levels. Similarly, procedures developed for nuclear safety control ensure that process conditions are monitored and maintained to prevent precipitation of fissile materials in process equipment. In addition, the location of neutron monitoring instruments at strategic points in the 1B and the 2A banks of the solvent extraction system detect abnormal accumulations of fissile material.

Frequent sampling and continuing monitoring of process conditions, reagent solution composition, and fissile material balances act to minimize the probability of a criticality event occurring.

In the unlikely event that a nuclear incident should occur in the process equipment, the massive concrete walls of the canyon building serve as a radiation shield to protect operating personnel. Only the crane operator in the warm canyon and some specific locations as the maintenance bridge and the gang valve corridor have no benefit from this shield.

The canyon areas are limited access areas. Large numbers of personnel are not permitted in the zones and these areas are completely vacated during times of suspected process upsets. The following additional precautions are taken:

1. The warm canyon process will normally be shut down whenever maintenance requires that some cell covers be removed. It is improbable that a criticality incident would occur with the process in a shutdown condition.
2. A second layer of cell covers is placed over the mixer-settler section.

5.6.5 Low Energetic Events

The dominant mitigating circumstance in the canyon facilities in relation to low energetic events is the design and construction of the canyon building. The massive walls and roof of the system will alternate beta-gamma radiation of any material released to the canyon. Released liquids will collect and remain in the sumps; dispersed materials that survive deposition in the cell (and ductwork) will be effectively removed from the spent canyon ventilation air by the system filters.

An effective system of instrumentation, designed to detect abnormal conditions and alert operating personnel, provides for quick response to halt operations and take corrective action to halt or prevent the release of hazardous material to the environment.

In addition to the design of the confinement and instrumentation systems, two others also merit consideration. The first is the design of the segregated cooling water system which operates in conjunction with the cooling water diversion system. Each provides for monitoring the water for contamination before discharging it to the seepage basins. The other is the installation of cash regulators on the heating/cooling coils of process vessels which act to minimize the consequence of leaks in those items.

5.6.6 Residual Activity Release Event

5.6.6.1 Suckback

The primary mitigation measure is the automatic air blow timer on the gang valves. Radiation monitors are also set to detect abnormally high levels and sound an alarm (locally and in the control room) that alerts operating personnel to the hazardous condition. Training requires evacuation from the affected area via previously identified routes.

5.6.6.2 Air Reversal

Since the usual course of an air reversal is an unexpected compromise of some confinement barrier, the training and experience of operating personnel are important to minimizing consequences. Administrative controls require that operating personnel be qualified to perform tasks assigned to them, and detailed reviews of planned abnormal operation act to minimize the occurrence of these events. Pressure drop instrumentation is also available to indicate air reversals.

5.6.7 Chemical Hazards

Training qualifies the operating personnel to handle hazardous cold chemicals used in preparing process reagent streams. Protective equipment is provided. High ventilating system air flow rates also prevents the accumulation of hazardous chemical fumes in occupied areas.

5.6.8 Support Systems

5.6.8.1 Shops and Decontamination Facilities

The training and qualification of personnel acts to minimize the frequency of unanticipated and unprepared for occurrences. Continuous surveillance by Health Protection Department personnel, who also review all procedures, acts to minimize the hazards to personnel by providing guidance to their preparation and providing personnel safety equipment. Radiation monitors are also provided.

5.6.8.2 Cell Covers

Administrative controls require continuous reviews of potentially hazardous conditions and procedures by qualified Separations and Health Protection personnel. Consequences of an event involving dropping of cell covers in the canyon are limited by the canyon confinement systems.

5.6.8.3 Gang Valve Corridors

Consequences of events in this area are minimized by the radiation detectors, located strategically, to signal abnormal radiation levels. Fire detectors and protection systems are designed to detect and limit the spread of fire if it occurs. Cleanup and maintenance procedures have been, and continue to be, developed to maximize the safety of personnel. Design of the confinement system reduces the likelihood of exposure of the remaining operating personnel and the public at large.

5.6.8.4 Cranes

Crane operators are qualified by training, and demonstrated performance, so that misoperations are limited. Exposure of facility personnel is limited to personnel of the operator/maintenance group who function under rigid procedural controls, augmented by safety requirements and procedures developed by the Health Protection Department. Radiation monitors are also provided in the crane cabs. Canyon confinement systems limit exposure to the public at large.

5.6.8.5 Sample Aisles

Administrative controls require trained and qualified operator personnel, as well as Health Protection surveillance. Radiation alarms alert operator personnel to abnormal radiation levels. The confinement system is designed to limit release of contaminants to the environment.

5.6.8.6 Steam Distribution

Disabling the steam distribution system results in a shutdown of the system because of the lack of process heat and, in many cases, the capability to transfer process solutions. No hazard to operating personnel or to the public is anticipated as a result.

5.6.8.7 Compressed Air

Loss of instrument and process air results in automatic shutdown because the system instrumentation and controls are designed to progress to a safe mode. Three of the four compressed air systems must fail simultaneously to achieve total air supply failure because the systems are cross connected by piping. The compressors of at least one of these systems is supplied with emergency power. Loss of breathing air interferes with maintenance or other work that requires the worker to wear a mask; however, an emergency supply of breathing air in cylinders provides sufficient time for such workers to halt the work in progress and leave the contaminated area.

5.6.8.8 Electrical Distribution

Power is received from an area loop. Critical equipment is automatically connected to an emergency power supply sufficient to achieve an orderly shutdown of the process. The emergency diesel generator and the automatic switching gear are maintained in an immediately available status, ensured by routinely scheduled preventive maintenance and an inspection/test program.

5.6.8.9 Cooling Water Distribution

Cooling water is assured by two supply sources powered by redundant electrically-driven pumps, backed by one steam-turbine-driven pump. Each of the supply systems is independently piped and provided with control instrumentation that automatically starts the independent backup system if the pressure in the main system drops to a preset value.

5.6.9 Engineered Safety Features

5.6.9.1 Ventilation System

The design basis of the ventilation system includes housing in blast resistant structures that resist failures due to high winds and earthquake. The filters are designed to reduce the airborne particulate content of the ventilation air

by factors of 10^3 to 10^4 before discharge. The exhausters are redundant and normal power is supplemented by emergency power, automatically switched on in the event of a failure in the normal supply. There are also dedicated diesels for two of the four main exhausters.

Administrative controls are in place that act to bring extra exhausters on line, adjust dampers to regulate and control air flow direction in emergency or abnormal conditions, and prevent air reversals. Similarly, equipment failures (fans, dampers, etc.) result in operator response (to alarms) to bring standby equipment on line.

5.6.9.2 Water Return and Diversion

Alarms stemming from radiation monitors in the water return line signal a failure of a system upstream in the canyon equipment. The segregated cooling water system is designed to contain leakage from leaking coils and signal operating personnel of the existence of abnormal conditions. Failure of the segregated cooling water system results in contamination of the system but prevents its discharge to the seepage basins because operational procedures, in this case, call for a hold up until the nature of the contamination can be examined and its magnitude established. Normally, if verified, the contaminated cooling water is returned to an appropriate waste evaporator. The monitors are also supplied with emergency power.

5.6.9.3 Seepage Basins

The seepage basins are designed as the last line of defense against release of hazardous materials to offsite areas by seepage into the soil for a sufficient length of time to allow their decay to tolerable levels.

5.6.9.4 Secondary Confinement

The secondary confinement system (canyon and the air filtering systems) is designed to restrict and control the flow of hazardous materials from the canyon process systems to the environment. The ventilation system and maintenance of a strict surveillance program to detect and repair flaws or penetrations in the confinement structures ensures early detection of releases and the initiation of corrective measures, thereby limiting the consequences to tolerable levels.

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6.0 SAFETY-RELATED ITEMS

The F-Canyon complex is equipped with a number of physical and administrative features to prevent, detect, or mitigate accidents. Many of these features are described in Sections 3, 4, and 5. The safety-related systems, structures, and items are listed in this section.

6.1 LIST OF SAFETY-RELATED SYSTEMS, STRUCTURES, AND ITEMS

Safety-related systems, structures, and items are listed in Table 6-1. Safety instrumentation, monitors, and controls required for operation within safety limits and the surveillance requirements for these instruments and controls are listed in Table 6-2. Operational Safety Requirements (OSRs) for these systems are described in Reference 1. Safety-related items required for each system in which a QA criterion involving safety can be exceeded are discussed in the following sections. Detailed procedures which limit the potential for exceeding a QA criterion are referenced in Section 7.

The formal administrative control system discussed in Section 4.3 ensures that all process operations and many allied activities, such as maintenance operations, are performed according to written procedures. The intent of these procedures is to ensure systematic control of safety, quality, and yield by adherence to approved Technical Standards that define the process limits within which the facilities are operated. These standards ensure the safety of personnel, permanently installed equipment, and the environment. Thus, the procedures and standards that are required for the safe operation of the facilities are included as safety-related items. These procedures are also discussed in the following sections.

6.1.1 Target Receipt and Storage

Irradiated materials are received at Building 221-F in transfer casks mounted on special railroad cars. The cars are pushed from the railroad tunnel air lock into the railroad tunnel by a battery-powered locomotive. The targets, in the form of slugs in buckets, are removed from the cask car using the hot canyon crane and are either charged directly to the dissolver or stored in bucket storage.

An unplanned nuclear criticality could occur in Target Receipt and Storage only if the targets received are enriched in the ^{235}U isotope. Mark 31A/B targets currently being received are depleted in the ^{235}U isotope. Plutonium scrap, Rocky Flats Scrub Alloy (RFSA) and MK-42 irradiated material are specialty materials received at F-Canyon that are processed in limited quantities because of critical mass restrictions. These materials are packed into numbered bundles that are cross checked against an approved list of material designated for processing.

TABLE 6-1. Safety-Related Systems, Structures, and Items

Safety Features	Systems
Engineered Safety Features	Ventilation System Cooling Water Return and Seepage Basins
Structures	Secondary Confinement for Liquids
Safety-related items	Cash Pressure Regulators
Support Systems	Emergency Electric Power
Administrative Controls	Target Receipt and Storage Dissolving Head End Solvent Extraction and Recovery Ion Exchange High Activity Waste Rerun Chemical Handling and Storage Cooling Water

TABLE 6-2. Safety Instrumentation, Monitors, and Controls

Item	Surveillance and/or Test Interval
Sand Filter	
Atmospheric release	1 week
Efficiency	18 months
Emergency Power	
Electrical interlock	12 months
Diesel auxiliary	4 months
NIM	18 months
Segregated and Recirculating Water System	3 months
Neutron Monitors	3 months
Evaporator and Dissolver Temperature Instruments	18 months
Solvent Storage and Solvent Extraction Feed Tank Temperature Instruments	18 months
Accountability Tank Liquid Level and Specific Gravity	3 months
Accountability Tank Temperature Instruments	18 months
Hydrogen Dilution Controls	6 months
Ion Exchange Column Temperature Instruments	18 months
AgNO₃ Reactor Temperature Instruments	18 months
Solvent Extraction Conductivity Meters	6 months

6.1.2 Dissolving

Purex dissolver feed in Building 221-F consists of depleted uranium targets (Mark 31A/B) which contain varying amounts of plutonium and a maximum of about 0.30% ^{235}U . The aluminum cladding is removed with 50% NaOH in the presence of an NaNO_3 solution. The uranium is dissolved in nitric acid and the resulting solution is processed through solvent extraction to recover plutonium and uranium.

The allowable mass of ^{238}U targets in water is infinite at a ^{235}U enrichment of 0.69%. Mark 31A/B targets are enriched to less than 0.20% ^{235}U . Any resulting uranyl nitrate solutions would be critically safe. Irradiated fuel is more critically safe than unirradiated fuel. The dissolver uranyl nitrate solution is actually subcritical to an equivalent enrichment of 1.1% ^{235}U .

An unplanned nuclear criticality can be caused in this system with excessive ^{235}U enrichment.

The RFSA and the MK-42 irradiated materials bundles are designed for charging into inserts that fit into the four dissolver charging ports. The geometry of the insert specifically matches the bundle and the size of the well or receptacle in the insert and limits the quantity of material that can be charged. The dissolution of these specialty materials is with Hg catalyzed nitric acid and hydrofluoric acid.

The Separations Department will confirm that the bundles and MK 31A/B targets received are acceptable for processing to ensure that the dissolvers are prepared to receive the designated material.

The Separations Department will confirm that targets received are type MK 31A/B to ensure that dissolver solutions are subcritical.

6.1.3 Head End

Purex dissolver solution in Building 221-F is treated with a gelatin solution to remove silicon. Silicon is removed in head end to prevent formation of solids that can cause emulsions in the solvent extraction equipment. In special campaigns where zirconium and niobium are high, a strike with gelatin, to remove silica, and a strike with MnO_2 , to remove fission product, are used.

An unplanned nuclear criticality can occur in this system through exceeding the critical plutonium concentration or by selective precipitation of ^{239}Pu in the centrifuge or a vessel. Administrative controls and process control instrumentation, as described below, are used to exclude selective precipitation of plutonium and to maintain concentration of fissionable ^{239}Pu below the critical concentration:

1. Procedures for operating the centrifuge comply with all limits specified in 221 Building Technical Standards, or any Test Authorization. The maximum limit for ^{239}Pu concentration in a vessel or centrifuge is 6.75 g/l (1).

2. Prior to the start of each Purex campaign, installation of required nuclear safety blanks (NSB) must be confirmed.
3. Operate head end as specified by approved operating procedures to maintain the ^{239}Pu concentration within the Technical Standard limits.
4. Abnormal plutonium concentration and/or selective precipitation of plutonium in the centrifuge or head end vessel are detected and prevented by:
 - (a) Dissolver solution acidity and P-Factor, are indicators of the presence of precipitated/polymerized plutonium, and must be determined for each dissolver run before a gelatin strike is performed on the solution.
 - (b) Cakes from more than six head end runs will not be accumulated in a head-end vessel before neutralizing unless authorized by Separations Department and Separations Technology Department Area Supervisors.

6.1.4 First Solvent Extraction Cycle

First solvent extraction cycle in Building 221-F has two functions: 1) to provide decontamination of the Pu and U from fission products and chemical impurities, and 2) to separate Pu and U into individual streams for further processing. First cycle consists of centrifugal contactor 1A, jumbo mixer-settlers 1B and 1C, decanters, and process tanks.

A nuclear criticality caused by exceeding the critical concentration of ^{239}Pu in a vessel or mixer-settler can occur in this cycle. Concentration and inventories of fissile ^{239}Pu below the critical limits are maintained through administrative control and process control instrumentation as described below:

1. Procedures for operating the first solvent extraction cycle comply with all limits specified in 221 Building Technical Standards or any applicable test authorization. The limits for maximum ^{239}Pu are 6.75 g/l (1).
2. Prior to start of each Purex or special campaign, installation of required NSBs must be confirmed.
3. Operate first cycle as specified by approved operating procedures to maintain the ^{239}Pu concentration within Technical Standard limits.
4. Prevent abnormally high plutonium inventory in mixer-settler 1B by maintaining the 1BX reductant concentration above the minimum limit specified in the Technical Standard. The Technical Standard further requires operation of at least one of the dual neutron monitors on mixer-settler 1B to assure

early detection of a build-up of fissile inventory in 1B while in operation (1).

The neutron monitors on 1B mixer-settler must be calibrated and tested as specified by procedure.

6.1.5 Second Pu Solvent Extraction Cycle

In the second plutonium cycle, the plutonium bearing stream (1BP) from the first cycle solvent extraction is treated to adjust the plutonium valence and adjust the nitric acid concentration. After adjustments the solution is fed to mixer-settler 2A where the plutonium is extracted into the organic solvent phase (30% tributyl phosphate in n-Paraffin). The solvent stream 2AP from mixer-settler 2A is fed to the end stage of mixer-settler 2B where plutonium valence is reduced to the nonextractable Pu III state and the plutonium is stripped into a dilute acid stream (2BP). This plutonium stream 2BP is routed to FB-line for further processing. A nuclear criticality can occur by exceeding the critical concentration of plutonium in a vessel or mixer-settler in this cycle.

Administrative controls which maintain the concentration of fissionable plutonium less than the critical concentration are outlined below:

1. Prior to the start of each campaign, the installation of NSBs must be confirmed. During operation, a NSB can only be removed per a special procedure requiring approval by the superintendents of the Separations and the Separations Technology Departments.
2. Prior to receiving transfers from FB-Line or during extended (6 months) downtime periods, concentration of fissile material in the vessels will be controlled.
3. Abnormally high plutonium concentrations in the mixer-settler are detected by dual neutron monitors on each mixer-settler. At least one shall be functioning while the cycle is operating to permit monitoring and corrective actions to be taken.
4. 2AS acidity must be monitored during cycle operation by the 2AS conductivity meter, or head tanks must be sampled for percent acid each time acid is received for use in the process.
5. The equipment that is required to be functional in (3) and (4) must be calibrated and tested per Table 6-2.

6.1.6 Second Uranium Solvent Extraction Cycle

The aqueous uranium bearing stream 1CU exiting first cycle is concentrated in a continuous evaporator before it is fed to the second uranium solvent extraction cycle. The concentrate is chemically adjusted before the uranium is further decontaminated by extracting it into solvent in mixer-settler 1D and stripping it from the solvent in mixer-settler 1E. The aqueous stream

leaving mixer-settler 1E is transferred to Building 221-1F, where uranium is processed to uranium trioxide.

A nuclear criticality can occur in this cycle if the critical concentration of ^{239}Pu is exceeded in a vessel or mixer-settler. For this to occur, plutonium must exit the first cycle in the 1CU stream. This abnormal condition requires that the 1B mixer-settler solution be insufficient in reducing normality to partition Pu from U. Sufficient reductant must be supplied to the 1BX Stream to divert plutonium to the 1BP stream and, thus prevent it from exiting in the 1CU stream with the uranium.

Concentration of ^{239}Pu is maintained below the critical concentration through administrative control and process control instrumentation as described below:

1. Procedures for operating the first solvent extraction cycle and second uranium solvent extraction cycle comply with all limits specified in 221 Building Technical Standards or any test authorization. The limits for ^{239}Pu concentration in a vessel is 6.75 g/l (1).
2. Prior to start of each Purex or special campaign, installation of required NSBs must be confirmed.
3. Operate first cycle as specified by approved operating procedures to maintain the ^{239}Pu concentration within Technical Standard limits.
4. Nuclear Safety Training will be given to new operators and reviewed every six months thereafter.
5. Abnormally high plutonium inventory in 1B mixer-settler is detected and prevented by:
 - (a) Operating properly one of the dual neutron monitors on 1B mixer-settler while feeding fissile material (1).
 - (b) Monitoring of the 1BP reducing normality and plutonium concentration while feeding fissile material.
 - (c) Calibrating and testing the neutron monitors on 1B mixer-settler quarterly as specified in (1).
6. Operate the second uranium cycle as specified by approved operating procedures to maintain the ^{239}Pu concentration within Technical Standard limits (1).

6.1.7 Solvent Recovery

Each of the three solvent extraction cycles (first cycle, second plutonium cycle, second uranium cycle) has a separate continuous solvent recovery system. Each solvent recovery process removes degradation products and

radioactive contamination from spent solvent by alternating alkaline and acidic washes. The washed solvent is then returned to the respective extraction process.

A nuclear criticality can occur if the critical mass of ^{239}Pu in second Pu cycle alkaline solvent washer is exceeded. For this to occur, an insufficient reducing normality in mixer-settler 2B of the second plutonium cycle must exist. Plutonium would not be stripped from the solvent and exiting the cycle in the 2BW stream.

Plutonium inventory in second Pu cycle solvent washer is maintained below the critical mass through administrative control and process control instrumentation as outlined below:

1. Installation of NSBs must be confirmed prior to the start of each campaign.
2. Concentrations of fissile material in vessels will be controlled as required by procedures during extended (6 months) downtime periods and prior to receiving transfers from FB-Line.
3. Abnormally high plutonium concentrations in the mixer-settler are detected by dual neutron monitors on each mixer-settler. At least one shall be functioning while the second plutonium cycle is operating to permit monitoring and corrective actions to be taken (1).
4. 2AS acidity must be monitored during cycle operation by the 2AS conductivity meter, or head tanks must be sampled for percent acid each time acid is received for use in the process.
5. The equipment that is required to be functional in (3) and (4) must be calibrated and tested as specified by procedures (1).
6. The second Pu cycle solvent recovery system must be operated as specified in procedures to maintain the ^{239}Pu mass within Technical Standard limits (1).

A solvent fire causing a crane operator fatality or disability is not credible because of the combination events required. These are as follows:

- The crane is directly above the cell where the fire occurs.
- The cell cover is off (process not operated with cell covers off).
- Hot gases enter the crane cab and are inhaled by the operator. Fresh air supply to the crane cab would have to fail during incident.

6.1.8 Ion Exchange

Frame II-F is a modular steel structure containing three ion exchange columns and five tanks. In the Frame II-F process neptunium and plutonium recovered on the primary recovery column (PRC) from high activity waste (HAW) are separated and the neptunium decontaminated. The first two ion exchange columns use anion resin; the last column uses cation resin.

PRC eluate is fed to the first anion exchange column where the plutonium and neptunium are absorbed on the resin. The plutonium is partitioned from the neptunium and recycled to HAW. The neptunium is eluted from the column, and valence adjusted. The neptunium is fed to the second anion exchange column for absorption and further decontamination. The neptunium eluted from the second column is fed to the cation column for removal of thorium isotopes. The neptunium product is shipped to HB-Line as a solution.

An unplanned nuclear criticality can occur if the critical concentration of ^{239}Pu in one of the frames anion exchange columns is exceeded. Concentration of fissionable ^{239}Pu is maintained below the critical concentration through administrative control as described below:

- Operate the frame process as specified by procedure to maintain the ^{239}Pu concentration within Technical Standard limits or any Test Authorization. The maximum limit for ^{239}Pu concentration in a vessel is 6.75 g/l (1).

6.1.9 High Activity Waste

High activity waste streams are processed to 1) recover neptunium and plutonium, 2) recover nitric acid, and 3) reduce waste volume. The waste streams are fed to two continuous evaporators that are operated in series. Concentrate from the first stage is sampled, adjusted, and fed to the PRC. Neptunium and plutonium are absorbed on the anion resin, eluted, and transferred to Frame II-F for purification of the neptunium and recycle of the plutonium to the process. The PRC raffinate is transferred to a batch evaporator; and free acid is removed by steam stripping, reducing waste volume prior to disposal. Overheads from the first stage go to the second stage evaporator. Overheads from the second stage evaporator are transferred to the acid recovery unit for recovery of nitric acid.

A nuclear criticality can occur only by exceeding the critical ^{239}Pu concentration in a vessel (resin column, evaporator, tank). Concentration of fissionable ^{239}Pu is maintained below the critical concentration using administrative controls as outlined below:

- Operate the HAW-PRC process as specified by procedure to maintain ^{239}Pu concentration within Technical Standard limits or any Test Authorization. Maximum limit for ^{239}Pu concentration in a vessel is 6.75 g/l (1).

Vigorous exothermic reactions can occur between nitric acid or heavy metal nitrates and some TBP-diluent mixtures under conditions that might arise during concentration of waste and product solutions. The evaporator operation

is controlled to prevent the introduction of solvent and to prevent exceeding the safe temperature limit of 130°C.

- The amount of entrained solvent is limited to 0.5 volume percent.
- The evaporator temperature is limited by restricting the steam coil pressure to 25 psig.

6.1.10 Rerun

The principal functions of the rerun station in Building 221-F are to treat and convert off-standard process streams and sump materials that contain recoverable amounts of actinide elements to solutions that are suitable for reprocessing or disposal (both aqueous waste streams and spent solvent).

A nuclear criticality can occur if the critical concentration or inventory of fissile material in a vessel is exceeded. Fissile material processed through rerun is maintained at a less than critical concentration or inventory by restricting, through administrative control, total Pu in any vessel to a value less than the Technical Standard limit (1). Administrative controls are used to prevent polymerization or precipitation of Pu in process solutions by additions or alkaline solutions. Tank contents are sampled and analyzed prior to making any additions to process solutions.

6.1.11 Ventilation System

The 221-F ventilation system is designed to supply fresh air to occupied sections of the building and prevent the spread of airborne contamination within the facility or to the environment. The canyon building ventilation is supplied by four systems: canyon air, center section air, gang valve corridor air, and FB-Line air. It is exhausted by five systems: canyon air, center section, process vessel vent, recycle vessel vent, and FB-Line exhaust. The 221-F ventilation system includes the supply fans, exhaust fans, HEPA filters, process vessel vent filters, sand filters (294-F and 294-1F), stack (291-F), and associated duct work.

An explosion of ammonium nitrate in the warm canyon process vessel vent (PVV) filters could result in the fatality or permanent disability of the warm canyon crane operator if he were working directly above, or near, section 5 at the time of the explosion. This event was discussed in Section 5.1.9.1.

To reduce the risk of an explosion in the PVV filters, the PVV filter in the warm canyon was flushed in July 1982. Need for further flushing of the PVV filters will be reviewed every year until installation of ammonia scrubbers.

6.1.12 Chemical Handling and Storage

The primary operations in chemical handling and storage are the receipt, storage, and dispersal of chemical supplies for the area processes. The chemical handling and storage facilities in F-Area are made up of the Building

221-F third level feed tank gallery, Building 211-F outside storage tanks, and Building 222-F cold feed preparation.

There are two types of incidents which could cause death or permanent disability to personnel: 1) explosion or fire from a chemical spill and 2) exposure to toxic chemical. These hazards are discussed in Section 5.1.7.

In the event of a chemical leak or overflow where personnel exposure to hazardous chemicals is required, exposure is limited by administrative control and protective equipment as follows:

- Safety Manual Items 85 and 86 define the Minimum Protective Equipment for Handling Hazardous Materials and Toxic Materials (2).
- Employee exposure to any of the hazardous materials listed in OSHA Regulations, Title 29 Code of Federal Regulations, Part 1000 are limited as specified. Controls of exposure are based on the most restrictive threshold limit value (TLV) (2). Employee exposure is monitored and maintained below the TLV by the Health Protection Department.
- New employees receive occupational health training as part of the new employee orientation. Retraining is conducted at least annually as discussed in Section 4.

6.1.13 Waste Removal

Waste removal from Building 221-F includes disposal of 1) solid waste from each canyon, 2) solid waste from regulated areas, 3) solvent, and 4) liquid waste via the waste headers. In general, solid waste is packaged and shipped via rails from the hot canyon and transported by truck from remaining locations within the facility. All solid waste is destined for the Burial Ground (Building 643-G). Liquid waste is neutralized and transferred from canyon vessels to the Waste Farm (Building 241-F) via waste headers.

A nuclear criticality can occur if the critical ^{239}Pu concentration in waste handling vessels is exceeded. Concentrations of fissionable ^{239}Pu are maintained below the critical concentration using administrative controls as described below:

- Procedures for operating the dissolver to remove cladding must comply with all limits specified in 221 Building Technical Standards or any Test Authorization. The maximum limit for ^{239}Pu concentration in a vessel is 6.75 g/l (1).
- Prior to start of each Purex campaign installation of required NSBs must be confirmed.
- Procedures for operating the HAW-PRC process comply with all limits specified in 221 Building Technical Standards or any Test Authorization. Maximum limit for ^{239}Pu concentration in a vessel is 6.75 g/l (1).

- Procedures for operating the centrifuge comply with all limits specified in 221 Building Technical Standards or any Test Authorization. The maximum limit for ^{239}Pu concentration in a vessel or centrifuge is 6.75 g/l (1).
- Cakes from more than six head end runs will not be accumulated in a vessel before neutralizing unless authorized by Separations Department and Separations Technology Department Area Supervisors.
- Operate the process as specified by procedures.
- Fissile material processed through rerun is maintained at a less than critical concentration or inventory by restricting, through administrative control, total Pu in any vessel to a value less than the Technical Standard limit (1). Administrative controls require representative tank samples be taken to prevent polymerization or precipitation of Pu from unwanted alkaline additions.

If a piece of highly contaminated canyon equipment were removed from either canyon without sufficient decontamination, personnel handling the equipment could be exposed to very high radiation levels. Radiation monitors in the tunnel would warn personnel of high radiation levels. Administrative controls that will prevent fatalities or total disabilities include:

- Operating staff must comply with procedures for removal of contaminated canyon equipment.
- A special procedure is required to remove contaminated canyon equipment not covered by existing procedures.

6.1.14 Cooling Water Systems

In F- and H-Canyons and outside facilities, the coils of vessels containing process solutions are kept under pressure to prevent radioactive solution from entering the cooling water system through a failed coil and contaminating the environment. In vessels served by steam and cooling water to common coils, that is accomplished with automatic air pressurization systems utilizing Cash air pressure regulators on the coil inlets, and back-pressure valves in the coil outlets. In vessels having only cooling water to their coils there is a continuous supply of water which is never completely valved off. This, together with the elevated inlet and outlet nozzles of the cooling water jumpers, maintains a hydrostatic pressure sufficient to overcome any opposing static head of process solution in the vessels.

To provide a continuous flow of water for coil pressurization and assure a supply of water for cooling condensers and certain key vessels in an emergency, two cooling water supply systems are used. They are designated as the "normal" and "independent" cooling water supply systems. The important difference between the two systems is that the independent cooling water header remains pressurized by a steam turbine pump in the event of a power failure, while the normal cooling water header does not.

The normal cooling water header supplies cooling water to the coils of all canyon vessels except those which might require continuous cooling if the normal cooling water pressure should fail. The pot coils in the dissolvers and all dissolver and evaporator condensers are supplied from the independent header. Each vessel coil which does not have an air pressurization system is connected to the independent cooling water supply by a 3/4-in line with a manual block valve at the cooling water inlet nozzle on the second level of Building 221. The block valves remain cracked open, providing a small amount of cooling water to maintain positive coil pressure.

Cooling water leaves the building in three streams: clean cooling water, circulated cooling water, and segregated cooling water.

- The clean cooling water returns only from nonregulated parts of the building. It comprises clean normal cooling water from air conditioning units and air compressors and steam condensate from the center section of the building. It returns underground directly to the 282-2F pump basin from where it is returned to the Building 285-F reservoir without monitoring. Although the clean water return cannot be contaminated by the equipment that it cools it can contain radioactivity originally present in the cooling water supply.
- The circulated cooling water return stream accumulates from Building 211/A-Line and canyon vessel coils which are supplied only with cooling water. It is monitored continuously for alpha and beta-gamma contamination at the 281-4F monitor, then flows through the 281-1F delaying basin to the 281-2F pump basin, from where it is returned to the Building 285-F reservoir.
- The segregated cooling water accumulates from canyon vessel coils that are supplied with both cooling water and steam. It flows underground to the 281-6F monitor, into the 281-5F delaying basin, then is discarded to Four Mile Creek if within discard limits. In F-Area, there is only a single, flow-through basin, so the 281-6F monitor alone is relied on to detect contamination in the segregated cooling water.

If either monitor detects activity in the cooling water and the presence of contamination is confirmed, that stream is diverted to the lined retention basin 281-8F by pressing the diversion button in the control room or at the delaying basin for the affected stream. The retention basin can hold all the water that might be diverted for a major release of activity. However, if the circulated water is diverted for more than 30 minutes, a low water level may be reached in the Building 285-F reservoir, causing the segregating valve to close and leaving only the independent cooling water supply headers pressurized.

The maximum credible release of activity to the cooling water from a leaking vessel coil or evaporator steam chest has been calculated to be about 100,000 curies in F-Area.

DELETED VERSION

Action plans for minimizing the release of radioactive materials include:

1. Constant pressure in the coils of vessels containing radioactive materials to prevent or minimize contamination of the segregated cooling water with process solutions. This is accomplished by a combination of administrative controls and design features. In vessels discharging cooling water to the segregated system, the automatic air pressurization provided by the Cash regulators and back pressure valves prevents loss of coil pressure. Positive operation of the air pressurization systems must be assured by monthly testing and calibration of the Cash regulators according to procedures.
2. The activity monitor at Building 281-6F must provide reliable and continuous monitoring of the segregated cooling water discharges. If a monitor malfunctions, manual samples must be taken and checked by Health Protection at regular intervals until monitor is restored. A trouble alarm in the control room signals electronic malfunction or low flow through the monitor.
3. When an activity release is detected and confirmed in the segregated cooling water system, the contaminated stream must be diverted to the 281-8F retention basin before a major release of activity to the environment occurs. Procedural control of this action is provided.
4. A functional check of the segregated cooling water monitor is performed routinely by Health Protection personnel.

6.1.15 Emergency Power Distribution

Emergency power from diesel generators is supplied to critical equipment, such as gang valves, instruments, lighting, air compressors, ventilation equipment, and the public address system. Other equipment is supplied emergency power from batteries. The emergency power system makes possible the orderly shut-down of Building 221-F if the loss of normal electrical power occurs. The main emergency power for Building 221-F is supplied by a 750-kva generator, driven by a 1080-hp General Motors diesel [REDACTED]. The diesel starts automatically from batteries. There is an identical power source in Building 292-F. Details of the emergency power distribution system are included in Section 3.2.4. Critical equipment that is supplied with emergency power is listed in Appendix A.

6.2 SURVEILLANCE REQUIREMENTS

A detailed description of surveillance requirements for the 200-Area, including F-Canyon, is presented in Reference 1. The objective of the requirements is to ensure the operability of systems, equipment, and components important to safety during F-Canyon operations. The requirements include:

DELETED VERSION

- Surveillance and testing of the safety systems as specified in procedures shall be performed at intervals no greater than those shown in Table 6-2.
- Auditable records of the surveillance and testing shall be maintained.
- Substandard equipment performance noted shall be corrected promptly.

6.3 REFERENCES

1. Operational Safety Requirements for 200-F and 200-H Separations Areas (Excluding Tritium and Waste Management). Internal Report DPW-85-101, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, SC, July 1985.
2. Savannah River Plant Safety Manual. Internal Report, E. I. du Pont de Nemours and Co., Savannah River Plant, Aiken, SC (1984).

7.0 QUALITY ASSURANCE

7.1 QUALITY ASSURANCE PLAN AND MANUAL

A Quality Assurance Plan, Reference 1, and Manual, Reference 2, provide general procedures for implementing the QA policy principles. The manual establishes methods, practices, and requirements for the Savannah River Quality Assurance Program. Although the QA requirements in the manual are specific, additional instructions and procedures in lower level documents are normally required in order to perform the QA functions. The particular organization performing the quality assurance function provides for compliance with its organization procedures and instructions.

7.2 PROCESS HAZARDS

SRP facilities, processes, projects, and equipment are assessed in accordance with Reference 3.

7.3 REFERENCES

1. Savannah River Quality Assurance Plan. DPSPM-SITE-1, March 4, 1987.
2. Savannah River Quality Assurance Manual. DPF-83-111-3, April 1984.
3. SRP Process Hazards Review Manual. DPSPM-GEN-13, June 1987.

8.0 GLOSSARY OF TERMS

Absorbed Dose	The amount of energy absorbed per unit of mass of irradiated material. Measured in rads.
Activity	The number of spontaneous nuclear disintegrations occurring in a given quantity of material during a suitable, small interval of time, divided by that interval of time.
Actinides	Radioactive chemical elements with atomic numbers greater than that of actinium (i.e., >88).
Airborne Activity	Radioactive materials which are present-in-air.
Alpha Decay	Radioactive decay in which an alpha particle is emitted. (This lowers the atomic number of the nucleus by two and its mass number by four.)
Alpha Radiation	An emission of particles (helium nuclei) from a material undergoing nuclear transformation.
Atom	A unit of matter consisting of a single nucleus surrounded by a number of electrons equal to the number of protons in the nucleus.
Attenuation	The reduction of a radiation quantity upon passage of radiation through matter resulting from all types of interaction with that matter.
Background (Radiation)	Ionizing radiation present in the region of interest and coming from sources other than that of primary concern.
Beta Decay	Radioactive decay in which a beta particle is emitted or in which orbital electron capture occurs.
Beta Particle	An electron, of either positive or negative charge, which has been emitted by an atomic nucleus or neutron in a nuclear transformation.

Beta Radiation	Electrons and positrons emitted from the nucleus of an atom undergoing nuclear transformation.
Body Burden	The total quantity of radionuclide present in the body.
Body Burden, maximum permissable (MPBB)	That body burden of a radionuclide that, if maintained at a constant level, would produce the maximum permissable dose equivalent in the critical organ.
Bone Seeker	Any substance which migrates, <u>in vivo</u> , preferentially into bone.
X/Q	Chi/Q is the calculated air concentration due to a specific air release. Its units are (Ci/m ³ /Ci/sec), or (sec/m ³).
Contamination	Radioactive materials located where it is undesirable (outside its normal location).
Controlled Area	A specified area in which exposure of personnel to radiation or radioactive material is controlled and that is under the supervision of a person who has knowledge of the appropriate radiation protection practices. These practices include pertinent regulations, and who has responsibility for applying them.
Cooling (Radioactive)	The reduction of radioactivity of a material by radioactive decay.
Critical Mass	The minimum mass of fissionable material that can be made critical with a specified geometrical arrangement and material composition.
Critical Organ	That organ (or tissue) in which the dose equivalent would be most significant due to a combination of the organic radiosensitivity and a particular dose pattern through the body.
Curie	A unit of radioactivity equal to 3.7×10^{10} disintegrations per second.

Decay, Radioactive	Disintegration of the nucleus of an unstable nuclide by the spontaneous emission of charged particles and/or photons.
Decontamination	The removal of radioactivity material from a surface or from within.
Decontamination Factor (DF)	The ratio of the amount of undesired radioactive material initially present to the amount remaining after a process step.
Decommissioning	The measures taken at the end of the facilities operating lifetime.
Dose, Absorbed	The energy imparted to matter in a volume element by ionizing radiation, divided by the mass of irradiated material in that volume element.
Fissile	A nuclide, capable of undergoing fission by interaction with slow neutrons.
Fission Products	Nuclides produced either by fission or by the subsequent radioactive decay of the nuclides thus formed.
Fissionable	A nuclide capable of undergoing fission by any process.
Fuel Assemblies	Fissionable material placed in a specific form for use as the source of power in a nuclear reactor.
Half-life, Biological	The time required for the amount of a particular substance in a biological system to be reduced to one-half of its original value by biological processes when the rate of removal is approximately exponential.
Individual Dose	The dose to a hypothetical individual located at the point of interest.

Ion Exchange	Process for selective removal of a constituent from a particular solution. The process reversibly transfers ions between an insoluble solid and the fluid stream.
Ionization	Any process by which an atom, molecule, or ion gains or loses electrons.
Man-rem Dose (person-rem)	A measure of population dose. It is calculated by summing the dose received by each person in the population discussed.
Neutrons	An elementary particle having no electric charge, a rest mass of 1.67×10^{-27} kg, and a mean life of 1000 seconds.
Neutron Radiation	Emission of a neutron from a material during a nuclear transition.
Nuclear Materials	Materials produced as a result of a nuclear reaction such as those produced in a nuclear reactor.
Nuclear Reactor	Equipment in which a chain reaction of fissionable material is initiated and controlled.
Nuclide	A species of atom having a specific mass, atomic number, and nuclear energy state.
Offsite	Beyond the boundary line marking the limits of the plant property.
Onsite	Within the boundary line marking the limits of the plant property.
Quality Assurance	The systematic actions to provide adequate confidence that a material, component system, process, or facility performs satisfactorily or as planned in service.
Rad	The energy imparted to matter by ionizing radiation. One rad is the absorption of 100 ergs/gram of absorbing material.

Radioactive

A Radioactive nuclide.

Safety Related Item

An item providing information that may prevent or mitigate the frequency or consequence of accidents.

Transuranics

Nuclides having an atomic number greater than uranium (i.e., >92).

SYMBOLS AND ABBREVIATIONS

Atomic Energy Division	AED
Curie	Ci
Design Basis Earthquake	DBE
Design Basis Tornado	DBT
Decontamination Factor	DF
Department of Energy	DOE
Disintegrations per Minute	d/m
Disintegrations per Minute per Milliliter	d/m/ml
Di-Octylphthalate	DOP
Du Pont Savannah Operations Log	DPSOL
Du Pont Savannah River Operations Procedure	DPSOP
Defense Waste Processing Facility	DWPF
Emergency Operating Coordinator	EOC
Equipment Piece	EP
Facility Coordinator	FC
Hydroxylamine Nitrate	HAN
High Activity Waste	HAW
High Efficiency Particulate Air	HEPA
Health Monitoring	HM
Low Activity Waste	LAW
Motor Control Center	MCC
Maximum Resistance Construction	MRC
Materials Test Facility	MTF
Nuclear Incident Monitor	NIM
Normal Paraffin Hydrocarbons	NPH
Nuclear Safety Blanks	NSB

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Nuclear Safety Review Committee	NSRC
Oak Ridge National Laboratory	ORNL
Operational Safety Requirement	OSR
Public Address	PA
Process Control Center	PCC
Program Management Team	PMT
Products of Combustion	POC
Primary Recovery Column	PRC
Process Vessel Vent	PVV
Quality Assurance	QA
Roentgen per hour	R/hr
Regulated Area	RA
Roentgen Absorbed Dose	rad
Receiving Basin for Offsite Fuels	RBOF
Radiation Control Guide	RCG
Roentgen Equivalent Man	rem
Rocky Flats Scrub Alloy	RFSA
Resistance Temperature Device	RTD
Radiation Zone	RZ
Silica Gel	SG
Savannah River Laboratory	SRL
Savannah River Plant	SRP
Test Authorization	TA
Tributyl Phosphate or Tri-n-butyl Phosphate	TBP
Thermoluminescent Dosimeter	TLD
Thermoluminescent Neutron Dosimeter	TLND

Threshold Limit Value

TLV

Transuranic

TRU

Use Every Time

UET

Unusual Incident

UI

APPENDIX A

F-CANYON EMERGENCY POWER TO CRITICAL EQUIPMENT

EMERGENCY POWER

Motor control panels are grouped in motor control centers [REDACTED] and [REDACTED] on first level, sections [REDACTED] and [REDACTED] respectively.

Motor control centers are divided into separate panels controlling all individual equipment (except for instrument air compressors which take power directly from the emergency bus). Each panel contains fuses and a switch to energize its associated equipment. Actual starting must be done at the equipment or control room. An automatic overload control de-energizes the equipment. Some of these controls must be reset manually at the MCC panel.

The emergency power to critical equipment is listed in Table A-1.

TABLE A-1. Emergency Power to Critical Equipment

[REDACTED] - Emergency Power Feeder [REDACTED]

Fire Alarm Box 15 must be bypassed before [REDACTED] is de-energized. Instrument Air Compressor 2 will not operate when [REDACTED] is de-energized.

Air conditioning unit [REDACTED] electric shop
Warm Canyon shielding door, motor [REDACTED], and Warm Canyon removal rail drive motor

Room [REDACTED]-hood exhaust fan, DA202-50-1 [REDACTED]
Warm Canyon crane, [REDACTED]
Hot Canyon shielding door, motor [REDACTED], and Hot Canyon removal rail drive motor.

Instrument air dryer ([REDACTED])
Battery charger for diesel unit [REDACTED]
Cooling unit for diesel [REDACTED]
To FB-Line for emergency lights, and instrument panel.

Exhaust blower [REDACTED])
ATF-FPDF cabinet exhaust fan motor [REDACTED]
Instrument power panel AB [REDACTED]. Panel AB supplies power to the motor-operated dampers on the canyon supply fans. Loss of power to damper motors automatically cuts off supply fans.
Emergency light panels EMA ([REDACTED]), and EMC [REDACTED] and fire box 15 ([REDACTED]).
Emergency light panels EME [REDACTED] and EMK [REDACTED]
Panel EME supplies power to locks on personnel and RR access gates at Building 221-F, RR cut.

Welding receptacle [REDACTED]
Plant air compressor [REDACTED]

Recirculating PP 7-1/2 hp variable speed [REDACTED]
Refrigeration machine [REDACTED]
Refrigeration machine [REDACTED]
Emergency power to lighting panel X and radio equipment via transfer switch on [REDACTED]

TABLE A-1. Emergency Power to Critical Equipment (Continued)

Emergency Power Feeder

Chem pump ATF scrubbers
 Feed for TC change shack transfers
 Hot Canyon crane

ATF-FPDF cabinet exhaust fan motor
 Spare
 Exhaust Fan #2
 ATF chem pump
 Swimming pool equipment (recirculating pump, sump Gang Valve, 440-V disconnect,
 Plant air compressor

Feeder 1-1A from Transfer EP

182 sump pump
 183 sump pump
 440 V to Warm Canyon shop (east wall)

Agitator tank 181
 Pump tank 181

Grinder and drill press in maintenance shop
 181 sump pump

Spare
 Swimming pool jib crane A/C unit, CFP

Feeder from 200-amp Disconnect at

Water chiller
 3/4-hp grinder
 3/4-hp drill press
 Lighting transformer

2-hp blower
 1 1/2-hp blower
 1 1/2-hp exhauster
 Welder

^aIn crane maintenance area.

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] - Emergency Power Feeder [REDACTED]

Gang Valve 5.5 to 7.5
 Gang Valve 5.5 to 6.8 or spray
 Open

Gang Valve 7.7E to 6.8
 Gang Valve 7.6E to P.O.
 Open
 Emergency lighting panel EMH [REDACTED]

Gang Valve 7.5 to 9.6
 Gang Valve 7.6E to 6.8
 Open

Gang Valve 7.5 B-1-1 spare (7W) (Dual Gang Valve)
 Gang Valve 9.7 to 8.7
 Open

Open
 Open
 Open
 Open
 Open

Instrument power panel AC [REDACTED]
 Gang Valve 7.8 spray

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] Emergency Power Feeder [REDACTED]

Gang Valve 6.8 to Bldg 241-F, header 3
Gang Valve 8.5E to 7.8
Open

Spare
Gang Valve spare (8W) 8.5E reboiler to 8.5E BT and 8.5 condensate tank to B2-1.
Open

Gang Valve 7.8 to 6.8 (alternate to PO)
Gang Valve 6.8 spray
Open

Gang Valve 7.8 to 7.6E
10-in. motor-operated valve [REDACTED]

Gang Valve 9.7 to P.O.
Gang Valve 6.5 to 12.2
Open

Gang Valve 6.8 to Bldg 241-F, header 4
Supply Blower [REDACTED] Fan [REDACTED]

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] - Emergency Power Feeder [REDACTED]

Gang Valve 9.5 to 9.6
Gang Valve 9.5 to 9.8
Open

Gang Valve 5.3 to 6.4D
Gang Valve spare (9W)
Open

Gang Valve 9.5 to 9.7
Gang Valve 8.7 to P.O.
Open

Fence lighting transformer 3
Gang Valve 7.8 to 7.7E
Open

Gang Valve 9.6 to 12.5
Gang Valve spare (9W)
Open

Open

6.6 agitator

Gang Valve 10.8 to 9.5
Gange Valve 10.8 to 9.8

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

	- Emergency Power Feeder
--	--

Gang Valve 10.5 spray
 Gang Valve 10.5 to PO
 Open

Gang Valve 9.8 to 9.5
 5.3 agitator
 Spare

Open
 Open
 Open
 Open
 Open

Gang Valve 9.8 spray
 Gang Valve 9.8 to 10.2
 Open

Gang Valve 9.8 to
 Gang Valve 9.8 to
 Open

Spare
 Open
 Open
 Open

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] - Emergency Power Feeder [REDACTED]

Gang Valve 6.1D to 8.1, No. 1
Gang Valve 6.1D to 5.3

Gang Valve 6.1D to 5.2
Gang Valve 6.1D to 8.1, No. 2

Gang Valve 6.4D to 8.1, No. 1

Gang Valve 6.4D to 5.3
Gang Valve 6.4D to 8.1, No. 2

Gang Valve 6.4D to 5.2
Gang Valve 5.3 to 5.2

Instrument Power Panel AG-1
Gang Valve 13.3 to 12.2
13.3 S agitator
13.3 N agitator
Gang Valve 5.3 to 6.1D

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] Emergency Power Feeder [REDACTED]

Gang Valve 7.4 to 8.3
Gang Valve 7.4 spray
Open

Gang Valve 8.1 to 12.2
Gang Valve 5.8-3 to 6.6, 8.3, or 6.8
Open

Gang Valve 5.2 to Bldg 241-F, header 4
Gang Valve 5.8-3 to 12.6 (Purex) or 5.8-1
Open

12-in. motor-operated valve [REDACTED]
Gang Valve 5.8-8 to product tank or 5.5
Open

Gang Valve section-8 Hot Canyon sump to tank 820
Gang Valve 5.2 to Bldg 241-F, header 3
Open

Gang Valve 6.6 to 8.3
Gang Valve section-5 Hot Canyon sump to tank 820
Open

Alternate feed to section 9 condensate pump
Gang Valve 7.3 to 8.3

TABLE A-1. Emergency Power to Critical Equipment (Continued)

Emergency Power Feeder

Gang Valve section-7 Hot Canyon sump to tank 820
Gang Valve section-6 Hot Canyon sump to tank 820
Open

Spare
Gang Valve section-10 Hot Canyon sump to tank 820
Open

Gang Valve section-14 Hot Canyon sump to tank 820
Gang Valve 8.3 to PO
Open

Gang Valve 9.1E reboiler to PO (8.3)
Gang Valve 9.3E reboiler to 9.3 BT
Open

Gang Valve 10.1 to Bldg 241-F, header 1
Gang Valve 10.1-2 to 6.5
Open

Gang Valve section-15 Hot Canyon sump to tank 820
Gang Valve 10.1 to 8.3
Spare

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] - Emergency Power Feeder [REDACTED]

Gang Valve 10.4 to 11.1C "A"
Gang Valve 11.3E sparge
Open

Gang Valve section-9 Hot Canyon sump to tank 820
Gang Valve 10.4 to 11.1C "B"
Open

Gang Valve 10.4 to P.O.
Gang Valve 10.2 to 11.2
Open

Gang Valve 10.2 to 11.2
Gang Valve 10.1 to 11.3E
Open

Gang Valve 10.1 to Bldg 241-F, header 2
Open
Open
Open

Gang Valve 10.2 to 11.2
Gang Valve section-13 Hot Canyon sump to tank 820
Open

[NOTE] Alternate power is supplied to Instrument Power Panels AD, AF, and AG
[REDACTED] from the [REDACTED] bus, via fused disconnect on north wall of ECR [REDACTED]
and switchgear in ECR [REDACTED]

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

Emergency Power Feeder

Gang Valve 11.7 to 12.5
Spare Power to 6.8, 12.1 IBX meter pumps and 13C the pump
Instrument power panel AD

Gang Valve 11.7 to 8.7
Gang Valve 11.8 to 12.5
Open

Gang Valve 11.8 to 11.5M
Gang Valve 11.8 spray
Open
Emergency lighting panel EMB

Gang Valve 10.8 CS to 10.8 aqueous or 10.8 organic to PO
Gang Valve 12.5 to 12.6
12.5 to 11.2
Open
Emergency lighting panel EMF (level 1, section 12)

Gang Valve 12.5 to 11.5M
Gang Valve 12.5 to 11.8
Open

Instrument power panel AF WCCR Panelboard
Instrument power panel AG HCCR Panelboard
Open
Open
Open

[NOTE] Alternate power is supplied to Instrument Power Panels AD, AF, and AG from bus.

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

Emergency Power Feeder

Gang Valve 9.7 spray
Gang Valve 12.6 spray
Open

Gang Valve 12.6 to 12.5
Gang Valve 12.6 to 11.8

Spare
Open
Open
Open

Open
Open
Spare
Open

Supply Blower No. 1
Open
Open

Open
Open
Open
Open
Open

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] Emergency Power Feeder [REDACTED]

Spare
Gang Valve 12.8 to 13.5
Open

Gang Valve 12.8 to spray
Gang Valve 13.1 to P.O.
Open

Spare
Spare
Open

Gang Valve 13.5 to PO
Gang Valve 13.5 to tank 901
Open

Gang Valve 13.5 to 13.6-2S
Gang Valve 13.6-1S organic to 13.7
Open

Gang Valve 13.6-2S organic to 13.7
Open
Open
Open

[REDACTED] - Emergency Power Feeder [REDACTED]

Gang Valve 13.7 to spray
Open
Open
Open

Gang Valve 13.7 to 17.5
Gang Valve 13.7 to 7.8
Open

Gang Valve 13.8 to 14.5-1
Gang Valve 13.8 to tank 901
Open

Gang Valve 13.8 to 13.6-1S
Gang Valve 13.8 spray
Open

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] - Emergency Power Feeder [REDACTED]

Gang Valve 14.5 stage 1 to 13.8 (alternate PO)
 Gang Valve 14.7 to 13.8
 Open

Welding receptacle in level-4 fan room
 Gang Valve 14.5 stage 2 to 13.8 (alternate PO)
 Open

Spare
 Gang Valve 14.8 spray
 Open

Spare
 Gang Valve 14.8 to 13.8
 Open

Gang Valve 14.5-1 agit mix can to 14.5-1
 14.5-2 to 14.5-1
 Open
 Open
 Open

[REDACTED] - Emergency Power Feeder [REDACTED]

Gang Valve 14.2M aqueous decant to pump tank
 13.3M organic decant to pump tank
 Gang Valve spare
 Open

Gang Valve 16.4 to 17.1
 Open

Open
 Open
 Open

Open
 Open
 Open
 Open
 Open

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

- [REDACTED] - Emergency Power Feeder [REDACTED]
--

Gang Valve 11.3E to 12.1
Gang Valve section-18 HC sump to tank 820
Open

Gang Valve 10.3 to 11.2
11.2 to 13.1

Gang Valve section 11 hot sump to tank 820

Gang Valve 11.2 to 13.1
Gang Valve spare (10H)
Open

Gang Valve 11.1C to 12.1
Open
Open
Open

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

Emergency Power Feeder

Gang Valve 11.4 to 12.3CC

Gang Valve section-16 Hot Canyon sump to tank 820

Open

Gang Valve to 11.4 P.O.

Gang Valve 16.3 to 16.1

Open

Gang Valve 15.4 to Bldg 241-F, header 1

Gang Valve 12.2 to 10.4

Open

12 H sump to 12.3 CC

Gang Valve 12.1 to Bldg 241-F, header 2

Open

Gang Valve 15.4 spray

Gang Valve 12.1 to Bldg 241-F, header 1

Open

Gang Valve 12.1 to 241

Open

Open

Open

DELETED VERSION

DPSTSA-200-10, SUP-4

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] - Emergency Power Feeder [REDACTED]

Gang Valve spare (12H)
Gang Valve spare (12H)
Open

Gang Valve spare (16H) 16.3 to 17.3-1S
Gang Valve 16.3 to spray
Open

Gang Valve 1AP decant to pump tank (12H)
Gang Valve section-12 Hot Canyon sump to tank 820
Open

Gang Valve 1AW decant to pump tank (12H)
Gang Valve 13.3M aqueous decant to pump tank
Open

Gang Valve 16.4 to 11.4
Gang Valve spare (16H)
Open

Gang Valve 16.4 to 16.3 (direct)
Open
Open
Open

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] Emergency Power Feeder [REDACTED]

Welding receptacle cold shop [REDACTED]

Spare

Spare

Gang Valve 17.7E bottoms tank to 18.5

B-4-2 pump to tank 820 [REDACTED]

Instrument power panel AE [REDACTED]

Segregated CW pump, [REDACTED]

Segregated CW pump, [REDACTED]

Power to 251-F control house for MG sets and air compressor (emergency power)

RIX-1 feed pump [REDACTED]

Gang Valve 17.7 reboiler to bottoms tank

To 211-F control house instrument power

Gang Valve 18.7 to 18.6 (Z6)

Gang Valve 18.7 to 18.6 (Z5)

Open

Gang Valve 18.7 to 17.5

Gang Valve 18.7 spray

Open

Spare

Gang Valve 18.8 to B-3-2

Open

DELETED VERSION

DELETED VERSION

DPSTSA-200-10, SUP-4

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] - Emergency Power Feeder [REDACTED]

Gang Valve 18.6E to 241-F header 3
Open

Gang Valve 15.5M organic decant to pump tank
Gang Valve 18.6 to 18.7
Open

GE (Li) tube positioner [REDACTED]
GE (Li) tube positioner [REDACTED]
Open

Spare
Gang Valve 15.5M aqueous decant to pump tank
Open

Fire water valve [REDACTED]
Fire water valve [REDACTED] Warm Canyon
Open

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

Emergency Power Feeder	
Lighting substation EML (440-V, 3-phase)	
Open	
Switch for 15-k VA transformer to power supply cabinet for computer system in	
Open	
Gang Valve 18.8 spray	
Gang Valve 18.8 to 18.7	
Open	
Instrument panel AH (440-V, 3-phase)	
Gang Valve 18.5 to PO	
Open	
Open	
Open	
Gang Valve 15.7M aqueous decant to pump tank	
Open	
Vacuum pump	
Spare	
Open	
Open	
Open	
Spare	
Open	
Gang Valve 16.7 to 13.1	
Gang Valve 17.5 to PO	
Open	
Gang Valve 17.5 to solvent loadout via nozzle 82A (17H)	
Alt to 16.4	
Gang Valve 17.5 to 16.3	
Open	
Spare	
MPPF Exhaust Fan	
Open	

DELETED VERSION

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[REDACTED] - Emergency Power Feeder [REDACTED]	
Gang Valve 17.1 to 17.3E	
Gang Valve 17.1 to 16.4	
Open	
Gang Valve 17.1 to 16.1	
Gnag Valve MPPF waste to 17.2	
Open	
[REDACTED] - Emergency Power Feeder [REDACTED]	
Hot Canyon fire water valve HGVC	
Section-17H sump to tank 820	
Open	
Gang Valve 17.3E to 17.1	
Segregated Hot Water circ pump 3-hp [REDACTED]	
Open	
Spare for MPPF	
Gang Valve Rack pan sump to 17.1	
Open	
Gang Valve 14.2M organic decant to pump tank	
Gang Valve 15.1M aqueous decant to pump tank	
(alternate to POH)	
Gang Valve 15.4 to 11.4	
Open	
Gnag Valve 15.1M organic decant to pump tank	
Spare	
Open	
To starter rack 211-F-12	
Open	
Gang Valve 15.4 to 17.1	
Open	

TABLE A-1. Emergency Power to Critical Equipment (Continued)

<div>Emergency Power Feeder</div>	
Gang Valve 16.3 to 16.4 Gang Valve 820 cell sump to tank 820 to nozzle 54, section 16, HGVC Open	
Gang Valve 15.4 to 12.2 Gang Valve 15.4 to Bldg 241-F, header 2 Open	
Gang Valve 16.2 to B-2-2 Gang Valve 16.1E to 17.1 Open	
Segregated Hot Water circ pump, 3-hp HGVC, section 17, south pump Gang Valve 16.1E to 16.3 Open	
Gang Valve 16.2 to 15.4 Hot Canyon fire water valve Open	
Gang Valve 17.2 to 15.4 Gang Valve 17.2 to 17.1 Open	
Main switch for (440-V) Process air compressor No. 1 Open Process air dryer, 5-hp	
Process air compressor No. 2 Spare Open	

APPENDIX B

F-CANYON EVENT TREES AND FAULT TREE

EXTERNALLY INDUCED FAILURE INITIATOR	VESSEL CONTENTS LOST	VENTILATION SYSTEM FAILS	SAND FILTER FAILS	SEQUENCE FREQUENCY /hr	RELEASE Ci	RELEASE RATE Ci/hr**
YES ↑ ↓ NO				1.0×10^{-8}	0	0
				0	TABLE C-2	0
				1.0×10^{-6} *	TABLE C-2	*
				NA	0	0
				NA	0	0

* SEQUENCE FREQUENCY IS 1×10^{-6} FOR EACH UNIT OPERATION

** RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

FIGURE B-1 Event Tree For Externally Induced Failure Releases from the Stack.

MEDIUM ENERGETIC EVENT INITIATOR	VENTILATION SYSTEM FAILS	SAND FILTER FAILS	SEQUENCE FREQUENCY /hr	RELEASE Ci	RELEASE RATE Ci/hr*
YES ↑↓ NO	< 0.01/d TABLE B-1		—	0	0
			0	TABLE C-3, C-4, C-5	0
	0.99/d TABLE B-1		TABLE B-1	TABLE C-3, C-4, C-5	*
	~1 TABLE B-1		NA	0	0

* RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

FIGURE B-2 Event Tree for Medium Energetic Release from the Stack

TABLE B-1. Data for Quantification of Medium Energetic Release from the Stack Event Tree

Initiator Location	Medium Energetic Event Initiator	Sequence Frequency, /hr	Release, Ci	Release Rate, Ci/hr
Dissolver	Uncontrolled Reaction	3.0E-05	2.3E-01	6.9E-06
Head End	Uncontrolled Reaction	5.0E-06	2.1E-01	1.0E-06
First Cycle	Fire	7.0E-08	4.6E-01	3.2E-08
	Uncontrolled Reaction	3.3E-07	3.0E-01	9.9E-08
2nd U Cycle	Fire	7.0E-08	3.6E-06	2.5E-13
	Uncontrolled Reaction	3.3E-07	1.4E-07	4.6E-14
2nd Pu Cycle	Fire	7.0E-08	3.5E-03	2.4E-10
	Uncontrolled Reaction	3.3E-07	1.3E-04	4.4E-11
HAW Evaporator	Uncontrolled Reaction	7.2E-05	6.0E-01	4.3E-05
Low Evaporator	Uncontrolled Reaction	1.3E-05	6.3E-06	1.2E-10
Ion Exchange	Fire	3.0E-07	9.1E-01	3.2E-07
	Uncontrolled Reaction	6.0E-05	7.9E-01	4.7E-05
High Heat Waste	Uncontrolled Reaction	5.0E-06	6.0E-01	3.0E-06

HEAT SOURCE AVAILABLE	SOLVENT OUTSIDE CONTACTORS	SOLVENT HEATED ABOVE FIRE POINT	IGNITION SOURCE PRESENT	SEQUENCE FREQUENCY /hr	RELEASE Ci	RELEASE RATE Ci/hr**
<div>YES ↕ NO</div>	<div><div>0.1/d</div><div>0.1/d</div><div>0.9/d</div><div>0.9/d</div><div>0.93/d</div></div>			2.1×10^{-7} *	TABLE C-3	**
				NA	0	0
				NA	0	0
				NA	0	0
				NA	0	0

* Each cycle assumed to have equal probability of fire (7×10^{-8} /hr)

** RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

FIGURE B-3 Event Tree for Fire in Solvent Extraction

RESIN LEAK OUTSIDE COLUMN	RESIN DRIES OUT	IGNITION SOURCE PRESENT	SEQUENCE FREQUENCY /hr	RELEASE CI	RELEASE RATE CI/hr*
YES ↔ NO	0.1/d		9.0×10^{-7}	TABLE C-3	*
	< 0.01/d		NA	0	0
	0.9/d		NA	0	0
	0.99/d		NA	0	0

* RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (CI)

FIGURE B-4 Event Tree for Fire in Ion Exchange

CRITICALITY	VENTILATION SYSTEM FAILS	SAND FILTER FAILS	SEQUENCE FREQUENCY /hr	RELEASE Ci	RELEASE RATE Ci/hr*
YES ↑ ↓ NO	3.6×10^{-7} /hr	< 0.01/d	3.6×10^{-9}	0	0
		0	0	TABLE C-5	0
		0.99/d	3.6×10^{-7}	TABLE C-5	*
			NA	0	0

* RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

SYSTEM	SEQUENCE FREQUENCY /hr
DISSOLVING	1.2×10^{-10}
HEAD END	2.0×10^{-9}
FIRST CYCLE	1.8×10^{-7}
2ND Pu CYCLE	1.8×10^{-7}
TOTAL	3.6×10^{-7}

FIGURE B-5 Event Tree for Criticality Release to the Stack

TRANSFER ERROR INITIATOR	EVENT INVOLVES CONTAMINATED MATERIAL	MATERIAL TRANSFERRED OUTSIDE CANYON	MATERIAL RELEASED OUTSIDE VESSEL	SEQUENCE FREQUENCY /hr	RELEASE Ci	RELEASE RATE Ci/hr*
<p>YES ↕ NO</p>	<p>0.01/d</p> <p>TABLE B-2</p>			TABLE B-2	TABLE C-6	*
	<p>TABLE B-2</p>			NA	0	0
	<p>TABLE B-2</p>			NA	0	0
	<p>TABLE B-2</p>			NA	0	0
	<p>TABLE B-2</p>			NA	0	0

* RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

FIGURE B-6 Event Tree for Air borne Release at Ground Level due to Transfer Error

TABLE B-2. Data for Quantification of Airborne Release at Ground Level Event Tree

Initiator Location	Fraction	Initiator Frequency, /hr	Event Involves Contaminated Material, /d	Material Transferred Outside, /d	Material Released Outside Vessel, /d	Sequence Frequency, /hr	Release, Ci	Release Rate, Ci/hr
Decontamination	1.00	5.0E-06	5.0E-01	1.2E-01	1.0E-02	3.0E-09	9.9E-05	3.0E-13
Dissolver	1.00	6.0E-05	5.0E-01	4.0E-02	1.0E-02	1.2E-08	5.4E+01	6.5E-07
Head End	1.00	5.0E-06	1.0E+00	4.0E-02	1.0E-02	2.0E-09	4.8E+01	9.6E-08
First Cycle	0.25	7.0E-05	5.0E-01	4.0E-02	1.0E-02	3.5E-09	4.8E+01	1.7E-07
2nd 0 Cycle	0.15	7.0E-05	5.0E-01	4.0E-02	1.0E-02	2.1E-09	3.9E-09	8.2E-18
2nd Pu Cycle	0.60	7.0E-05	5.0E-01	4.0E-02	1.0E-02	8.4E-09	2.5E-04	2.1E-12
MAN Evaporator	0.50	5.0E-05	1.0E+00	4.0E-02	1.0E-02	1.0E-08	1.4E+02	1.4E-06
LAN Evaporator	0.50	5.0E-05	1.0E+00	4.0E-02	1.0E-02	1.0E-08	4.8E-06	4.8E-14
Ion Exchange	1.00	4.0E-05	5.0E-01	4.0E-02	1.0E-02	8.0E-09	1.3E+01	1.0E-07
High Heat Waste	1.00	5.0E-06	1.0E+00	4.0E-02	1.0E-02	2.0E-09	1.4E+02	2.8E-07

* RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (CI)

FIGURE B-7 Event Tree for Low Energetic Event Release to Stack

TABLE B-3. Data for Quantification of Airborne Release from the Stack due to Transfer Error Event Tree

Initiator Location	Fraction	Initiator Frequency, /hr	Event Involves Contaminated Material, /d	Material Released to Sump, /d	Sequence Frequency, /hr	Release, Ci	Release Rate, Ci/hr
Decontamination	1.00	5.0E-06	5.0E-01	1.5E-01	3.8E-07	4.8E-07	1.8E-13
Dissolver	1.00	6.0E-05	5.0E-01	1.5E-01	4.5E-06	2.6E-01	1.2E-06
Head End	1.00	5.0E-06	1.0E+00	1.5E-01	7.5E-07	2.4E-01	1.8E-07
First Cycle	0.25	7.0E-05	5.0E-01	1.5E-01	1.3E-06	2.4E-01	3.1E-07
2nd U Cycle	0.15	7.0E-05	5.0E-01	1.5E-01	7.9E-07	1.9E-11	1.5E-17
2nd Pu Cycle	0.60	7.0E-05	5.0E-01	1.5E-01	3.1E-06	1.2E-06	3.8E-12
HAN Evaporator	0.50	5.0E-05	1.0E+00	1.5E-01	3.8E-06	7.0E-01	2.6E-06
LAN Evaporator	0.50	5.0E-05	1.0E+00	1.5E-01	3.8E-06	2.4E-08	9.0E-14
Ion Exchange	1.00	4.0E-05	5.0E-01	1.5E-01	3.0E-06	6.4E-02	1.9E-07
High Heat Waste	1.00	5.0E-06	1.0E+00	1.5E-01	7.5E-07	7.0E-01	5.3E-07

TABLE B-4. Data for Quantification of Airborne Release from the Stack due to Overflow Event Tree

Initiator Location	Fraction	Initiator Frequency, /hr	Event Involves Contaminated Material, %	Material Released to Sump, /d	Sequence Frequency, /hr	Release, Ci	Release Rate, Ci/hr
Decontamination	1.00	1.0E-06	5.0E-01	1.0E+00	5.0E-07	2.8E-07	1.4E-13
Dissolver	1.00	5.0E-06	1.0E+00	1.0E+00	5.0E-06	1.5E-01	7.5E-07
Head End	1.00	5.0E-06	1.0E+00	1.0E+00	5.0E-06	1.4E-01	7.0E-07
First Cycle	0.40	1.3E-04	4.5E-01	1.0E+00	2.3E-05	1.4E-01	2.5E-06
2nd U Cycle	0.15	1.3E-04	4.5E-01	1.0E+00	8.8E-06	1.0E-10	6.8E-16
2nd Pu Cycle	0.45	1.3E-04	4.5E-01	1.0E+00	2.6E-05	7.2E-07	1.5E-11
HAN Evaporator	0.50	5.0E-05	1.0E+00	1.0E+00	2.5E-05	4.1E-01	1.0E-05
LAN Evaporator	0.50	5.0E-05	1.0E+00	1.0E+00	2.5E-05	1.4E-08	3.5E-13
Ion Exchange	1.00	4.0E-05	1.0E+00	1.0E+00	4.0E-05	2.2E-01	8.8E-06
High Heat Waste	1.00	4.0E-05	1.0E+00	1.0E+00	4.0E-05	4.1E-01	1.6E-05

TABLE B-5. Data for Quantification of Airborne Release from the Stack due to Leak Event Tree

Initiator Location	Fraction	Initiator Frequency, /hr	Event Involves Contaminated Material, /d	Material Released to Sump, /d	Sequence Frequency, /hr	Release, Ci	Release Rate, Ci/hr
Decontamination	1.00	5.0E-04	5.0E-01	1.0E+00	2.5E-04	8.9E-08	2.2E-11
Dissolver	1.00	1.3E-04	1.0E+00	1.0E+00	1.3E-04	2.2E-02	2.2E-06
Head End	1.00	8.0E-04	1.0E+00	1.0E+00	8.0E-04	1.9E-02	1.5E-05
First Cycle	0.48	5.4E-03	5.0E-01	1.0E+00	1.3E-03	1.9E-02	2.3E-05
2nd U Cycle	0.13	5.4E-03	5.0E-01	1.0E+00	3.5E-04	1.6E-12	5.2E-16
2nd Pu Cycle	0.39	5.4E-03	5.0E-01	1.0E+00	1.1E-03	1.0E-07	9.8E-11
MAN Evaporator	0.50	7.3E-03	5.0E-01	1.0E+00	1.8E-03	5.8E-02	1.1E-04
LAN Evaporator	0.50	7.3E-03	5.0E-01	1.0E+00	1.8E-03	2.0E-09	3.7E-12
Ion Exchange	1.00	9.0E-04	1.0E+00	1.0E+00	9.0E-04	7.6E-02	6.8E-05
High Heat Waste	1.00	1.0E-03	1.0E+00	1.0E+00	1.0E-03	5.8E-02	5.8E-05

INITIATING EVENT	INITIATOR FREQUENCY, /hr	FREQUENCY FOR SUBSEQUENT EVENTS, /d				SEQUENCE FREQUENCY, /hr	
						A	B
OVERFLOW	1.4×10^{-4}	0.46	0.33	0.07	0.85	1.5×10^{-6}	1.7×10^{-5}
LEAK	2.2×10^{-3}	0.75	0.089	0.22	0.70	3.2×10^{-5}	8.0×10^{-5}
TRANSFER ERROR	2.4×10^{-4}	0.15	0.33	0.07	0.85	8.3×10^{-7}	9.4×10^{-6}
UNCONTROLLED REACTION	3.8×10^{-5}	0.42	0.29	0.57	0.86	2.6×10^{-6}	1.7×10^{-6}
TOTAL						3.7×10^{-5}	1.1×10^{-4}

FIGURE B-8 Internal Release Event Tree for Chemical Hazards

FIGURE B-9 Fault Tree for Liquid Releases to Four Mile Creek

FIGURE B-9 Fault Tree for Liquid Releases to Four Mile Creek (continued)

APPENDIX C

F-CANYON SOURCE TERM CALCULATIONS

C-2

TABLE C-1. Curies of Specific Nuclides Released at Ground Level due to Earthquake (F-Canyon)

Nuclide	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Pu Cycle	MAN Evaporator	LAN Evaporator	Ion Exchange	High Heat Waste
89Sr	1.54E-03	1.40E-03	1.40E-03	1.10E-13	1.02E-09	2.49E-03	6.39E-11	2.28E-03	2.49E-03
90Sr	1.46E-04	1.33E-04	1.33E-04	1.06E-14	9.98E-11	2.39E-04	6.05E-12	2.20E-04	2.39E-04
90Y	1.46E-04	1.33E-04	1.33E-04	1.06E-14	9.98E-11	2.39E-04	6.05E-12	2.20E-04	2.39E-04
91Y	2.37E-03	2.15E-03	2.15E-03	1.73E-13	1.63E-09	3.83E-03	9.67E-11	3.51E-03	3.83E-03
95Zr	2.37E-03	2.15E-03	2.15E-03	8.64E-13	1.63E-08	3.83E-03	9.33E-10	3.51E-03	3.83E-03
95Nb	6.32E-04	5.74E-04	5.74E-04	2.29E-13	1.02E-09	1.02E-03	7.08E-11	9.37E-04	1.02E-03
103Ru	1.17E-01	1.06E-01	1.06E-01	2.52E-11	8.76E-08	1.88E-01	6.05E-09	1.73E-01	1.88E-01
106Ru	1.19E-01	1.08E-01	1.08E-01	2.59E-11	8.86E-08	1.94E-01	6.22E-09	1.79E-01	1.94E-01
106Rh	1.19E-03	1.08E-03	1.08E-03	8.64E-14	8.04E-10	1.94E-03	4.84E-11	1.79E-03	1.94E-03
110Ag	1.50E-05	1.36E-05	1.36E-05	1.20E-15	1.02E-11	2.42E-05	6.22E-13	2.23E-05	2.42E-05
123Sn	1.98E-05	1.79E-05	1.79E-05	1.46E-15	1.32E-11	3.19E-05	8.29E-13	2.93E-05	3.19E-05
125Sb	2.57E-05	2.33E-05	2.33E-05	1.83E-15	1.73E-11	4.15E-05	1.05E-12	3.81E-05	4.15E-05
127Te	3.76E-05	3.41E-05	3.41E-05	2.79E-15	2.65E-11	6.38E-05	1.59E-12	5.86E-05	6.38E-05
129Te	2.37E-05	2.15E-05	2.15E-05	1.76E-15	1.63E-11	3.83E-05	9.85E-13	3.51E-05	3.83E-05
134Cs	2.57E-05	2.33E-05	2.33E-05	1.86E-14	1.73E-11	4.15E-05	1.05E-12	3.81E-05	4.15E-05
137Cs	1.96E-04	1.78E-04	1.78E-04	1.43E-14	1.32E-10	3.19E-04	8.12E-12	2.93E-04	3.19E-04
141Ce	7.51E-04	6.82E-04	6.82E-04	5.65E-14	5.19E-10	1.24E-03	3.11E-11	1.14E-03	1.24E-03
144Ce	3.56E-03	3.23E-03	3.23E-03	2.56E-13	2.44E-09	5.74E-03	1.47E-10	5.27E-03	5.74E-03
144Pr	3.56E-03	3.23E-03	3.23E-03	2.56E-13	2.44E-09	5.74E-03	1.47E-10	5.27E-03	5.74E-03
147Pm	6.32E-04	5.74E-04	5.74E-04	3.99E-14	4.28E-10	1.02E-03	2.59E-11	9.37E-04	1.02E-03
148Pm	7.31E-06	6.64E-06	6.64E-06	5.31E-16	5.50E-12	1.21E-05	3.28E-13	1.11E-05	1.21E-05
155Eu	1.11E-05	1.00E-05	1.00E-05	7.97E-16	7.53E-12	1.82E-05	0.00E+00	1.67E-05	1.82E-05
234D	2.96E-11	2.69E-11	2.69E-11	2.13E-16	2.04E-18	4.46E-15	1.04E-19	4.10E-15	4.46E-15
235U	4.74E-10	4.30E-10	4.30E-10	3.32E-15	3.16E-17	7.33E-14	1.73E-18	6.73E-14	7.33E-14
236U	4.15E-10	3.77E-10	3.77E-10	3.32E-15	3.16E-17	7.33E-14	1.73E-18	6.73E-14	7.33E-14
238U	4.35E-08	3.95E-08	3.95E-08	3.16E-13	2.95E-15	7.33E-12	1.73E-16	6.73E-12	7.33E-12
237Np	6.13E-11	5.56E-11	5.56E-11	4.65E-21	4.28E-17	9.89E-11	2.59E-18	9.89E-11	9.89E-11
238Pu	2.96E-08	2.69E-08	2.69E-08	6.31E-17	2.04E-10	1.43E-10	3.46E-15	1.32E-10	1.43E-10
239Pu	8.89E-06	8.07E-06	8.07E-06	1.93E-14	6.01E-08	4.46E-08	1.11E-12	4.10E-08	4.46E-08
240Pu	1.98E-06	1.79E-06	1.79E-06	4.32E-15	1.32E-08	9.89E-09	2.59E-13	9.89E-09	9.89E-09
241Pu	1.14E-04	1.22E-04	1.22E-04	2.92E-13	9.16E-07	6.70E-07	1.68E-11	6.15E-07	6.70E-07
242Pu	1.96E-10	1.78E-10	1.78E-10	4.32E-19	1.32E-12	9.57E-13	2.42E-17	8.78E-13	9.57E-13
Total	2.53E-01	2.29E-01	2.29E-01	5.39E-11	1.19E-06	4.11E-01	1.39E-08	3.77E-01	4.11E-01
Frequency, /hr	2.26E-08	2.26E-08	2.26E-08	2.26E-08	2.26E-08	2.26E-08	2.26E-08	2.26E-08	2.26E-08

TABLE C 2. Curves of Specific Nuclides Released from 200-ft Stack due to Externally Induced Failures (F-Canyon)

Nuclide	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Fu Cycle	MAN Evaporator	LAN Evaporator	Ion Exchange	High Heat Waste
89Sr	1.51E-02	1.37E-02	1.37E-02	1.07E-12	9.98E-09	2.44E-02	6.26E-10	2.24E-02	2.44E-02
90Sr	1.43E-03	1.30E-03	1.30E-03	1.04E-13	9.78E-10	2.34E-03	5.93E-11	2.15E-03	2.34E-03
90Y	1.43E-03	1.30E-03	1.30E-03	1.04E-13	9.78E-10	2.34E-03	5.93E-11	2.15E-03	2.34E-03
91Y	2.32E-02	2.11E-02	2.11E-02	1.69E-12	1.60E-08	3.75E-02	9.48E-10	3.44E-02	3.75E-02
95Zr	2.32E-02	2.11E-02	2.11E-02	8.46E-12	1.60E-07	3.75E-02	9.14E-09	3.44E-02	3.75E-02
95Nb	6.20E-03	5.62E-03	5.62E-03	2.25E-12	9.98E-09	1.00E-02	6.94E-10	9.18E-03	1.00E-02
103Ru	1.14E+00	1.04E+00	1.04E+00	2.47E-10	8.58E-07	1.84E+00	5.93E-08	1.69E+00	1.84E+00
106Ru	1.16E+00	1.06E+00	1.06E+00	2.54E-10	8.68E-07	1.91E+00	6.09E-08	1.75E+00	1.91E+00
106Rh	1.16E-02	1.06E-02	1.06E-02	8.46E-13	7.88E-09	1.91E-02	4.74E-10	1.75E-02	1.91E-02
110Ag	1.47E-04	1.34E-04	1.34E-04	1.17E-14	9.98E-11	2.38E-04	6.09E-12	2.18E-04	2.38E-04
123Sn	1.94E-04	1.76E-04	1.76E-04	1.43E-14	1.30E-10	3.13E-04	6.13E-12	2.87E-04	3.13E-04
125Sb	2.52E-04	2.29E-04	2.29E-04	1.79E-14	1.70E-10	4.06E-04	1.03E-11	3.73E-04	4.06E-04
127Te	3.68E-04	3.34E-04	3.34E-04	2.73E-14	2.59E-10	6.25E-04	1.56E-11	5.74E-04	6.25E-04
129Te	2.32E-04	2.11E-04	2.11E-04	1.73E-14	1.60E-10	3.75E-04	9.65E-12	3.44E-04	3.75E-04
134Cs	2.52E-04	2.29E-04	2.29E-04	1.82E-13	1.70E-10	4.06E-04	1.03E-11	3.73E-04	4.06E-04
137Cs	1.92E-03	1.74E-03	1.74E-03	1.40E-13	1.30E-09	3.13E-03	7.96E-11	2.87E-03	3.13E-03
141Ce	7.36E-03	6.68E-03	6.68E-03	5.53E-13	5.09E-09	1.22E-02	3.05E-10	1.12E-02	1.22E-02
144Ce	3.49E-02	3.16E-02	3.16E-02	2.51E-12	2.40E-08	5.63E-02	1.44E-09	5.17E-02	5.63E-02
144Pr	3.49E-02	3.16E-02	3.16E-02	2.51E-12	2.40E-08	5.63E-02	1.44E-09	5.17E-02	5.63E-02
147Pm	6.20E-03	5.62E-03	5.62E-03	3.91E-13	4.19E-09	1.00E-02	2.54E-10	9.18E-03	1.00E-02
148Pm	7.17E-05	6.50E-05	6.50E-05	5.21E-15	5.39E-11	1.19E-04	3.22E-12	1.09E-04	1.19E-04
155Eu	1.09E-04	9.84E-05	9.84E-05	7.81E-15	7.38E-11	1.78E-04	0.00E+00	1.64E-04	1.78E-04
234U	2.91E-10	2.64E-10	2.64E-10	2.08E-15	2.00E-17	4.38E-14	1.02E-18	4.02E-14	4.38E-14
235U	4.65E-09	4.22E-09	4.22E-09	3.26E-14	3.09E-16	7.19E-13	1.69E-17	6.60E-13	7.19E-13
236U	4.07E-09	3.69E-09	3.69E-09	3.26E-14	3.09E-16	7.19E-13	1.69E-17	6.60E-13	7.19E-13
238U	4.26E-07	3.87E-07	3.87E-07	3.09E-12	2.89E-14	7.19E-11	1.69E-15	6.60E-11	7.19E-11
237Np	6.00E-10	5.45E-10	5.45E-10	4.56E-20	4.19E-16	9.69E-10	2.54E-17	8.90E-10	9.69E-10
238Pu	2.91E-07	2.64E-07	2.64E-07	6.18E-16	2.00E-09	1.41E-09	3.39E-14	1.29E-09	1.41E-09
239Pu	8.72E-05	7.91E-05	7.91E-05	1.89E-13	5.89E-07	4.38E-07	1.08E-11	4.02E-07	4.38E-07
240Pu	1.94E-05	1.76E-05	1.76E-05	4.23E-14	1.30E-07	9.69E-08	2.54E-12	8.90E-08	9.69E-08
241Pu	1.32E-03	1.20E-03	1.20E-03	2.86E-12	8.98E-06	6.56E-06	1.64E-10	6.03E-06	6.56E-06
242Pu	1.92E-09	1.74E-09	1.74E-09	4.23E-18	1.30E-11	9.38E-12	2.37E-16	8.61E-12	9.38E-12
Total	2.48E+00	2.25E+00	2.25E+00	5.28E-10	1.17E-05	4.02E+00	1.36E-07	3.70E+00	4.02E+00
Frequency, /hr	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06

DELETED VERSION

C-4

TABLE C-3. Curies of Specific Nuclides Released from 200-ft Stack due to Fire (F-Canyon)

Nuclide	First Cycle	2nd U Cycle	2nd Pu Cycle	Ion Exchange
89Sr	3.62E-02	1.22E-07	3.48E-06	7.10E-02
90Sr	4.71E-03	1.18E-08	3.41E-07	6.82E-03
90Y	4.71E-03	1.18E-08	3.41E-07	6.82E-03
91Y	5.57E-02	1.92E-07	5.57E-06	1.09E-01
95Zr	5.57E-02	9.59E-07	5.57E-05	1.09E-01
95Nb	6.89E-03	2.55E-07	3.48E-06	2.91E-02
103Ru	2.74E-02	2.80E-07	2.99E-06	5.37E-02
106Ru	2.78E-02	2.88E-07	3.03E-06	5.55E-02
106Rh	2.78E-02	9.59E-08	2.75E-06	5.55E-02
110Ag	3.53E-04	1.33E-09	3.48E-08	5.92E-03
123Sn	1.64E-04	1.62E-09	4.52E-08	9.10E-04
125Sb	6.03E-04	2.03E-09	5.92E-08	1.18E-03
127Te	8.82E-04	3.10E-09	9.05E-08	1.82E-03
129Te	5.57E-04	1.96E-09	5.57E-08	1.09E-03
134Cs	6.03E-04	2.07E-09	5.92E-08	1.18E-03
137Cs	1.59E-03	1.59E-08	4.52E-07	9.10E-03
141Ce	1.76E-02	6.27E-08	1.77E-06	3.55E-02
144Ce	8.35E-02	2.84E-07	8.35E-06	1.64E-01
144Pr	8.35E-02	2.84E-07	8.35E-06	1.64E-01
147Pm	1.48E-02	4.43E-08	1.46E-06	2.91E-02
148Pm	1.72E-04	5.90E-10	1.88E-08	3.46E-04
155Eu	2.60E-04	8.89E-10	2.58E-08	5.18E-04
234U	6.96E-10	2.36E-10	6.96E-15	1.27E-13
235U	1.11E-08	3.69E-09	1.08E-13	2.09E-12
236U	9.74E-09	3.69E-09	1.08E-13	2.09E-12
238U	1.02E-06	3.51E-07	1.01E-11	2.09E-10
237Np	1.44E-09	5.17E-15	1.46E-13	2.82E-09
238Pu	6.96E-07	7.01E-11	6.96E-07	4.09E-09
239Pu	2.09E-04	2.14E-06	2.05E-04	1.27E-06
240Pu	4.64E-05	4.80E-09	4.52E-05	2.82E-07
241Pu	3.16E-03	3.25E-07	3.13E-03	1.91E-05
242Pu	4.59E-09	4.80E-13	4.52E-09	2.73E-11
Total	4.58E-01	3.63E-06	3.48E-03	9.11E-01
Frequency, /hr	7.00E-06	7.00E-08	7.00E-08	9.00E-07

TABLE C-4. Curves of Specific Nuclides Released From 200-ft Stack due to Uncontrolled Reactions (F-Canyon)

Nuclide	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Pu Cycle	MAN Evaporator	LAN Evaporator	Ion Exchange	High Heat Waste
89Sr	2.67E-03	2.53E-03	2.51E-03	4.62E-09	1.32E-07	5.68E-03	2.49E-07	7.58E-03	5.68E-03
90Sr	2.54E-04	2.40E-04	2.37E-04	4.48E-10	1.29E-08	5.45E-04	2.36E-08	7.02E-04	5.45E-04
90Y	2.54E-04	2.40E-04	2.37E-04	4.48E-10	1.29E-08	5.45E-04	2.36E-08	7.02E-04	5.45E-04
91Y	4.11E-03	3.90E-03	3.05E-03	7.28E-09	2.11E-07	8.74E-03	3.77E-07	1.12E-02	8.74E-03
95Zr	4.11E-03	3.90E-03	3.05E-03	3.64E-08	2.11E-06	8.74E-03	3.64E-06	1.12E-02	8.74E-03
95Nb	1.10E-03	1.04E-03	1.03E-03	9.66E-09	1.32E-07	2.33E-03	2.76E-07	2.99E-03	2.33E-03
103Ru	9.95E-02	8.94E-02	8.93E-02	1.06E-08	1.13E-07	2.63E-01	2.36E-07	3.49E-01	2.63E-01
106Ru	1.01E-01	9.09E-02	9.08E-02	1.09E-08	1.15E-07	2.72E-01	2.43E-07	3.60E-01	2.72E-01
106Rh	2.06E-03	1.96E-03	9.08E-02	3.64E-09	1.04E-07	4.44E-03	1.89E-07	5.70E-03	4.44E-03
110Ag	2.61E-05	2.48E-05	2.44E-05	5.04E-11	1.32E-09	5.53E-05	2.43E-09	7.11E-05	5.53E-05
123Sn	3.43E-05	3.26E-05	3.21E-05	6.16E-11	1.72E-09	7.28E-05	3.23E-09	9.36E-05	7.28E-05
125Sb	4.46E-05	4.24E-05	4.18E-05	7.70E-11	2.24E-09	9.47E-05	4.11E-09	1.22E-04	9.47E-05
127Te	1.10E-05	6.20E-05	6.11E-05	1.17E-10	3.43E-09	1.46E-04	6.20E-09	1.87E-04	1.46E-04
129Te	4.11E-05	3.91E-05	3.85E-05	7.42E-11	2.11E-09	8.74E-05	3.84E-09	1.12E-04	8.74E-05
134Cs	4.46E-05	4.24E-05	4.18E-05	7.84E-11	2.24E-09	9.44E-05	4.11E-09	1.22E-04	9.44E-05
137Cs	3.39E-04	3.22E-04	3.18E-04	6.02E-10	1.72E-08	7.28E-04	3.17E-08	9.36E-04	7.28E-04
141Ce	1.30E-03	1.24E-03	1.22E-03	2.38E-09	6.73E-08	2.84E-03	1.21E-07	3.65E-03	2.84E-03
144Ce	6.17E-03	5.86E-03	5.79E-03	1.08E-08	3.17E-07	1.31E-02	5.73E-07	1.68E-02	1.31E-02
144Pr	6.17E-03	5.86E-03	5.79E-03	1.08E-08	3.17E-07	1.31E-02	5.73E-07	1.68E-02	1.31E-02
147Pm	1.10E-03	1.04E-03	1.03E-03	1.68E-09	5.54E-08	2.33E-03	1.01E-07	2.99E-03	2.33E-03
148Pm	1.27E-05	1.20E-05	1.19E-05	2.24E-11	7.13E-10	2.77E-05	1.28E-09	3.55E-05	2.77E-05
155Eu	1.92E-05	1.82E-05	1.80E-05	3.36E-11	9.77E-10	4.15E-05	0.00E+00	5.33E-05	4.15E-05
234U	5.14E-11	4.89E-11	4.82E-11	8.96E-12	2.64E-16	1.02E-14	4.04E-16	1.31E-14	1.02E-14
235U	8.22E-10	7.81E-10	7.72E-10	1.40E-10	4.09E-15	1.67E-13	6.74E-15	2.15E-13	1.67E-13
236U	7.20E-10	6.84E-10	6.75E-10	1.40E-10	4.09E-15	1.67E-13	6.74E-15	2.12E-13	1.67E-13
238U	7.54E-08	7.16E-08	7.07E-08	1.33E-08	3.83E-13	1.67E-11	6.74E-13	2.12E-11	1.67E-11
237Np	1.06E-10	1.01E-10	9.97E-11	1.96E-16	5.54E-15	2.26E-10	1.01E-14	2.90E-10	2.26E-10
238Pu	5.14E-08	4.88E-08	4.82E-08	2.66E-12	2.64E-08	3.28E-10	1.35E-11	4.21E-10	3.28E-10
239Pu	1.54E-05	1.47E-05	1.45E-05	8.12E-10	7.79E-06	1.02E-07	4.31E-09	1.31E-07	1.02E-07
240Pu	3.43E-06	3.26E-06	3.21E-06	1.82E-10	1.72E-06	2.26E-08	1.01E-09	2.90E-08	2.26E-08
241Pu	2.33E-04	3.22E-04	2.18E-04	1.23E-08	1.19E-04	1.53E-06	6.54E-08	1.96E-06	1.53E-06
242Pu	3.39E-10	3.22E-10	3.16E-10	1.82E-14	1.72E-10	2.18E-12	9.44E-14	0.00E+00	2.18E-12
241Am	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.80E-12	0.00E+00
Total	2.31E-01	2.09E-01	2.96E-01	1.38E-07	1.32E-04	5.99E-01	6.75E-06	7.91E-01	5.99E-01
Frequency, /hr	3.00E-05	5.00E-06	1.33E-07	3.33E-07	3.33E-07	7.20E-05	1.80E-05	6.00E-05	5.00E-06

TABLE C-5. Curies of Specific Nuclides Released
from 200-ft Stack due to Criticality (F-Canyon)

Nuclide	Volatile Fission Products
83Br	9.20E+00
84Br	7.00E+01
85Br	2.60E+02
83mKr	5.10E+00
85mKr	1.20E+01
87Kr	7.60E+01
88Kr	2.20E+02
89Kr	2.00E+03
90Kr	1.50E+04
131I	1.70E+00
132I	1.40E+01
133I	3.00E+01
134I	3.80E+02
135I	1.00E+02
136I	2.30E+04
133Xe	1.80E+00
135mXe	1.10E+02
135Xe	3.20E+01
137Xe	4.90E+03
138Xe	1.40E+03
Total	4.76E+04
Frequency, /hr	3.62E-07

TABLE C-6. Curves of Specific Nuclides Released at Ground Level due to Transfer Error to 211-F (F-Canyon)

Nuclide	Decon- tamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Pu Cycle	HAN Evaporator	LAN Evaporator	Ion Exchange	High Heat Waste
89Sr	6.21E-06	3.26E-01	2.93E-01	2.93E-01	7.92E-12	2.14E-07	8.65E-01	2.22E-08	7.96E-02	8.65E-01
90Sr	6.31E-07	3.10E-02	2.78E-02	2.78E-02	7.68E-13	2.09E-08	8.32E-02	2.10E-09	7.66E-03	8.32E-02
90Y	1.08E-05	3.10E-02	2.78E-02	2.78E-02	7.68E-13	2.09E-08	8.32E-02	2.10E-09	7.66E-03	8.32E-02
91Y	0.00E+00	5.02E-01	4.51E-01	4.51E-01	1.25E-11	3.42E-07	1.33E+00	3.37E-08	1.23E-01	1.33E+00
95Zr	1.68E-05	5.02E-01	4.51E-01	4.51E-01	6.24E-11	3.42E-06	1.33E+00	3.25E-07	1.23E-01	1.33E+00
95Nb	3.06E-05	1.34E-01	1.20E-01	1.20E-01	1.66E-11	2.14E-07	3.55E-01	2.46E-08	3.27E-02	3.55E-01
103Ru	5.62E-06	2.47E+01	2.22E+01	2.22E+01	1.82E-09	1.84E-05	6.55E-01	2.10E-06	6.02E+00	6.55E-01
106Ru	7.19E-06	2.51E+01	2.26E+01	2.26E+01	1.87E-09	1.86E-05	6.77E-01	2.16E-06	6.23E+00	6.77E-01
106Rh	0.00E+00	2.51E-01	2.26E-01	2.26E-01	6.24E-12	1.69E-07	6.77E-01	1.68E-08	6.23E-02	6.77E-01
110Ag	0.00E+00	3.18E-03	2.86E-03	2.86E-03	8.64E-14	2.14E-09	8.43E-03	2.16E-10	7.76E-04	8.43E-03
123Sn	0.00E+00	4.18E-03	3.76E-03	3.76E-03	1.06E-13	2.78E-09	1.11E-02	2.88E-10	1.02E-03	1.11E-02
125Sb	0.00E+00	5.44E-03	4.89E-03	4.89E-03	1.32E-13	3.63E-09	1.44E-02	3.67E-10	1.33E-03	1.44E-02
127Te	0.00E+00	7.95E-03	7.14E-03	7.14E-03	2.02E-13	5.55E-09	2.22E-02	5.53E-10	2.04E-03	2.22E-02
129Te	0.00E+00	5.02E-03	4.51E-03	4.51E-03	1.27E-13	3.42E-10	1.33E-02	3.43E-10	1.23E-03	1.33E-02
134Cs	3.55E-07	5.44E-03	4.89E-03	4.89E-03	1.34E-13	3.63E-09	1.44E-02	3.67E-10	1.33E-03	1.44E-02
137Cs	9.76E-07	4.14E-02	3.72E-02	3.72E-02	1.03E-12	2.78E-08	1.11E-01	2.82E-09	1.02E-02	1.11E-01
141Ce	0.00E+00	1.59E-01	1.43E-01	1.43E-01	4.08E-12	1.09E-07	4.33E-01	1.08E-08	3.98E-02	4.33E-01
144Ce	1.66E-05	7.53E-01	6.77E-01	6.77E-01	1.85E-11	5.12E-07	2.00E+00	5.11E-08	1.84E-01	2.00E+00
144La	5.02E-05	5.02E-01	4.51E-01	4.51E-01	1.25E-11	3.42E-07	1.33E+00	3.37E-08	1.23E-01	1.33E+00
147Pm	2.78E-05	1.34E-01	1.20E-01	1.20E-01	2.88E-12	8.97E-08	3.55E-01	3.01E-09	3.27E-02	3.55E-01
148Pm	0.00E+00	1.55E-03	1.39E-03	1.39E-03	3.84E-14	1.15E-09	4.22E-03	1.14E-10	3.58E-04	4.22E-03
155Eu	0.00E+00	2.34E-03	2.11E-03	2.11E-03	5.76E-14	1.58E-09	6.32E-03	0.00E+00	5.82E-04	6.32E-03
234U	0.00E+00	6.27E-09	5.64E-09	5.64E-09	1.54E-14	4.27E-16	1.55E-12	3.61E-17	1.43E-13	1.55E-12
235U	2.96E-16	1.00E-07	9.03E-08	9.03E-08	2.40E-13	6.62E-15	2.55E-11	6.01E-16	2.35E-12	2.55E-11
236U	1.97E-16	8.78E-08	7.90E-08	7.90E-08	2.40E-13	6.62E-15	2.55E-11	6.01E-16	2.35E-12	2.55E-11
238U	2.86E-14	9.20E-06	8.27E-06	8.27E-06	2.28E-11	6.19E-13	2.55E-09	6.01E-14	2.35E-10	2.55E-09
237Np	0.00E+00	1.30E-08	1.17E-08	1.17E-08	3.36E-19	8.97E-15	3.44E-08	9.01E-16	3.17E-09	3.44E-08
238Pu	1.38E-11	6.27E-06	5.64E-06	5.64E-06	4.56E-15	4.27E-08	4.99E-08	1.20E-12	4.60E-09	4.99E-08
239Pu	1.48E-10	1.88E-03	1.69E-03	1.69E-03	1.39E-12	1.26E-05	1.55E-05	3.85E-10	1.43E-06	1.55E-05
240Pu	2.66E-11	4.18E-04	3.76E-04	3.76E-04	3.12E-13	2.78E-06	3.44E-06	9.01E-11	3.17E-07	3.44E-06
241Pu	1.28E-09	2.84E-02	2.56E-02	2.56E-02	2.11E-11	1.92E-04	2.33E-04	5.83E-09	2.14E-05	2.33E-04
242Pu	0.00E+00	4.14E-08	3.72E-08	3.72E-08	3.12E-17	2.78E-10	3.33E-10	8.41E-15	3.06E-11	3.33E-10
Total	9.86E-05	5.35E+01	4.81E+01	4.81E+01	3.89E-09	2.50E-04	1.43E+02	4.81E-06	1.31E+01	1.43E+02
Frequency, /hr	3.00E-09	1.20E-08	2.00E-09	3.50E-09	2.10E-09	8.40E-09	1.00E-08	1.00E-08	8.00E-09	2.00E-09

TABLE C-7. Curies of Specific Nuclides Released from 200-ft Stack due to Transfer Error to Sump (F-Canyon)

Nuclide	Decon- tamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Pu Cycle	HAN Evaporator	LAN Evaporator	Ion Exchange	High Heat Write ..
89Sr	3.04E-08	1.56E-03	1.44E-03	1.44E-03	3.88E-14	1.05E-09	4.24E-03	1.09E-10	3.90E-04	4.24E-03
90Sr	3.08E-09	1.52E-04	1.36E-04	1.36E-04	3.76E-15	1.03E-10	4.08E-04	1.03E-11	3.75E-05	4.08E-04
90Y	5.31E-08	1.52E-04	1.36E-04	1.36E-04	3.76E-15	1.03E-10	4.08E-04	1.03E-11	3.75E-05	4.08E-04
91Y	0.00E+00	2.46E-03	2.21E-03	2.21E-03	6.11E-14	1.65E-09	6.52E-03	1.65E-10	6.00E-04	6.52E-03
95Zr	8.21E-08	2.46E-03	2.21E-03	2.21E-03	3.06E-13	1.65E-08	6.52E-03	1.59E-09	6.00E-04	6.52E-03
95Nb	1.50E-07	6.56E-04	5.90E-04	5.90E-04	8.11E-14	1.05E-09	1.74E-03	1.21E-10	1.60E-04	1.74E-03
103Ru	2.75E-08	1.21E-01	1.09E-01	1.09E-01	8.93E-12	9.00E-08	3.21E-01	1.03E-04	2.95E-02	3.21E-01
106Ru	3.52E-08	1.23E-01	1.11E-01	1.11E-01	9.17E-12	9.10E-08	3.32E-01	1.06E-04	3.05E-02	3.32E-01
106Rh	0.00E+00	1.23E-03	1.11E-03	1.11E-03	3.06E-14	8.26E-10	3.32E-03	8.24E-11	3.05E-04	3.32E-03
110Ag	0.00E+00	1.56E-05	1.40E-05	1.40E-05	4.23E-16	1.05E-11	4.13E-05	1.06E-12	3.80E-06	4.13E-05
123Sn	0.00E+00	2.05E-05	1.84E-05	1.84E-05	5.17E-16	1.36E-11	5.44E-05	1.41E-12	5.00E-06	5.44E-05
125Sb	0.00E+00	2.66E-05	2.40E-05	2.40E-05	6.47E-16	1.78E-11	7.07E-05	1.80E-12	6.50E-06	7.07E-05
127Te	0.00E+00	3.89E-05	3.50E-05	3.50E-05	9.87E-16	2.72E-11	1.09E-04	2.71E-12	1.00E-05	1.09E-04
129Te	0.00E+00	2.46E-05	2.21E-05	2.21E-05	6.23E-16	1.67E-11	6.52E-05	1.60E-12	6.00E-06	6.52E-05
134Cs	1.74E-09	2.66E-05	2.40E-05	2.40E-05	6.58E-16	1.78E-11	7.07E-05	1.80E-12	6.50E-06	7.07E-05
137Cs	4.78E-09	2.03E-04	1.82E-04	1.82E-04	5.05E-15	1.36E-10	5.44E-04	1.38E-11	5.00E-05	5.44E-04
141Ce	0.00E+00	7.78E-04	7.00E-04	7.00E-04	2.00E-14	5.34E-10	2.12E-03	5.30E-11	1.95E-04	2.12E-03
144Ce	8.21E-08	3.69E-03	3.32E-03	3.32E-03	9.05E-14	2.51E-09	9.78E-03	2.50E-10	9.01E-04	9.78E-03
144Pr	0.00E+00	3.69E-03	3.32E-03	3.32E-03	9.05E-14	2.51E-09	9.78E-03	2.50E-10	9.01E-04	9.78E-03
147Pm	1.35E-08	6.56E-04	5.90E-04	5.90E-04	1.41E-14	4.39E-10	1.74E-03	4.42E-11	1.60E-04	1.74E-03
148Pm	0.00E+00	7.58E-06	6.82E-06	6.82E-06	1.88E-16	5.65E-12	2.07E-05	5.59E-13	1.90E-06	2.07E-05
155Eu	0.00E+00	1.15E-05	1.03E-05	1.03E-05	2.82E-16	7.74E-12	3.10E-05	1.77E-19	2.85E-06	3.10E-05
234U	0.00E+00	3.07E-11	2.76E-11	2.76E-11	7.52E-17	2.09E-18	7.61E-15	0.00E+00	7.01E-16	7.61E-15
235U	1.45E-18	4.92E-10	4.42E-10	4.42E-10	1.18E-15	3.24E-17	1.25E-13	2.94E-18	1.15E-14	1.25E-13
236U	4.65E-19	4.30E-10	3.87E-10	3.87E-10	1.18E-15	3.24E-17	1.25E-13	2.94E-18	1.15E-14	1.25E-13
238U	1.40E-16	4.51E-08	4.05E-08	4.05E-08	1.12E-13	3.03E-15	1.25E-11	2.94E-16	1.15E-12	1.25E-11
237Np	0.00E+00	6.55E-11	5.71E-11	5.71E-11	1.65E-21	4.39E-17	1.69E-10	4.42E-18	1.55E-11	1.69E-10
238Pu	6.76E-14	3.07E-08	2.76E-08	2.76E-08	2.23E-17	2.09E-10	2.45E-10	5.89E-15	2.55E-11	2.45E-10
239Pu	7.24E-13	9.22E-06	8.29E-06	8.29E-06	6.82E-15	6.17E-08	7.61E-08	1.88E-12	7.01E-09	7.61E-08
240Pu	1.30E-13	2.05E-06	1.84E-06	1.84E-06	1.53E-15	1.36E-08	1.69E-08	4.42E-13	1.55E-09	1.69E-08
241Pu	6.28E-12	1.39E-04	1.25E-04	1.25E-04	1.03E-13	9.42E-07	1.14E-06	2.86E-11	1.05E-07	1.14E-06
242Pu	0.00E+00	2.03E-10	1.82E-10	1.82E-10	1.53E-19	1.36E-12	1.63E-12	4.12E-17	1.50E-13	1.63E-12
Total	4.83E-07	2.62E-01	2.36E-01	2.36E-01	1.91E-11	1.23E-06	7.60E-01	2.36E-08	6.44E-02	7.00E-01
Frequency, hr	3.80E-07	4.50E-06	7.50E-07	1.30E-06	7.90E-07	3.10E-06	3.80E-06	3.60E-06	3.00E-06	7.50E-07

TABLE C-8. Curries of Specific Nuclides Released from 200-ft Stack due to Overflow to Sumps (F-Canyon)

Nuclide	Decon- tamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Pu Cycle	MAN Evaporator	LAN Evaporator	Ion Exchange	High Heat Raste
89Sr	1.78E-08	9.39E-04	8.41E-04	8.41E-04	2.27E-14	6.12E-10	2.49E-03	6.40E-11	1.33E-03	2.49E-03
90Sr	1.81E-09	8.90E-05	7.98E-05	7.98E-05	2.20E-15	6.00E-11	2.39E-04	6.05E-12	1.28E-04	2.39E-04
90Y	0.00E+00	8.90E-05	7.98E-05	7.98E-05	2.20E-15	6.00E-11	2.39E-04	6.05E-12	1.28E-04	2.39E-04
91Y	3.12E-08	1.44E-03	1.29E-03	1.29E-03	3.58E-14	9.79E-10	3.83E-03	9.68E-11	2.04E-03	3.83E-03
95Zr	4.81E-08	1.44E-03	1.29E-03	1.29E-03	1.80E-13	9.79E-09	3.83E-03	9.33E-10	2.04E-03	3.83E-03
95Nb	8.78E-08	3.85E-04	3.45E-04	3.45E-04	4.74E-14	6.12E-10	1.02E-03	7.09E-11	5.44E-04	1.02E-03
103Ru	1.61E-08	7.10E-02	6.36E-02	6.36E-02	5.23E-12	5.26E-08	1.88E-01	6.05E-09	1.00E-01	1.88E-01
106Ru	2.07E-08	7.22E-02	6.47E-02	6.47E-02	5.36E-12	5.32E-08	1.95E-01	6.22E-09	1.04E-01	1.95E-01
106Rh	0.00E+00	7.22E-04	6.47E-04	6.47E-04	1.79E-14	4.83E-10	1.95E-03	4.04E-11	1.04E-03	1.95E-03
110Ag	0.00E+00	9.15E-06	8.19E-06	8.19E-06	2.48E-16	6.12E-12	2.43E-05	6.22E-13	1.29E-05	2.43E-05
123Sn	0.00E+00	1.20E-05	1.08E-05	1.08E-05	3.03E-16	7.95E-12	3.19E-05	8.30E-13	1.70E-05	3.19E-05
125Sb	0.00E+00	1.56E-05	1.40E-05	1.40E-05	3.78E-16	1.04E-11	4.15E-05	1.05E-12	2.21E-05	4.15E-05
127Te	0.00E+00	2.29E-05	2.05E-05	2.05E-05	5.78E-16	1.59E-11	6.38E-05	1.59E-12	3.40E-05	6.38E-05
129Te	0.00E+00	1.44E-05	1.29E-05	1.29E-05	3.64E-16	9.79E-12	3.83E-05	9.85E-13	2.04E-05	3.83E-05
134Cs	1.02E-09	1.56E-05	1.40E-05	1.40E-05	3.85E-16	1.04E-11	4.15E-05	1.05E-12	2.21E-05	4.15E-05
137Cs	2.80E-09	1.19E-04	1.07E-04	1.07E-04	2.96E-15	7.95E-11	3.19E-04	8.12E-12	1.70E-04	3.19E-04
141Ce	0.00E+00	4.57E-04	4.10E-04	4.10E-04	1.17E-14	3.12E-10	1.25E-03	3.11E-11	6.63E-04	1.25E-03
144Ce	4.81E-08	2.17E-03	1.94E-03	1.94E-03	5.29E-14	1.47E-09	5.74E-03	1.47E-10	3.06E-03	5.74E-03
144Pr	0.00E+00	2.17E-03	1.94E-03	1.94E-03	5.29E-14	1.47E-09	5.74E-03	1.47E-10	3.06E-03	5.74E-03
147Pm	7.93E-09	3.85E-04	3.45E-04	3.45E-04	8.25E-15	2.57E-10	1.02E-03	2.59E-11	5.44E-04	1.02E-03
148Pm	0.00E+00	4.45E-06	3.99E-06	3.99E-06	1.10E-16	3.30E-12	1.21E-05	3.28E-13	6.46E-06	1.21E-05
155Eu	0.00E+00	6.74E-06	6.04E-06	6.04E-06	1.65E-16	4.53E-12	1.82E-05	1.04E-19	9.69E-06	1.82E-05
234B	0.00E+00	1.81E-11	1.62E-11	1.62E-11	4.40E-17	1.22E-18	4.67E-15	0.00E+00	2.38E-15	4.67E-15
235U	8.50E-19	2.89E-10	2.59E-10	2.59E-10	6.88E-16	1.90E-17	7.34E-14	1.73E-18	3.91E-14	7.34E-14
236U	5.66E-19	2.53E-10	2.26E-10	2.26E-10	6.88E-16	1.90E-17	7.34E-14	1.73E-18	3.91E-14	7.34E-14
238U	8.21E-17	2.65E-08	2.37E-08	2.37E-08	6.53E-14	1.77E-15	7.34E-12	1.73E-16	3.91E-12	7.34E-12
237Np	0.00E+00	3.73E-11	3.34E-11	3.34E-11	9.63E-22	2.57E-17	9.89E-11	2.59E-18	5.27E-11	9.89E-11
238Pu	3.96E-14	1.81E-08	1.62E-08	1.62E-08	1.31E-17	1.22E-10	1.44E-10	3.46E-15	7.65E-11	1.44E-10
239Pu	4.25E-13	5.42E-06	4.85E-06	4.85E-06	3.99E-15	3.61E-08	4.47E-08	1.11E-12	2.38E-08	4.47E-08
240Pu	7.65E-14	1.20E-06	1.08E-06	1.08E-06	8.94E-16	7.95E-09	9.89E-09	2.59E-13	5.27E-09	9.89E-09
241Pu	3.68E-12	8.18E-05	7.33E-05	7.33E-05	6.05E-14	5.51E-07	6.70E-07	1.68E-11	3.57E-07	6.70E-07
242Pu	0.00E+00	1.19E-10	1.07E-10	1.07E-10	8.94E-20	7.95E-13	9.57E-13	2.42E-17	5.10E-13	9.57E-13
Total	2.83E-07	1.54E-01	1.38E-01	1.38E-01	1.12E-11	7.17E-07	4.11E-01	1.39E-08	2.19E-01	4.11E-01
Frequency, /hr	5.00E-07	5.00E-06	5.00E-06	2.30E-05	6.80E-06	2.60E-05	2.50E-05	2.50E-05	4.00E-05	4.00E-05

TABLE C-9. Curies of Specific Nuclides Released from 200-ft Stack due to Leaks to Sumps (F-Canyon)

Nuclide	Decon- tamination	Dissolver	Head End	First Cycle	2nd W Cycle	2nd Pu Cycle	MAN Evaporator	LAN Evaporator	Ion Exchange	High Heat Waste
89Sr	5.58E-09	1.33E-04	1.18E-04	1.18E-04	3.18E-15	8.58E-11	3.53E-04	9.86E-12	4.63E-04	3.53E-04
90Sr	5.67E-10	1.26E-05	1.12E-05	1.12E-05	3.08E-16	8.40E-12	3.39E-05	8.57E-13	4.45E-05	3.39E-05
90Y	0.00E+00	1.26E-05	1.12E-05	1.12E-05	3.08E-16	8.40E-12	3.39E-05	8.57E-13	4.45E-05	3.39E-05
91Y	9.74E-09	2.05E-04	1.81E-04	1.81E-04	5.01E-15	1.37E-10	5.42E-04	1.37E-11	7.12E-04	5.42E-04
95Zr	1.51E-08	2.05E-04	1.81E-04	1.81E-04	2.51E-14	1.37E-09	5.42E-04	1.32E-10	7.12E-04	5.42E-04
95Nb	2.74E-08	5.45E-05	4.83E-05	4.83E-05	6.65E-15	8.58E-11	1.45E-04	1.00E-11	1.90E-04	1.45E-04
103Ru	5.05E-09	1.01E-02	8.91E-03	8.91E-03	7.32E-13	7.38E-09	2.67E-02	8.57E-10	3.50E-02	2.67E-02
106Ru	6.46E-09	1.02E-02	9.06E-03	9.06E-03	7.52E-13	7.46E-09	2.76E-02	8.81E-10	3.62E-02	2.76E-02
106Rh	0.00E+00	1.02E-04	9.06E-05	9.06E-05	2.51E-15	6.77E-11	2.76E-04	6.86E-12	3.62E-04	2.76E-04
110Ag	0.00E+00	1.30E-06	1.15E-06	1.15E-06	3.47E-17	8.58E-13	3.44E-06	8.81E-14	4.51E-06	3.44E-06
123Sn	0.00E+00	1.70E-06	1.51E-06	1.51E-06	4.24E-17	1.12E-12	4.52E-06	1.18E-13	5.93E-06	4.52E-06
125Sb	0.00E+00	2.22E-06	1.96E-06	1.96E-06	5.30E-17	1.46E-12	5.88E-06	1.49E-13	7.71E-06	5.88E-06
127I	0.00E+00	3.24E-06	2.87E-06	2.87E-06	8.05E-17	2.23E-12	9.04E-06	2.25E-13	1.19E-05	9.04E-06
129Te	0.00E+00	2.05E-06	1.81E-06	1.81E-06	5.11E-17	1.37E-12	5.42E-06	1.10E-13	7.12E-06	5.42E-06
134Cs	3.19E-10	2.22E-06	1.96E-06	1.96E-06	5.40E-17	1.46E-12	5.88E-06	1.49E-13	7.71E-06	5.88E-06
137Cs	8.76E-10	1.69E-05	1.50E-05	1.50E-05	4.14E-16	1.12E-11	4.52E-05	1.15E-12	5.93E-05	4.52E-05
141Ce	0.00E+00	6.48E-05	5.74E-05	5.74E-05	1.64E-15	4.37E-11	1.76E-04	4.41E-12	2.31E-04	1.76E-04
144Ce	1.51E-08	3.07E-04	2.72E-04	2.72E-04	7.42E-15	2.06E-10	8.14E-04	2.08E-11	1.07E-03	8.14E-04
144Pr	0.00E+00	3.07E-04	2.72E-04	2.72E-04	7.42E-15	2.06E-10	8.14E-04	2.08E-11	1.07E-03	8.14E-04
147Pm	2.48E-09	5.45E-05	4.83E-05	4.83E-05	1.16E-15	3.60E-11	1.45E-04	3.67E-12	1.90E-04	1.45E-04
148Pm	0.00E+00	6.31E-07	5.59E-07	5.59E-07	1.54E-17	4.63E-13	1.72E-06	4.65E-14	2.25E-06	1.72E-06
155Eu	0.00E+00	9.54E-07	8.46E-07	8.46E-07	2.31E-17	6.35E-13	2.58E-06	0.00E+00	3.38E-06	2.58E-06
234U	0.00E+00	2.56E-12	2.27E-12	2.27E-12	6.17E-18	1.72E-19	6.33E-16	1.47E-20	8.30E-16	6.33E-16
235U	2.66E-19	4.09E-11	3.63E-11	3.63E-11	9.64E-17	2.66E-18	1.04E-14	2.45E-19	1.36E-14	9.64E-17
236U	1.77E-19	3.58E-11	3.17E-11	3.17E-11	9.64E-17	2.66E-18	1.04E-14	2.45E-19	1.36E-14	9.64E-17
238U	2.57E-17	3.75E-09	3.32E-09	3.32E-09	9.15E-15	2.49E-16	1.04E-12	2.45E-17	1.36E-12	1.04E-12
237Np	0.00E+00	5.28E-12	4.68E-12	4.68E-12	1.35E-22	3.60E-18	1.40E-11	3.67E-19	1.84E-11	1.40E-11
238Pu	1.24E-14	2.56E-09	2.27E-09	2.27E-09	1.83E-18	1.72E-11	2.03E-11	4.90E-16	2.67E-11	2.03E-11
239Pu	1.33E-13	7.67E-07	6.80E-07	6.80E-07	5.59E-16	5.06E-09	6.33E-09	1.57E-13	8.30E-09	6.33E-09
240Pu	2.39E-14	1.70E-07	1.51E-07	1.51E-07	1.25E-16	1.12E-09	1.40E-09	3.67E-14	1.84E-09	1.40E-09
241Pu	1.15E-12	1.16E-05	1.03E-05	1.03E-05	8.48E-15	7.72E-08	9.49E-08	2.38E-12	1.25E-07	9.49E-08
242Pu	0.00E+00	1.69E-11	1.50E-11	1.50E-11	1.25E-20	1.12E-13	1.36E-13	3.43E-18	1.78E-13	1.36E-13
Total	8.86E-08	2.18E-02	1.93E-02	1.93E-02	1.56E-12	1.00E-07	5.82E-02	1.97E-09	7.63E-02	5.82E-02
Frequency, /hr	2.50E-04	1.30E-04	8.00E-04	1.30E-03	3.50E-04	1.10E-03	1.80E-03	1.80E-03	9.00E-04	1.00E-03

TABLE C-10. Curries of Specific Nuclides Released to Retention Basin due to Coil and Tube Failures (F-Canyon)

Nuclide	Decon- tamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Pu Cycle	MAN Evaporator	LAN Evaporator	Ion Exchange	High Heat Waste ¹
89Sr	1.26E-02	1.32E+00	1.32E+00	1.32E+00	1.07E-10	6.07E-06	1.33E+00	4.53E-08	1.33E+00	1.33E+00
90Sr	1.28E-03	1.26E-01	1.26E-01	1.26E-01	1.02E-11	6.55E-07	1.28E-01	4.35E-09	1.28E-01	1.28E-01
90Y	0.00E+00	1.26E-01	1.26E-01	1.26E-01	1.02E-11	6.55E-07	1.28E-01	4.35E-09	1.28E-01	1.28E-01
91Y	2.20E-02	2.02E+00	2.02E+00	2.02E+00	1.64E-10	1.05E-05	2.06E+00	6.99E-08	2.06E+00	2.06E+00
95Zr	3.40E-02	2.02E+00	2.02E+00	2.02E+00	1.64E-10	1.05E-05	2.04E+00	6.94E-08	2.04E+00	2.04E+00
95Nb	6.20E-02	5.42E-01	5.42E-01	5.42E-01	4.39E-11	2.02E-06	5.47E-01	1.06E-08	5.47E-01	5.47E-01
103Ru	1.14E-02	9.96E-01	9.96E-01	9.96E-01	8.07E-11	5.10E-06	1.01E+00	3.42E-08	1.01E+00	1.01E+00
106Ru	1.46E-02	1.02E+00	1.02E+00	1.02E+00	8.26E-11	5.30E-06	1.03E+00	3.50E-08	1.03E+00	1.03E+00
106Rh	0.00E+00	1.02E+00	1.02E+00	1.02E+00	8.26E-11	5.30E-06	1.03E+00	3.50E-08	1.03E+00	1.03E+00
110Ag	0.00E+00	1.30E-02	1.30E-02	1.30E-02	1.05E-12	6.76E-08	1.30E-02	4.42E-10	1.30E-02	1.30E-02
123Sn	0.00E+00	1.70E-02	1.70E-02	1.70E-02	1.38E-12	8.84E-08	1.75E-02	5.95E-10	1.75E-02	1.75E-02
125Sb	0.00E+00	2.20E-02	2.20E-02	2.20E-02	1.78E-12	1.14E-07	2.10E-02	7.14E-10	2.10E-02	2.10E-02
127Te	0.00E+00	3.30E-02	3.30E-02	3.30E-02	2.67E-12	1.72E-07	3.30E-02	1.12E-09	3.30E-02	3.30E-02
129Te	0.00E+00	2.10E-02	2.10E-02	2.10E-02	1.70E-12	1.09E-07	2.10E-02	7.14E-10	2.10E-02	2.10E-02
134Cs	7.20E-04	2.20E-02	2.20E-02	2.20E-02	1.78E-12	1.14E-07	2.22E-01	7.55E-09	2.22E-01	2.22E-01
137Cs	1.98E-03	1.69E-01	1.69E-01	1.69E-01	1.37E-11	8.79E-07	1.70E-01	5.78E-09	1.70E-01	1.70E-01
141Ce	0.00E+00	6.55E-01	6.55E-01	6.55E-01	5.31E-11	3.41E-06	6.60E-01	2.24E-08	6.60E-01	6.60E-01
144Ce	3.40E-02	3.04E+00	3.04E+00	3.04E+00	2.46E-10	1.58E-05	3.08E+00	1.05E-07	3.08E+00	3.08E+00
144Pr	0.00E+00	3.04E+00	3.04E+00	3.04E+00	2.46E-10	1.58E-05	3.08E+00	1.05E-07	3.08E+00	3.08E+00
147Pm	5.60E-03	5.46E-01	5.46E-01	5.46E-01	4.42E-11	2.84E-06	5.51E-01	1.87E-08	5.51E-01	5.51E-01
148Pm	0.00E+00	6.36E-03	6.36E-03	6.36E-03	5.15E-13	3.31E-08	6.43E-03	2.19E-10	6.43E-03	6.43E-03
155Eu	0.00E+00	9.57E-03	9.57E-03	9.57E-03	7.75E-13	4.98E-08	9.66E-03	3.28E-10	9.66E-03	9.66E-03
234U	0.00E+00	2.52E-08	2.52E-08	2.52E-08	2.04E-18	1.31E-13	2.40E-12	8.16E-20	2.40E-12	2.40E-12
235U	6.00E-13	4.03E-07	4.03E-07	4.03E-07	3.26E-17	2.10E-12	4.00E-11	1.36E-18	4.00E-11	4.00E-11
236U	4.00E-13	3.64E-07	3.64E-07	3.64E-07	2.95E-17	1.89E-12	4.00E-11	1.36E-18	4.00E-11	4.00E-11
238U	5.80E-11	3.72E-05	3.72E-05	3.72E-05	3.01E-15	1.93E-10	4.00E-09	1.36E-16	4.00E-09	4.00E-09
237Np	0.00E+00	5.36E-08	5.36E-08	5.36E-08	4.34E-18	2.79E-13	1.09E-09	3.71E-17	5.41E-08	1.09E-09
238Pu	2.80E-08	2.53E-05	2.53E-05	2.53E-05	2.05E-15	1.32E-10	1.60E-09	5.44E-17	7.06E-08	1.60E-09
239Pu	3.00E-07	7.62E-03	7.62E-03	7.62E-03	6.17E-13	3.96E-08	4.60E-07	1.56E-14	2.31E-06	4.60E-07
240Pu	1.35E-06	1.73E-03	1.73E-03	1.73E-03	1.40E-13	9.00E-09	1.05E-06	3.57E-14	5.24E-06	1.05E-06
241Pu	2.60E-06	1.15E-01	1.15E-01	1.15E-01	9.32E-12	5.98E-07	7.00E-06	2.30E-13	1.50E-04	7.00E-06
242Pu	0.00E+00	1.68E-07	1.68E-07	1.68E-07	1.36E-17	8.74E-13	1.02E-11	3.47E-19	5.10E-05	1.02E-11
Total	2.00E-01	1.69E+01	1.69E+01	1.69E+01	1.37E-09	8.80E-05	1.72E+01	5.84E-07	1.72E+01	1.72E+01
Frequency, /hr	1.50E-06	7.20E-06	8.40E-06	5.33E-06	5.33E-06	5.33E-06	3.90E-05	3.90E-05	2.40E-06	8.40E-06

TABLE C-11. Curies of Specific Nuclides Released to Four Mile Creek due to Coil and Tube Failures (F-Canyon)

Nuclide	Decon- tamination	Dissolver	Head End	First Cycle	2nd B Cycle	2nd Pu Cycle	MAN Evaporator	LAN Evaporator	Ion Exchange	High Heat Waste
89Sr	1.26E-02	1.32E+00	1.32E+00	1.32E+00	1.07E-10	6.87E-06	1.33E+00	4.53E-08	1.33E+00	1.33E+00
90Sr	1.28E-03	1.26E-01	1.26E-01	1.26E-01	1.02E-11	6.55E-07	1.28E-01	4.35E-09	1.28E-01	1.28E-01
90Y	0.00E+00	1.26E-01	1.26E-01	1.26E-01	1.02E-11	6.55E-07	1.28E-01	4.35E-09	1.28E-01	1.28E-01
91Y	2.20E-02	2.02E+00	2.02E+00	2.02E+00	1.64E-10	1.05E-05	2.06E+00	6.99E-08	2.06E+00	2.06E+00
95Zr	3.40E-02	2.02E+00	2.02E+00	2.02E+00	1.64E-10	1.05E-05	2.04E+00	6.94E-08	2.04E+00	2.04E+00
95Nb	6.20E-02	5.42E-01	5.42E-01	5.42E-01	4.39E-11	2.82E-06	5.47E-01	1.86E-08	5.47E-01	5.47E-01
103Ru	1.14E-02	9.96E-01	9.96E-01	9.96E-01	8.07E-11	5.18E-06	1.01E+00	3.42E-08	1.01E+00	1.01E+00
106Ru	1.46E-02	1.02E+00	1.02E+00	1.02E+00	8.26E-11	5.30E-06	1.03E+00	3.50E-08	1.03E+00	1.03E+00
106Rh	0.00E+00	1.02E+00	1.02E+00	1.02E+00	8.26E-11	5.30E-06	1.03E+00	3.50E-08	1.03E+00	1.03E+00
110Ag	0.00E+00	1.30E-02	1.30E-02	1.30E-02	1.05E-12	6.76E-08	1.30E-02	4.42E-10	1.30E-02	1.30E-02
123Sn	0.00E+00	1.70E-02	1.70E-02	1.70E-02	1.38E-12	8.84E-08	1.75E-02	5.95E-10	1.75E-02	1.75E-02
125Sb	0.00E+00	2.20E-02	2.20E-02	2.20E-02	1.76E-12	1.14E-07	2.10E-02	7.14E-10	2.10E-02	2.10E-02
127Te	0.00E+00	3.30E-02	3.30E-02	3.30E-02	2.67E-12	1.72E-07	3.30E-02	1.12E-09	3.30E-02	3.30E-02
129Te	0.00E+00	2.10E-02	2.10E-02	2.10E-02	1.70E-12	1.09E-07	2.10E-02	7.14E-10	2.10E-02	2.10E-02
134Cs	7.20E-04	2.20E-02	2.20E-02	2.20E-02	1.78E-12	1.14E-07	2.22E-01	7.55E-09	2.22E-01	2.22E-01
137Cs	1.98E-03	1.69E-01	1.69E-01	1.69E-01	1.37E-11	8.79E-07	1.70E-01	5.78E-09	1.70E-01	1.70E-01
141Ce	0.00E+00	6.55E-01	6.55E-01	6.55E-01	5.31E-11	3.41E-06	6.60E-01	2.24E-08	6.60E-01	6.60E-01
144Ce	3.40E-02	3.04E+00	3.04E+00	3.04E+00	2.46E-10	1.58E-05	3.08E+00	1.05E-07	3.08E+00	3.08E+00
144Pr	0.00E+00	3.04E+00	3.04E+00	3.04E+00	2.46E-10	1.58E-05	3.08E+00	1.05E-07	3.08E+00	3.08E+00
147Pm	5.60E-03	5.46E-01	5.46E-01	5.46E-01	4.42E-11	2.84E-06	5.51E-01	1.87E-08	5.51E-01	5.51E-01
148Pm	0.00E+00	6.36E-03	6.36E-03	6.36E-03	5.15E-13	3.31E-08	6.43E-03	2.19E-10	6.43E-03	6.43E-03
155Eu	0.00E+00	9.57E-03	9.57E-03	9.57E-03	7.75E-13	4.98E-08	9.66E-03	3.28E-10	9.66E-03	9.66E-03
234U	0.00E+00	2.52E-08	2.52E-08	2.52E-08	2.04E-18	1.31E-13	2.40E-12	8.16E-20	2.40E-12	2.40E-12
235U	6.00E-13	4.03E-07	4.03E-07	4.03E-07	3.26E-17	2.10E-12	4.00E-11	1.36E-18	4.00E-11	4.00E-11
236U	4.00E-13	3.64E-07	3.64E-07	3.64E-07	2.95E-17	1.89E-12	4.00E-11	1.36E-18	4.00E-11	4.00E-11
238U	5.80E-11	3.72E-05	3.72E-05	3.72E-05	3.01E-15	1.93E-10	4.00E-09	1.36E-16	4.00E-09	4.00E-09
237Np	0.00E+00	5.36E-08	5.36E-08	5.36E-08	4.34E-18	2.79E-13	1.09E-09	3.71E-17	5.41E-08	1.09E-09
238Pu	2.80E-08	2.53E-05	2.53E-05	2.53E-05	2.05E-15	1.32E-10	1.60E-09	5.44E-17	7.06E-08	1.60E-09
239Pu	3.00E-07	7.62E-03	7.62E-03	7.62E-03	6.17E-13	3.96E-08	4.60E-07	1.56E-14	2.31E-06	4.60E-07
240Pu	1.35E-06	1.73E-03	1.73E-03	1.73E-03	1.40E-13	9.00E-09	1.05E-06	3.57E-14	5.24E-06	1.05E-06
241Pu	2.60E-06	1.15E-01	1.15E-01	1.15E-01	9.32E-12	5.98E-07	7.00E-06	2.38E-13	3.50E-04	7.00E-06
242Pu	0.00E+00	1.68E-07	1.68E-07	1.68E-07	1.36E-17	8.74E-13	1.02E-11	3.47E-19	5.10E-05	1.02E-11
Total	2.00E-01	1.69E+01	1.69E+01	1.69E+01	1.37E-09	8.80E-05	1.72E+01	5.84E-07	1.72E+01	1.72E+01
Frequency, /hr	6.00E-08	2.88E-07	3.36E-07	2.13E-07	2.13E-07	2.13E-07	1.56E-06	1.56E-06	9.60E-08	3.36E-07

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APPENDIX D
F-CANYON DOSE CALCULATIONS

TABLE D-1. Dose and Risk to the Offsite Population from Ground Level Release due to Earthquake

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/Y.	DOSE, PERSON-REM						
DISSOLVER	2.53E-01	2.00E-04	5.90E-06	2.95E-02	3.12E-01	9.72E-01	1.09E-01	1.56E-01	2.35E-04	4.15E+00
HEAD END	2.29E-01	2.00E-04	5.36E-06	2.68E-02	2.83E-01	8.83E-01	9.85E-02	1.42E-01	2.13E-04	3.77E+00
FIRST CYCLE	2.29E-01	2.00E-04	5.36E-06	2.68E-02	2.83E-01	8.83E-01	9.85E-02	1.42E-01	2.13E-04	3.77E+00
2ND U CYCLE	5.39E-11	2.00E-04	1.09E-14	5.44E-11	6.65E-11	2.01E-09	1.99E-10	2.32E-10	5.17E-14	1.39E-09
2ND Pu CYCLE	1.19E-06	2.00E-04	2.98E-08	1.49E-04	3.13E-07	5.93E-03	6.13E-04	6.25E-04	2.18E-10	3.50E-04
HAN EVAPORATOR	4.11E-01	2.00E-04	2.44E-06	1.22E-02	5.08E-01	1.60E-01	2.91E-02	1.04E-01	3.77E-04	6.68E+00
LAR EVAPORATOR	1.39E-08	2.00E-04	6.38E-13	3.19E-09	1.65E-08	1.15E-07	1.23E-08	1.50E-08	1.45E-11	2.24E-07
ION EXCHANGE	3.77E-01	2.00E-04	2.26E-06	1.13E-02	4.68E-01	1.47E-01	2.67E-02	9.59E-02	3.47E-04	6.16E+00
HIGH HEAT WASTE	4.11E-01	2.00E-04	2.44E-06	1.22E-02	5.08E-01	1.60E-01	2.91E-02	1.04E-01	3.77E-04	6.68E+00
TOTAL			2.38E-05							

TABLE D-2. Dose and Risk to the Maximum Individual from Ground Level Release due to Earthquake

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR							
DISSOLVER	2.53E-01	2.00E-04	2.28E-06	1.14E-02	6.36E-02	3.47E-01	3.62E-02	5.63E-02	1.18E-04	2.02E+00
HEAD END	2.29E-01	2.00E-04	2.06E-06	1.03E-02	5.77E-02	3.15E-01	3.28E-02	5.11E-02	1.07E-04	1.83E+00
FIRST CYCLE	2.29E-01	2.00E-04	2.06E-06	1.03E-02	5.77E-02	3.15E-01	3.28E-02	5.11E-02	1.07E-04	1.83E+00
2ND U CYCLE	5.39E-11	2.00E-04	3.88E-15	1.94E-11	1.36E-11	6.94E-10	6.05E-11	7.24E-11	2.60E-14	6.78E-10
2ND Pu CYCLE	1.19E-06	2.00E-04	9.66E-09	4.83E-05	6.40E-08	1.98E-03	1.86E-04	1.81E-04	1.09E-10	1.72E-04
NAF EVAPORATOR	4.11E-01	2.00E-04	1.37E-06	6.86E-03	1.04E-01	9.09E-02	1.41E-02	4.82E-02	1.89E-04	3.25E+00
LAF EVAPORATOR	1.39E-08	2.00E-04	2.26E-13	1.13E-09	3.35E-09	3.94E-08	3.86E-09	4.91E-09	7.28E-12	1.09E-07
ION EXCHANGE	3.77E-01	2.00E-04	1.26E-06	6.31E-03	9.55E-02	8.36E-02	1.30E-02	4.44E-02	1.74E-04	2.99E+00
HIGH HEAT WASTE	4.11E-01	2.00E-04	1.37E-06	6.86E-03	1.04E-01	9.09E-02	1.41E-02	4.82E-02	1.89E-04	3.25E+00
TOTAL			1.04E-05							

TABLE D-3. Dose and Risk to Onsite Population from Ground Level Release due to Earthquake

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
ONSITE POPULATION			PERSON-REM/YR		DOSE, PERSON-REM					
DISSOLVER	2.53E-01	2.00E-04	1.95E-06	9.73E-03	1.20E-01	3.31E-01	3.75E-02	5.22E-02	4.63E-05	1.16E+00
HEAD END	2.29E-01	2.00E-04	1.77E-06	8.83E-03	1.09E-01	3.00E-01	3.40E-02	4.73E-02	4.19E-05	1.06E+00
FIRST CYCLE	2.29E-01	2.00E-04	1.77E-06	8.83E-03	1.09E-01	3.00E-01	3.40E-02	4.73E-02	4.19E-05	1.06E+00
2ND U CYCLE	5.39E-11	2.00E-04	3.70E-15	1.85E-11	2.56E-11	6.89E-10	7.07E-11	8.14E-11	1.02E-14	3.89E-10
2ND Pu CYCLE	1.19E-06	2.00E-04	1.04E-08	5.21E-05	1.20E-07	2.06E-03	2.18E-04	2.25E-04	4.30E-11	9.78E-05
BAR EVAPORATOR	4.11E-01	2.00E-04	6.62E-07	3.31E-03	1.95E-01	4.34E-02	6.46E-03	3.09E-02	7.44E-05	1.87E+00
LAN EVAPORATOR	1.39E-08	2.00E-04	2.16E-13	1.08E-09	6.34E-09	3.96E-08	4.31E-09	5.18E-09	2.86E-12	6.29E-08
ION EXCHANGE	3.77E-01	2.00E-04	6.08E-07	3.04E-03	1.80E-01	3.99E-02	7.77E-03	2.85E-02	6.84E-05	1.73E+00
HIGH HEAT WASTE	4.11E-01	2.00E-04	6.62E-07	3.31E-03	1.95E-01	4.34E-02	6.46E-03	3.09E-02	7.44E-05	1.87E+00
TOTAL			7.42E-06							

TABLE D-4. Dose and Risk to the Offsite Population from Stack Release due to Externally Induced Failures

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/YR	DOSE, PERSON-REM						
DISSOLVER	2.48E+00	8.76E-03	1.87E-03	2.14E-01	2.24E+00	7.04E+00	7.87E-01	1.13E+00	2.49E-03	2.99E+01
HEAD END	2.25E+00	8.76E-03	1.70E-03	1.94E-01	2.05E+00	6.39E+00	7.14E-01	1.03E+00	2.27E-03	2.73E+01
FIRST CYCLE	2.25E+00	8.76E-03	1.70E-03	1.94E-01	2.05E+00	6.39E+00	7.14E-01	1.03E+00	2.27E-03	2.73E+01
2ND U CYCLE	5.28E-10	8.76E-03	3.44E-12	3.93E-10	4.81E-10	1.45E-08	1.44E-09	1.67E-09	5.53E-13	1.01E-08
2ND Pu CYCLE	1.17E-05	8.76E-03	9.46E-06	1.08E-03	2.26E-06	4.29E-02	4.44E-03	4.52E-03	2.33E-09	2.54E-03
HAN EVAPORATOR	4.02E+00	8.76E-03	7.88E-04	9.00E-02	3.68E+00	1.16E+00	2.12E-01	7.56E-01	4.02E-03	4.85E+01
LAN EVAPORATOR	1.36E-07	8.76E-03	2.01E-10	2.30E-08	1.19E-07	8.26E-07	8.83E-08	1.08E-07	1.55E-10	1.62E-06
ION EXCHANGE	3.70E+00	8.76E-03	7.24E-04	8.26E-02	3.38E+00	1.07E+00	1.95E-01	6.93E-01	3.69E-03	4.44E+01
HIGH HEAT WASTE	4.02E+00	8.76E-03	7.88E-04	9.00E-02	3.68E+00	1.16E+00	2.12E-01	7.56E-01	4.02E-03	4.85E+01
TOTAL			7.58E-03							

TABLE D-5. Dose and Risk to the Maximum Individual from Stack Release due to Externally Induced Failures

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT /YR	BONE	LIVER BODY	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR							
DISSOLVER	2.48E+00	8.76E-03	2.85E-04	3.25E-02	1.79E-01	9.83E-01	1.03E-01	1.59E-01	7.14E-04	5.68E+00
HEAD END	2.25E+00	8.76E-03	2.58E-04	2.95E-02	1.64E-01	8.92E-01	9.32E-02	1.45E-01	6.51E-04	5.18E+00
FIRST CYCLE	2.25E+00	8.76E-03	2.58E-04	2.95E-02	1.64E-01	8.92E-01	9.32E-02	1.45E-01	6.51E-04	5.18E+00
2ND U CYCLE	5.28E-10	8.76E-03	4.80E-13	5.48E-11	3.84E-11	1.96E-09	1.71E-10	2.05E-10	1.58E-13	1.92E-09
2ND Pu CYCLE	1.17E-05	8.76E-03	1.20E-06	1.37E-04	1.81E-07	5.59E-03	5.26E-04	5.12E-04	6.67E-10	4.86E-04
HAN EVAPORATOR	4.02E+00	8.76E-03	1.74E-04	1.99E-02	2.98E-01	2.61E-01	4.07E-02	1.39E-01	7.69E-04	9.35E+00
LAN EVAPORATOR	1.36E-07	8.76E-03	2.82E-11	3.22E-09	9.49E-09	1.11E-07	1.09E-08	1.38E-08	4.44E-11	3.07E-07
ION EXCHANGE	3.70E+00	8.76E-03	1.60E-04	1.83E-02	2.73E-01	2.40E-01	3.74E-02	1.27E-01	7.06E-04	8.57E+00
HIGH HEAT WASTE	4.02E+00	8.76E-03	1.74E-04	1.99E-02	2.98E-01	2.61E-01	4.07E-02	1.39E-01	7.69E-04	9.35E+00
TOTAL			1.31E-03							

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TABLE D-6. Dose and Risk to the Onsite Population from Stack Release due to Externally Induced Failures

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
ON-SITE POPULATION			PERSON-REM/YR		DOSE, PERSON-REM					
DISSOLVER	2.48E+00	8.76E-03	2.08E-04	2.37E-02	2.89E-01	8.01E-01	9.10E-02	1.26E-01	2.97E-04	2.80E+00
HEAD END	2.25E+00	8.76E-03	1.88E-04	2.15E-02	2.64E-01	7.27E-01	8.25E-02	1.15E-01	2.71E-04	2.56E+00
FIRST CYCLE	2.25E+00	8.76E-03	1.88E-04	2.15E-02	2.64E-01	7.27E-01	8.25E-02	1.15E-01	2.71E-04	2.56E+00
2ND U CYCLE	5.28E-10	8.76E-03	3.92E-13	4.47E-11	6.20E-11	1.67E-09	1.71E-10	1.97E-10	6.59E-14	9.42E-10
2ND Pu CYCLE	1.17E-05	8.76E-03	1.10E-06	1.26E-04	2.91E-07	4.99E-03	5.29E-04	5.45E-04	2.70E-10	2.37E-04
MAN EVAPORATOR	4.02E+00	8.76E-03	7.28E-05	8.31E-03	4.74E-01	1.05E-01	2.08E-02	7.52E-02	4.79E-04	4.55E+00
LAN EVAPORATOR	1.36E-07	8.76E-03	2.30E-11	2.62E-09	1.53E-08	9.52E-08	1.04E-08	1.25E-08	1.85E-11	1.52E-07
ION EXCHANGE	3.70E+00	8.76E-03	6.68E-05	7.63E-03	4.35E-01	9.68E-02	1.91E-02	6.90E-02	4.40E-04	4.17E+00
HIGH HEAT WASTE	4.02E+00	8.76E-03	7.28E-05	8.31E-03	4.74E-01	1.05E-01	2.08E-02	7.52E-02	4.79E-04	4.55E+00
TOTAL			7.98E-04							

TABLE D-7. Dose and Risk to the Offsite Population from Stack Release due to Fire

UNIT OPERATION	TOTAL C ₁ RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/YR		DOSE, PERSON-REM					
FIRST CYCLE	4.58E-01	6.13E-04	2.70E-04	4.41E-01	2.70E-01	1.64E+01	1.88E+00	1.79E+00	2.77E-04	3.96E+00
2ND U CYCLE	3.63E-06	6.13E-04	2.69E-08	4.38E-05	1.72E-06	1.64E-03	1.63E-04	1.88E-04	4.46E-09	4.41E-04
2ND Pu CYCLE	3.48E-03	6.13E-04	2.30E-04	3.75E-01	2.33E-04	1.49E+01	1.55E+00	1.57E+00	1.92E-07	8.75E-01
ION EXCHANGE	9.11E-01	7.86E-03	1.93E-04	2.45E-02	8.19E-01	3.54E-01	7.23E-02	1.80E-01	9.08E-04	1.07E-01
TOTAL			6.94E-04							

TABLE D-8. Dose and Risk to the Maximum Individual from Stack Release due to Fire

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR							
FIRST CYCLE	4.58E-01	6.13E-04	3.74E-05	6.10E-02	2.19E-02	2.25E+00	2.44E-01	2.16E-01	7.87E-05	7.52E-01
2ND W CYCLE	3.63E-06	6.13E-04	3.70E-09	6.03E-06	1.37E-07	2.21E-04	1.93E-05	2.15E-05	1.27E-09	8.44E-05
2ND Pu CYCLE	3.48E-03	6.13E-04	2.92E-05	4.76E-02	1.88E-05	1.95E+00	1.83E-01	1.78E-01	5.46E-08	1.68E-01
ION EXCHANGE	9.11E-01	7.88E-03	4.25E-05	5.39E-03	6.63E-02	7.92E-02	1.39E-02	3.31E-02	1.72E-04	2.07E+00
TOTAL			1.09E-04							

TABLE D-9. Dose and Risk to the Onsite Population from Stack Release due to Fire

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/YR	DOSE, PERSON-REM						
FIRST CYCLE	4.50E-01	6.13E-04	3.08E-05	5.03E-02	3.48E-02	1.88E+00	2.17E-01	2.12E-01	3.28E-05	3.72E-01
2ND U CYCLE	3.63E-06	6.13E-04	3.07E-09	5.00E-06	2.22E-07	1.88E-04	1.94E-05	2.14E-05	5.30E-10	4.11E-05
2ND Pu CYCLE	3.48E-03	6.13E-04	2.69E-05	4.39E-02	2.99E-05	1.74E+00	1.84E-01	1.98E-01	2.28E-08	8.17E-02
ION EXCHANGE	9.11E-01	7.88E-03	1.79E-05	2.27E-03	1.05E-01	3.23E-03	7.08E-03	1.88E-02	1.07E-04	1.01E+00
TOTAL			7.57E-05							

TABLE D-10. Dose and Risk to the Offsite Population from Stack Release due to Uncontrolled Reaction

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/YR		DOSE, PERSON-REM					
DISSOLVER	2.31E-01	2.63E-01	9.20E-03	3.50E-02	2.04E-01	1.22E+00	1.39E-01	1.65E-01	2.27E-04	2.72E+00
HEAD END	2.09E-01	4.30E-02	1.55E-03	3.55E-02	1.04E-01	1.27E+00	1.37E-01	1.65E-01	2.04E-04	2.46E+00
FIRST CYCLE	2.96E-01	2.92E-03	4.91E-05	1.60E-02	1.03E-01	5.40E-01	5.70E-02	0.73E-02	2.00E-04	2.41E+00
2ND B CYCLE	1.30E-07	2.92E-03	4.05E-09	1.66E-06	6.52E-00	6.21E-05	6.17E-06	6.03E-06	1.69E-10	1.67E-05
2ND Pu CYCLE	1.32E-04	2.92E-03	4.15E-05	1.42E-02	0.05E-06	5.60E-01	5.07E-02	5.90E-02	7.20E-09	3.33E-02
HAZ EVAPORATOR	5.99E-01	6.31E-01	1.03E-02	1.64E-02	5.30E-01	2.30E-01	4.90E-02	1.19E-01	5.90E-04	7.05E+00
LAH EVAPORATOR	6.75E-06	1.50E-01	1.35E-06	0.54E-06	2.05E-06	3.25E-04	3.51E-05	3.51E-05	1.20E-00	5.43E-05
ION EXCHANGE	7.91E-01	5.26E-01	1.12E-02	2.13E-02	7.11E-01	3.07E-01	6.20E-02	1.56E-01	7.01E-04	9.32E+00
HIGH HEAT WASTE	5.99E-01	4.30E-02	7.10E-04	1.64E-02	5.30E-01	2.30E-01	4.90E-02	1.19E-01	5.90E-04	7.05E+00
TOTAL			3.31E-02							

TABLE D-11. Dose and Risk to the Maximum Individual from Stack Release due to Uncontrolled Reaction

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR		DOSE, MREM					
DISSOLVER	2.31E-01	2.63E-01	1.34E-03	5.09E-03	1.63E-02	1.69E-01	1.81E-02	2.19E-02	6.49E-05	5.17E-01
HEAD END	2.09E-01	4.38E-02	2.26E-04	5.16E-03	1.47E-02	1.77E-01	1.79E-02	2.18E-02	5.86E-05	4.68E-01
FIRST CYCLE	2.96E-01	2.92E-03	8.21E-06	2.81E-03	1.46E-02	8.35E-02	8.56E-03	1.31E-02	5.77E-05	4.57E-01
2ND U CYCLE	1.38E-07	2.92E-03	6.69E-10	2.29E-07	5.19E-09	8.38E-06	7.34E-07	8.14E-07	4.81E-11	3.20E-06
2ND Pu CYCLE	1.32E-04	2.92E-03	5.29E-06	1.81E-03	7.13E-07	7.40E-02	6.96E-03	6.77E-03	2.87E-09	6.37E-03
BAR EVAPORATOR	5.99E-01	6.31E-01	2.27E-03	3.60E-03	4.36E-02	5.32E-02	9.41E-03	2.19E-02	1.13E-04	1.36E+00
LAN EVAPORATOR	6.75E-06	1.58E-01	1.78E-07	1.13E-06	2.24E-07	4.31E-05	4.33E-06	4.08E-06	3.64E-09	1.01E-05
ION EXCHANGE	7.91E-01	5.26E-01	2.46E-03	4.68E-03	5.76E-02	6.88E-02	1.21E-02	2.87E-02	1.49E-04	1.88E+00
HIGH HEAT WASTE	5.99E-01	4.38E-02	1.58E-04	3.60E-03	4.36E-02	5.32E-02	9.41E-03	2.19E-02	1.13E-04	1.36E+00
TOTAL			6.47E-03							

TABLE D-12. Dose and Risk to the Onsite Population from Stack Release due to Uncontrolled Reaction

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
ON-SITE POPULATION			PERSON-REM/YR		DOSE, PERSON-REM					
DISSOLVER	2.31E-01	2.63E-01	1.04E-03	3.94E-03	2.62E-02	1.40E-01	1.60E-02	1.89E-02	2.70E-05	2.55E-01
HEAD END	2.09E-01	4.38E-02	1.74E-04	3.98E-03	2.37E-02	1.44E-01	1.58E-02	1.89E-02	2.44E-05	2.31E-01
FIRST CYCLE	2.96E-01	2.92E-03	5.23E-06	1.79E-03	2.35E-02	5.93E-02	6.38E-03	9.54E-03	2.40E-05	2.20E-01
2ND CYCLE	1.38E-07	2.92E-03	5.55E-10	1.90E-07	8.42E-09	7.14E-06	7.35E-07	8.10E-07	2.01E-11	1.56E-06
2ND P _u CYCLE	1.32E-04	2.92E-03	4.88E-06	1.67E-03	1.14E-06	6.68E-02	7.00E-03	7.21E-03	8.65E-10	3.11E-03
BAR EVAPORATOR	5.99E-01	6.31E-01	9.59E-04	1.52E-03	6.93E-02	2.17E-02	4.79E-03	1.19E-02	7.02E-05	6.62E-01
LAN EVAPORATOR	6.75E-06	1.58E-01	1.56E-07	9.87E-07	3.70E-07	3.76E-05	4.14E-06	4.19E-06	1.52E-09	5.19E-06
ION EXCHANGE	7.91E-01	5.26E-01	1.04E-03	1.97E-03	9.15E-02	2.80E-02	6.15E-03	1.56E-02	9.30E-05	8.74E-01
HIGH HEAT WASTE	5.99E-01	4.38E-02	6.66E-05	1.52E-03	6.93E-02	2.17E-02	4.79E-03	1.19E-02	7.02E-05	6.62E-01
TOTAL			3.28E-03							

TABLE D-13. Dose and Risk to the Offsite Population from Stack Release due to Criticality

UNIT OPERATION	TOTAL C1 RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/YR		DOSE, PERSON-REM					
DISSOLVER	4.76E+04	1.05E-06	1.81E-06	1.72E+00	1.90E+00	1.96E+00	2.62E+00	3.44E+00	1.61E+02	1.35E+00
HEAD END	4.76E+04	1.75E-05	3.01E-05	1.72E+00	1.90E+00	1.96E+00	2.62E+00	3.44E+00	1.61E+02	1.35E+00
FIRST CYCLE	4.76E+04	1.56E-03	2.68E-03	1.72E+00	1.90E+00	1.96E+00	2.62E+00	3.44E+00	1.61E+02	1.35E+00
2ND Pu CYCLE	4.76E+04	1.56E-03	2.68E-03	1.72E+00	1.90E+00	1.96E+00	2.62E+00	3.44E+00	1.61E+02	1.35E+00
TOTAL			5.40E-03							

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TABLE D-14. Dose and Risk to the Maximum Individual from Stack Release due to Criticality

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR		DOSE, MREM					
DISSOLVER	4.76E+04	1.05E-06	2.92E-06	2.78E+00	2.79E+00	2.80E+00	2.95E+00	3.09E+00	2.15E+01	2.74E+00
HEAD END	4.76E+04	1.75E-05	4.87E-05	2.78E+00	2.79E+00	2.80E+00	2.95E+00	3.09E+00	2.15E+01	2.74E+00
FIRST CYCLE	4.76E+04	1.56E-03	4.33E-03	2.78E+00	2.79E+00	2.80E+00	2.95E+00	3.09E+00	2.15E+01	2.74E+00
2ND Pu CYCLE	4.76E+04	1.56E-03	4.33E-03	2.78E+00	2.79E+00	2.80E+00	2.95E+00	3.09E+00	2.15E+01	2.74E+00
TOTAL			8.72E-03							

TABLE D-15. Dose and Risk to the Onsite Population from Stack Release due to Criticality

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
ONSITE POPULATION			PERSON-REM/YR	DOSE, PERSON-REM						
DISSOLVER	4.76E+04	1.05E-06	1.64E-06	1.56E+00	1.50E+00	1.63E+00	1.81E+00	2.01E+00	4.68E+01	1.37E+00
HEAD END	4.76E+04	1.75E-05	2.73E-05	1.56E+00	1.50E+00	1.63E+00	1.81E+00	2.01E+00	4.68E+01	1.37E+00
FIRST CYCLE	4.76E+04	1.56E-03	2.43E-03	1.56E+00	1.50E+00	1.63E+00	1.81E+00	2.01E+00	4.68E+01	1.37E+00
2ND Pu CYCLE	4.76E+04	1.56E-03	2.43E-03	1.56E+00	1.50E+00	1.63E+00	1.81E+00	2.01E+00	4.68E+01	1.37E+00
TOTAL			4.89E-03							

TABLE D-16. Dose and Risk to the Offsite Population from Ground Level Release due to Transfer Error to 211-F

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/YR							
DECONTAMINATION	9.86E-05	2.63E-05	4.39E-10	1.67E-05	8.50E-05	3.03E-04	8.61E-05	5.59E-05	1.50E-07	8.84E-04
DISSOLVER	5.35E+01	1.05E-04	6.56E-04	6.24E+00	6.58E+01	2.06E+02	2.30E+01	3.30E+01	4.96E-02	8.76E+02
HEAD END	4.81E+01	1.75E-05	9.83E-05	5.61E+00	5.92E+01	1.85E+02	2.07E+01	2.97E+01	4.45E-02	7.89E+02
FIRST CYCLE	4.81E+01	3.07E-05	1.72E-04	5.61E+00	5.92E+01	1.85E+02	2.07E+01	2.97E+01	4.45E-02	7.89E+02
2ND U CYCLE	3.89E-09	1.84E-05	7.21E-14	3.92E-09	4.80E-09	1.45E-07	1.43E-08	1.67E-08	3.73E-12	1.00E-07
2ND Pu CYCLE	2.50E-04	7.36E-05	2.30E-06	3.12E-02	6.57E-05	1.24E+00	1.29E-01	1.31E-01	4.50E-08	7.35E-02
HAN EVAPORATOR	1.43E+02	8.76E-05	3.74E-04	4.27E+00	1.77E+02	5.59E+01	1.01E+01	3.63E+01	1.31E-01	2.33E+03
LAN EVAPORATOR	4.83E-06	8.76E-05	9.72E-11	1.11E-06	5.72E-06	3.99E-05	4.26E-06	5.20E-06	5.04E-09	7.78E-05
ION EXCHANGE	1.31E+01	7.01E-05	2.75E-05	3.93E-01	1.63E+01	5.14E+00	9.33E-01	3.34E+00	1.21E-02	2.15E+02
HIGH HEAT WASTE	1.43E+02	1.75E-05	7.40E-05	4.27E+00	1.77E+02	5.59E+01	1.01E+01	3.63E+01	1.31E-01	2.33E+03
TOTAL			1.40E-03							

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TABLE D-17. Dose and Risk to the Maximum Individual from Ground Level Release due to Transfer Error to 211-F

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR		DOSE, MREM					
DECONTAMINATION	9.86E-05	2.63E-05	2.31E-10	8.79E-06	1.75E-05	1.68E-04	4.16E-05	2.56E-05	7.54E-08	4.24E-04
DISSOLVER	5.35E+01	1.05E-04	2.52E-04	2.40E+00	1.34E+01	7.34E+01	7.65E+00	1.19E+01	2.49E-02	4.26E+02
HEAD END	4.81E+01	1.75E-05	3.78E-05	2.16E+00	1.21E+01	6.60E+01	6.88E+00	1.07E+01	2.24E-02	3.83E+02
FIRST CYCLE	4.81E+01	3.07E-05	6.62E-05	2.16E+00	1.21E+01	6.60E+01	6.88E+00	1.07E+01	2.24E-02	3.83E+02
2ND B CYCLE	3.89E-09	1.84E-05	2.58E-14	1.40E-09	9.79E-10	5.01E-08	4.36E-09	5.22E-09	1.87E-12	4.90E-08
2ND P _B CYCLE	2.50E-04	7.36E-05	7.43E-07	1.01E-02	1.34E-05	4.14E-01	3.90E-02	3.79E-02	2.30E-08	3.60E-02
BAR EVAPORATOR	1.43E+02	8.76E-05	2.09E-04	2.39E+00	3.61E+01	3.17E+01	4.92E+00	1.68E+01	6.60E-02	1.13E+03
LAN EVAPORATOR	4.83E-06	8.76E-05	3.45E-11	3.94E-07	1.16E-06	1.37E-05	1.34E-06	1.70E-06	2.53E-09	3.77E-05
ION EXCHANGE	1.31E+01	7.01E-05	1.54E-05	2.20E-01	3.32E+00	2.91E+00	4.52E-01	1.55E+00	6.07E-03	1.04E+02
HIGH HEAT WASTE	1.43E+02	1.75E-05	4.19E-05	2.39E+00	3.61E+01	3.17E+01	4.92E+00	1.68E+01	6.60E-02	1.13E+03
TOTAL			6.24E-04							

TABLE D-10. Dose and Risk to the Onsite Population from Ground Level Release due to Transfer Error to 211-F

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
ONSITE POPULATION			PERSON-REM/YR	DOSE, PERSON-REM						
DECONTAMINATION	9.86E-05	2.63E-05	1.25E-10	4.74E-06	3.30E-05	8.33E-05	2.52E-05	1.67E-05	2.98E-08	2.51E-04
DISSOLVER	5.35E+01	1.05E-04	2.17E-04	2.06E+00	2.53E+01	6.99E+01	7.93E+00	1.10E+01	9.77E-03	2.46E+02
HEAD END	4.81E+01	1.75E-05	3.24E-05	1.85E+00	2.28E+01	6.29E+01	7.13E+00	9.92E+00	8.78E-03	2.21E+02
FIRST CYCLE	4.81E+01	3.07E-05	5.67E-05	1.85E+00	2.28E+01	6.29E+01	7.13E+00	9.92E+00	8.78E-03	2.21E+02
2ND W CYCLE	3.89E-09	1.84E-05	2.45E-14	1.33E-09	1.85E-09	4.97E-08	5.10E-09	5.87E-09	7.35E-13	2.81E-08
2ND Pu CYCLE	2.50E-04	7.36E-05	8.02E-07	1.09E-02	2.53E-05	4.32E-01	4.58E-02	4.72E-02	9.02E-09	2.05E-02
HAR EVAPORATOR	1.43E+02	8.76E-05	1.01E-04	1.15E+00	6.81E+01	1.51E+01	2.95E+00	1.08E+01	2.59E-02	6.54E+02
LAR EVAPORATOR	4.83E-06	8.76E-05	3.29E-11	3.76E-07	2.20E-06	1.37E-05	1.50E-06	1.80E-06	9.93E-10	2.18E-05
ION EXCHANGE	1.31E+01	7.01E-05	7.43E-06	1.06E-01	6.27E+00	1.39E+00	2.71E-01	9.91E-01	2.38E-03	6.01E+01
HIGH HEAT WASTE	1.43E+02	1.75E-05	2.01E-05	1.15E+00	6.81E+01	1.51E+01	2.95E+00	1.08E+01	2.59E-02	6.54E+02
TOTAL			4.35E-04							

TABLE D-19. Dose and Risk to the Offsite Population from Stack Release due to Transfer Error to Sump

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/YR	DOSE, PERSON-REM						
DECONTAMINATION	4.83E-07	3.33E-03	2.01E-10	6.05E-08	3.04E-07	1.09E-06	3.11E-07	2.02E-07	8.11E-10	3.19E-06
DISSOLVER	2.62E-01	3.94E-02	8.91E-04	2.26E-02	2.38E-01	7.44E-01	8.32E-02	1.19E-01	2.64E-04	3.17E+00
HEAD END	2.36E-01	6.57E-03	1.34E-04	2.04E-02	2.15E-01	6.69E-01	7.48E-02	1.07E-01	2.38E-04	2.86E+00
FIRST CYCLE	2.36E-01	1.14E-02	2.32E-04	2.04E-02	2.15E-01	6.69E-01	7.48E-02	1.07E-01	2.38E-04	2.86E+00
2ND U CYCLE	1.91E-11	6.92E-03	9.90E-14	1.43E-11	1.98E-11	5.26E-10	5.19E-11	6.04E-11	1.99E-14	3.77E-10
2ND Pu CYCLE	1.23E-06	2.72E-02	3.07E-06	1.13E-04	2.37E-07	4.49E-03	4.65E-04	4.74E-04	2.44E-10	2.66E-04
HAN EVAPORATOR	7.00E-01	3.33E-02	5.19E-04	1.56E-02	6.40E-01	2.02E-01	3.68E-02	1.31E-01	7.01E-04	8.43E+00
LAR EVAPORATOR	2.36E-08	3.33E-02	1.33E-10	4.01E-09	2.07E-08	1.44E-07	1.54E-08	1.88E-08	2.70E-11	2.81E-07
ION EXCHANGE	6.44E-02	2.63E-02	3.78E-05	1.44E-03	5.88E-02	1.86E-02	3.39E-03	1.21E-02	6.44E-05	7.75E-01
HIGH HEAT WASTE	7.00E-01	6.57E-03	1.02E-04	1.56E-02	6.40E-01	2.02E-01	3.68E-02	1.31E-01	7.01E-04	8.43E+00
TOTAL			1.92E-03							

TABLE D-20. Dose and Risk to the Maximum Individual from Stack Release due to Transfer Error to Sump

UNIT OPERATION	TOTAL C ₁ RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR							
DECONTAMINATION	4.83E-07	3.33E-03	4.16E-11	1.25E-08	2.49E-08	2.37E-07	5.87E-08	3.62E-08	2.31E-10	5.98E-07
DISSOLVER	2.62E-01	3.94E-02	1.35E-04	3.43E-03	1.90E-02	1.04E-01	1.09E-02	1.69E-02	7.57E-05	6.02E-01
HEAD END	2.36E-01	6.57E-03	2.03E-05	3.09E-03	1.71E-02	9.34E-02	9.76E-03	1.52E-02	6.82E-05	5.43E-01
FIRST CYCLE	2.36E-01	1.14E-02	3.52E-05	3.09E-03	1.71E-02	9.34E-02	9.76E-03	1.52E-02	6.82E-05	5.43E-01
2ND U CYCLE	1.91E-11	6.92E-03	1.38E-14	2.00E-12	1.59E-12	7.15E-11	6.17E-12	7.38E-12	5.71E-15	7.19E-11
2ND Pu CYCLE	1.23E-06	2.72E-02	3.88E-07	1.43E-05	1.90E-08	5.86E-04	5.51E-05	5.36E-05	6.97E-11	5.09E-05
HEAVY EVAPORATOR	7.00E-01	3.33E-02	1.16E-04	3.47E-03	5.19E-02	4.54E-02	7.07E-03	2.41E-02	1.34E-04	1.62E+00
LAN EVAPORATOR	2.36E-08	3.33E-02	1.86E-11	5.60E-10	1.65E-09	1.93E-08	1.89E-09	2.41E-09	7.71E-12	5.34E-08
ION EXCHANGE	6.44E-02	2.63E-02	8.38E-06	3.19E-04	4.76E-03	4.18E-03	6.52E-04	2.22E-03	1.23E-05	1.49E-01
HIGH HEAT WASTE	7.00E-01	6.57E-03	2.28E-05	3.47E-03	5.19E-02	4.54E-02	7.07E-03	2.41E-02	1.34E-04	1.62E+00
TOTAL			3.38E-04							

TABLE D-21. Dose and Risk to the Onsite Population from Stack Release due to Transfer Error to Sump

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
			PERSON-REM/YR	DOSE, PERSON-REM						
ONITE POPULATION										
DECONTAMINATION	4.83E-07	3.33E-03	1.92E-11	5.78E-09	3.97E-08	1.01E-07	3.04E-08	2.02E-08	9.65E-11	3.02E-07
DISSOLVER	2.62E-01	3.94E-02	9.89E-05	2.51E-03	3.06E-02	8.46E-02	9.62E-03	1.34E-02	3.15E-05	2.97E-01
HEAD END	2.36E-01	6.57E-03	1.48E-05	2.26E-03	2.76E-02	7.61E-02	8.64E-03	1.20E-02	2.84E-05	2.68E-01
FIRST CYCLE	2.36E-01	1.14E-02	2.57E-05	2.26E-03	2.76E-02	7.61E-02	8.64E-03	1.20E-02	2.84E-05	2.68E-01
2ND U CYCLE	1.91E-11	6.92E-03	1.12E-14	1.62E-12	2.55E-12	6.04E-11	6.17E-12	7.10E-12	2.38E-15	3.53E-11
2ND Pu CYCLE	1.23E-06	2.72E-02	3.58E-07	1.32E-05	3.05E-08	5.23E-04	5.54E-05	5.71E-05	2.90E-11	2.48E-05
BAR EVAPORATOR	7.00E-01	3.33E-02	4.83E-05	1.45E-03	8.25E-02	1.83E-02	3.61E-03	1.31E-02	8.35E-05	7.91E-01
LAN EVAPORATOR	2.36E-08	3.33E-02	1.52E-11	4.57E-10	2.67E-09	1.66E-08	1.81E-09	2.17E-09	3.21E-12	2.64E-08
ION EXCHANGE	6.44E-02	2.63E-02	3.50E-06	1.33E-04	7.58E-03	1.69E-03	3.32E-04	1.20E-03	7.68E-06	7.27E-02
HIGH HEAT WASTE	7.00E-01	6.57E-03	9.53E-06	1.45E-03	8.25E-02	1.83E-02	3.61E-03	1.31E-02	8.35E-05	7.91E-01
TOTAL			2.01E-04							

TABLE D-22. Dose and Risk to the Offsite Population from Stack Release due to Overflow to Sump

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/YR		DOSE, PERSON-REM					
DECONTAMINATION	2.83E-07	4.38E-03	1.60E-10	3.65E-08	1.74E-07	6.79E-07	1.82E-07	1.18E-07	4.75E-10	1.99E-06
DISSOLVER	1.54E-01	4.38E-02	5.83E-04	1.33E-02	1.40E-01	4.37E-01	4.89E-02	7.81E-02	1.55E-04	1.86E+00
HEAD END	1.38E-01	4.38E-02	5.21E-04	1.19E-02	1.25E-01	3.92E-01	4.38E-02	6.28E-02	1.39E-04	1.67E+00
FIRST CYCLE	1.38E-01	2.01E-01	2.40E-03	1.19E-02	1.25E-01	3.92E-01	4.38E-02	6.28E-02	1.39E-04	1.67E+00
2ND W CYCLE	1.12E-11	7.71E-02	8.09E-18	1.05E-16	1.01E-11	3.86E-15	4.79E-16	4.36E-16	1.17E-14	2.96E-16
2ND Pu CYCLE	7.17E-07	2.28E-01	1.50E-05	6.60E-05	1.39E-07	2.63E-03	2.72E-04	2.77E-04	1.43E-10	1.55E-04
WATER EVAPORATOR	4.11E-01	2.19E-01	2.01E-03	9.19E-03	3.76E-01	1.19E-01	2.16E-02	7.72E-02	4.11E-04	4.95E+00
LAR EVAPORATOR	1.39E-08	2.19E-01	5.17E-10	2.36E-09	1.21E-08	8.48E-08	9.05E-09	1.11E-08	1.58E-11	1.65E-07
ION EXCHANGE	2.19E-01	3.50E-01	1.72E-03	4.92E-03	2.02E-01	6.36E-02	1.17E-02	4.13E-02	2.21E-04	2.65E+00
HIGH HEAT WASTE	4.11E-01	3.50E-01	3.22E-03	9.19E-03	3.76E-01	1.19E-01	2.16E-02	7.72E-02	4.11E-04	4.95E+00
TOTAL			1.05E-02							

TABLE D-23. Dose and Risk to the Maximum Individual from Stack Release due to Overflow to Sump

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MRHM/YR							
DECONTAMINATION	2.83E-07	4.38E-03	3.32E-11	7.57E-09	1.39E-08	1.48E-07	3.44E-08	2.12E-08	1.35E-10	3.74E-07
DISSOLVER	1.54E-01	4.38E-02	8.85E-05	2.02E-03	1.12E-02	6.10E-02	6.38E-03	9.90E-03	4.44E-05	3.53E-01
HEAD END	1.38E-01	4.38E-02	7.93E-05	1.81E-03	1.00E-02	5.47E-02	5.71E-03	8.87E-03	3.98E-05	3.17E-01
FIRST CYCLE	1.38E-01	2.01E-01	3.65E-04	1.81E-03	1.00E-02	5.47E-02	5.71E-03	8.87E-03	3.98E-05	3.17E-01
2ND CYCLE	1.12E-11	7.71E-02	9.79E-19	1.27E-17	8.20E-13	4.74E-16	5.56E-17	4.75E-17	3.35E-15	5.68E-17
2ND CYCLE	7.17E-07	2.28E-01	1.91E-06	8.38E-06	1.11E-08	3.43E-04	3.22E-05	3.14E-05	4.09E-11	2.98E-05
RAIN EVAPORATOR	4.11E-01	2.19E-01	4.47E-04	2.04E-03	3.04E-02	2.67E-02	4.15E-03	1.42E-02	7.86E-05	9.54E-01
LAR EVAPORATOR	1.39E-08	2.19E-01	7.23E-11	3.30E-10	9.69E-10	1.14E-08	1.12E-09	1.42E-09	4.53E-12	3.13E-08
ION EXCHANGE	2.19E-01	3.50E-01	3.82E-04	1.09E-03	1.63E-02	1.43E-02	2.25E-03	7.58E-03	4.22E-05	5.10E-01
HIGH HEAT WASTE	4.11E-01	3.50E-01	7.15E-04	2.04E-03	3.04E-02	2.67E-02	4.15E-03	1.42E-02	7.86E-05	9.54E-01
TOTAL			2.08E-03							

TABLE D-24. Dose and Risk to the Onsite Population from Stack Release due to Overflow to Sump

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
			PERSON-REM/YR	DOSE, PERSON-REM						
ONSITE POPULATION										
DECONTAMINATION	2.83E-07	4.38E-03	1.52E-11	3.48E-09	2.25E-08	6.23E-08	1.78E-08	1.18E-08	5.66E-11	1.89E-07
DISSOLVER	1.54E-01	4.38E-02	6.44E-05	1.47E-03	1.80E-02	4.97E-02	5.65E-03	7.85E-03	1.05E-05	1.74E-01
HEAD END	1.38E-01	4.38E-02	5.78E-05	1.32E-03	1.61E-02	4.45E-02	5.06E-03	7.03E-03	1.66E-05	1.56E-01
FIRST CYCLE	1.38E-01	2.01E-01	2.66E-04	1.32E-03	1.61E-02	4.45E-02	5.06E-03	7.03E-03	1.66E-05	1.56E-01
2ND U CYCLE	1.12E-11	7.71E-02	9.64E-19	1.25E-17	1.31E-12	4.57E-16	5.74E-17	5.31E-17	1.39E-15	2.76E-17
2ND Pu CYCLE	7.17E-07	2.28E-01	1.76E-06	7.73E-06	1.78E-08	3.06E-04	3.24E-05	3.34E-05	1.70E-11	1.45E-05
HAN EVAPORATOR	4.11E-01	2.19E-01	1.86E-04	8.48E-04	4.84E-02	1.08E-02	2.12E-03	7.68E-03	4.89E-05	4.64E-01
LAR EVAPORATOR	1.39E-08	2.19E-01	5.89E-11	2.69E-10	1.56E-09	9.77E-09	1.07E-09	1.28E-09	1.89E-12	1.55E-08
ION EXCHANGE	2.19E-01	3.50E-01	1.59E-04	4.55E-04	2.60E-02	5.77E-03	1.15E-03	4.10E-03	2.63E-05	2.48E-01
HIGH HEAT WASTE	4.11E-01	3.50E-01	2.97E-04	8.48E-04	4.84E-02	1.08E-02	2.12E-03	7.68E-03	4.89E-05	4.64E-01
TOTAL			1.03E-03							

TABLE D-25. Dose and Risk to the Offsite Population from Stack Release due to Leak to Sump

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION	PERSON-REM/YR			DOSE, PERSON-REM						
DECONTAMINATION	8.86E-08	2.19E+00	2.50E-08	1.14E-08	5.46E-08	2.13E-07	5.71E-08	3.71E-08	1.49E-10	6.23E-07
DISSOLVER	2.18E-02	1.14E+00	2.14E-03	1.88E-03	1.97E-02	6.19E-02	6.92E-03	9.92E-03	2.21E-05	2.63E-01
HEAD END	1.93E-02	7.01E+00	1.17E-02	1.67E-03	1.75E-02	5.49E-02	6.13E-03	8.88E-03	1.95E-05	2.33E-01
FIRST CYCLE	1.93E-02	1.14E+01	1.90E-02	1.67E-03	1.75E-02	5.49E-02	6.13E-03	8.88E-03	1.95E-05	2.33E-01
2ND & CYCLE	1.56E-12	3.07E+00	3.56E-12	1.16E-12	1.42E-12	4.29E-11	4.25E-12	4.95E-12	1.63E-15	2.98E-11
2ND P ₄ CYCLE	1.00E-07	9.64E+00	8.92E-05	9.26E-06	1.95E-08	3.69E-04	3.82E-05	3.89E-05	2.00E-11	2.18E-05
HEAT EVAPORATOR	5.82E-02	1.58E+01	2.05E-02	1.30E-03	5.32E-02	1.68E-02	3.06E-03	1.09E-02	5.83E-05	7.01E-01
LAM EVAPORATOR	1.97E-09	1.58E+01	5.27E-09	3.34E-10	1.72E-09	1.20E-08	1.28E-09	1.56E-09	2.24E-12	2.34E-08
ION EXCHANGE	7.63E-02	7.88E+00	1.35E-02	1.71E-03	6.98E-02	2.21E-02	4.03E-03	1.43E-02	7.65E-05	9.19E-01
HIGH HEAT WASTE	5.82E-02	8.76E+00	1.14E-02	1.30E-03	5.32E-02	1.68E-02	3.06E-03	1.09E-02	5.83E-05	7.01E-01
TOTAL			7.83E-02							

TABLE D-26. Dose and Risk to the Maximum Individual from Stack Release due to Leak to Sump

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR		DOSE, MREM					
DECONTAMINATION	8.86E-08	2.19E+00	5.21E-09	2.38E-09	4.34E-09	4.64E-08	1.08E-08	6.65E-09	4.23E-11	1.17E-07
DISSOLVER	2.18E-02	1.14E+00	3.26E-04	2.86E-04	1.58E-03	8.64E-03	9.03E-04	1.48E-03	6.32E-06	4.99E-02
HEAD END	1.93E-02	7.01E+00	1.77E-03	2.53E-04	1.40E-03	7.66E-03	8.01E-04	1.24E-03	5.58E-06	4.43E-02
FIRST CYCLE	1.93E-02	1.14E+01	2.88E-03	2.53E-04	1.40E-03	7.66E-03	8.01E-04	1.24E-03	5.58E-06	4.43E-02
2ND U CYCLE	1.56E-12	3.07E+00	4.97E-13	1.62E-13	1.14E-13	5.80E-12	5.06E-13	6.05E-13	4.68E-16	5.68E-12
2ND P _u CYCLE	1.00E-07	9.64E+00	1.14E-05	1.18E-06	1.56E-09	4.80E-05	4.52E-06	4.48E-06	5.73E-12	4.17E-06
RAW EVAPORATOR	5.82E-02	1.58E+01	4.54E-03	2.88E-04	4.31E-03	3.78E-03	5.89E-04	2.01E-03	1.12E-05	1.35E-01
LAN EVAPORATOR	1.97E-09	1.58E+01	7.35E-10	4.66E-11	1.37E-10	1.61E-09	1.58E-10	2.01E-10	6.41E-13	4.44E-09
ION EXCHANGE	7.63E-02	7.88E+00	2.98E-03	3.78E-04	5.65E-03	4.96E-03	7.74E-04	2.63E-03	1.46E-05	1.77E-01
HIGH HEAT WASTE	5.82E-02	8.76E+00	2.52E-03	2.88E-04	4.31E-03	3.78E-03	5.89E-04	2.01E-03	1.12E-05	1.35E-01
TOTAL			1.50E-02							

TABLE D-27. Dose and Risk to the Onsite Population from Stack Release due to Leak to Sump

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
ONSITE POPULATION			PERSON-REM/YR		DOSE, PERSON-REM					
DECONTAMINATION	8.86E-08	2.19E+00	2.39E-09	1.09E-09	7.06E-09	1.95E-08	5.59E-09	3.71E-09	1.77E-11	5.90E-08
DISSOLVER	2.18E-02	1.14E+00	2.38E-04	2.09E-04	2.54E-03	7.04E-03	8.00E-04	1.11E-03	2.63E-06	2.47E-02
HEAD END	1.93E-02	7.01E+00	1.30E-03	1.85E-04	2.26E-03	6.24E-03	7.09E-04	9.85E-04	2.32E-06	2.19E-02
FIRST CYCLE	1.93E-02	1.14E+01	2.11E-03	1.85E-04	2.26E-03	6.24E-03	7.09E-04	9.85E-04	2.32E-06	2.19E-02
2ND U CYCLE	1.56E-12	3.07E+00	4.05E-13	1.32E-13	1.83E-13	4.93E-12	5.06E-13	5.82E-13	1.95E-16	2.79E-12
2ND Pu CYCLE	1.00E-07	9.64E+00	1.04E-05	1.08E-06	2.50E-09	4.29E-05	4.55E-06	4.68E-06	2.39E-12	2.04E-06
BAR EVAPORATOR	5.82E-02	1.58E+01	1.89E-03	1.20E-04	6.86E-03	1.53E-03	3.00E-04	1.09E-03	6.95E-06	6.58E-02
LAN EVAPORATOR	1.97E-09	1.58E+01	6.01E-10	3.81E-11	2.22E-10	1.38E-09	1.51E-10	1.81E-10	2.67E-13	2.20E-09
ION EXCHANGE	7.63E-02	7.88E+00	1.25E-03	1.58E-04	8.99E-03	2.00E-03	3.95E-04	1.43E-03	9.11E-06	8.62E-02
HIGH HEAT WASTE	5.82E-02	8.76E+00	1.05E-03	1.20E-04	6.86E-03	1.53E-03	3.00E-04	1.09E-03	6.95E-06	6.58E-02
TOTAL			7.84E-03							

TABLE D-28. Dose and Risk from Stack Release due to Processing Short Cooled Fuels and Ruthenium Volatilization

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
OFFSITE POPULATION			PERSON-REM/YR	DOSE, PERSON-REM						
DISSOLVER-SHORT COOLED FUELS	1.65E+00	3.50E-02	2.72E-03	7.75E-02	2.06E-02	1.07E-01	1.35E-01	2.26E-01	4.36E+01	2.65E-03
DISSOLVER-Ru VOLATILIZATION	7.00E+00	5.26E-02	8.51E-03	1.62E-01	1.16E+01	1.29E+00		2.71E+00		1.64E+02
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR	DOSE, MREM						
DISSOLVER-SHORT COOLED FUELS	1.65E+00	3.50E-02	5.12E-04	1.46E-02	1.99E-03	2.54E-02	2.54E-02	4.13E-02	8.40E+00	5.21E-04
DISSOLVER-Ru VOLATILIZATION	7.00E+00	5.26E-02	1.94E-03	3.69E-02	9.36E-01	2.97E-01		4.01E-01		3.13E+01
ON-SITE POPULATION			PERSON-REM/YR	DOSE, PERSON-REM						
DISSOLVER-SHORT COOLED FUELS	1.65E+00	3.50E-02	2.74E-04	7.83E-03	2.62E-03	9.56E-03	1.34E-02	2.28E-02	4.38E+00	3.12E-04
DISSOLVER-Ru VOLATILIZATION	7.00E+00	5.26E-02	7.52E-04	1.43E-02	1.49E+00	1.13E-01		2.19E-01		1.53E+01

D-29

TABLE D-29. Dose and Risk to the Offsite Population from Liquid Releases to Four Mile Creek due to Coil and Tube Failure

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG	SKIN
OFFSITE POPULATION	PERSON-REM/YR			DOSE, PERSON-REM							
DECONTAMINATION	2.00E-01	5.26E-04	5.12E-06	9.75E-03	7.59E-02	4.67E-02	1.58E-02	5.46E-03	2.65E-05	1.82E-03	2.40E-05
DISSOLVER	1.69E+01	2.52E-03	2.09E-03	8.30E-01	6.46E+00	4.58E+00	1.35E+00	4.64E-01	2.25E-03	1.55E-01	2.04E-03
HEAD END	1.69E+01	2.94E-03	2.44E-03	8.30E-01	6.46E+00	4.58E+00	1.35E+00	4.64E-01	2.25E-03	1.55E-01	2.04E-03
FIRST CYCLE	1.69E+01	1.87E-03	1.55E-03	8.30E-01	6.46E+00	4.58E+00	1.35E+00	4.64E-01	2.25E-03	1.55E-01	2.04E-03
2ND U CYCLE	1.37E-09	1.87E-03	1.26E-13	6.73E-11	5.24E-10	3.72E-10	1.09E-10	3.76E-11	1.83E-13	1.25E-11	1.65E-13
2ND Pu CYCLE	8.80E-05	1.87E-03	8.06E-09	4.32E-06	3.37E-05	2.39E-05	7.02E-06	2.42E-06	1.17E-08	8.05E-07	1.06E-08
HAN EVAPORATOR	1.72E+01	1.37E-02	1.15E-02	8.41E-01	6.55E+00	4.66E+00	1.36E+00	4.69E-01	2.28E-03	1.57E-01	2.06E-03
LAN EVAPORATOR	5.84E-07	1.37E-02	3.90E-10	2.86E-08	2.22E-07	1.58E-07	4.62E-08	1.59E-08	7.73E-11	5.32E-09	7.01E-11
ION EXCHANGE	1.72E+01	8.41E-04	7.07E-04	8.41E-01	6.55E+00	4.66E+00	1.36E+00	4.69E-01	2.28E-03	1.57E-01	2.06E-03
HIGH HEAT WASTE	1.72E+01	2.94E-03	2.48E-03	8.41E-01	6.55E+00	4.66E+00	1.36E+00	4.69E-01	2.28E-03	1.57E-01	2.06E-03
TOTAL			2.08E-02								

D-30

DPSTSA-200-10, SUP-4

TABLE D-30. Dose and Risk to the Maximum Individual from Liquid Releases to Four Mile Creek due to Coil and Tube Failure

UNIT OPERATION	TOTAL CI RELEASED	FREQUENCY, /YR	RISK	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG	SKIN
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR		DOSE, MREM						
DECONTAMINATION	2.00E-01	5.26E-04	4.56E-07	8.68E-04	7.15E-03	9.62E-03	4.54E-03	1.48E-03	5.07E-06	5.30E-04	3.09E-06
DISSOLVER	1.69E+01	2.52E-03	1.86E-04	7.38E-02	6.09E-01	9.44E-01	3.87E-01	1.26E-01	4.32E-04	4.51E-02	2.63E-04
HEAD END	1.69E+01	2.94E-03	2.17E-04	7.38E-02	6.09E-01	9.44E-01	3.87E-01	1.26E-01	4.32E-04	4.51E-02	2.63E-04
FIRST CYCLE	1.69E+01	1.87E-03	1.38E-04	7.38E-02	6.09E-01	9.44E-01	3.87E-01	1.26E-01	4.32E-04	4.51E-02	2.63E-04
2ND U CYCLE	1.37E-09	1.87E-03	1.12E-14	5.99E-12	4.94E-11	7.65E-11	3.13E-11	1.02E-11	3.50E-14	3.66E-12	2.14E-14
2ND Pu CYCLE	8.80E-05	1.87E-03	7.17E-10	3.85E-07	3.17E-06	4.92E-06	2.01E-06	6.58E-07	2.25E-09	2.35E-07	1.37E-09
HAN EVAPORATOR	1.72E+01	1.37E-02	1.02E-03	7.48E-02	6.18E-01	9.56E-01	3.91E-01	1.28E-01	4.40E-04	4.57E-02	2.66E-04
LAN EVAPORATOR	5.84E-07	1.37E-02	3.47E-11	2.54E-09	2.10E-08	3.24E-08	1.33E-08	4.36E-09	1.50E-11	1.55E-09	9.03E-12
ION EXCHANGE	1.72E+01	8.41E-04	6.29E-05	7.48E-02	6.18E-01	9.56E-01	3.91E-01	1.28E-01	4.40E-04	4.57E-02	2.66E-04
HIGH HEAT WASTE	1.72E+01	2.94E-03	2.20E-04	7.48E-02	6.18E-01	9.56E-01	3.91E-01	1.28E-01	4.40E-04	4.57E-02	2.66E-04
TOTAL			1.85E-03								

APPENDIX E

RESPONSE TO SROO COMMENTS AND NON-NUCLEAR HAZARDS ADDENDUM

**RESPONSE TO SROO COMMENTS
CANYON SAFETY ANALYSIS REPORTS (SARs)
DPSTSA-200-10, SUP-4**

NUCLEAR SAFETY BRANCH

1. **Comment:** SAR references to DPSTS NIM-78 in Sections 3.0 and 4.0 should be updated to DPSTS NIM-85.

Response: Agreed, change references to DPSTS-NIM-78 in Sections 3.0 and 4.0 to DPSTS-NIM-85.

2. **Comment:** SAR Section 3.0 - SAR should list by number the safety guides, codes, standards and orders applied for the design of the Canyon facilities.

Response: On Page 3-1, Section 3.1, DuPont design standards have been compared to DOE requirements. The F- and H-Canyon have been in operation for approximately 30 years, and identification of the original design criteria would be a difficult and time-consuming task. However, DuPont is selecting a contractor who will locate all codes, safety guides, etc., for all new facilities and will attempt to compare these to the older facilities.

3. **Comment:** SAR Section 3.0 - SAR should identify the design basis events for the Canyon facilities.

Response: The design basis events are described on Pages 2-8 and 2-13, and 5-3 for a tornado and earthquake, respectively. Consistent with the requirements of the NRC format, Section 3.0 of the next updated version of the SAR's will be revised to include the design basis events.

4. **Comment:** SAR Section 3.0 - SAR should compare the Canyon facility design to the current design requirements of DOE Order 6430.1 and identify and justify any deviations from the requirements.

Response: On Page 3-1, Section 3.1, DuPont Design Standards have been compared to DOE Requirements. Page 5-3 of the F- and H-Canyon SARs reference the DOE Order 6430.1. Consistent with the requirements of the NRC format, Section 3.0 of the next updated version of the SAR's will be revised to include a section wherein the Canyon Facility design is compared to the current design requirements of DOE Order 6430.1 and any deviations are identified and justified.

5. **Comment:** SAR Section 3.0 - SAR should provide a definition for or identify the document which defines Maximum Resistant Structures and Class I Explosion-proof Structures.

**RESPONSE TO SROO COMMENTS
CANYON SAFETY ANALYSIS REPORTS (SARs)
DPSTSA-200-10, SUP-4**

NUCLEAR SAFETY BRANCH

1. **Comment:** SAR references to DPSTS NIM-78 in Sections 3.0 and 4.0 should be updated to DPSTS NIM-85.

Response: Agreed, change references to DPSTS-NIM-78 in Sections 3.0 and 4.0 to DPSTS-NIM-85.

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Response: On Page 3-1, Section 3.1, DuPont design standards have been compared to DOE requirements. The F- and H-Canyon have been in operation for approximately 30 years, and identification of the original design criteria would be a difficult and time-consuming task. However, DuPont is selecting a contractor who will locate all codes, safety guides, etc., for all new facilities and will attempt to compare these to the older facilities.

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4. **Comment:** SAR Section 3.0 - SAR should compare the Canyon facility design to the current design requirements of DOE Order 6430.1 and identify and justify any deviations from the requirements.

Response: On Page 3-1, Section 3.1, DuPont Design Standards have been compared to DOE Requirements. Page 5-3 of the F- and H-Canyon SARs reference the DOE Order 6430.1. Consistent with the requirements of the NRC format, Section 3.0 of the next updated version of the SAR's will be revised to include a section wherein the Canyon Facility design is compared to the current design requirements of DOE Order 6430.1 and any deviations are identified and justified.

5. **Comment:** SAR Section 3.0 - SAR should provide a definition for or identify the document which defines Maximum Resistant Structures and Class I Explosion-proof Structures.

Response: A Class I structure is a blast resistant structure as defined in DuPont Specification No. 3580, originally issued 2/7/58. A Class I explosion proof Maximum Resistance Construction (MRC) structure is designed to withstand pressures of 1000 lb/ft². Consistent with the requirements of the NRC format, Section 3.0 of the next updated version of the SAR's will be revised to include documented, clear definitions of "Maximum Resistant Structures" and of "Class I Explosion-proof Structures".

6. **Comment:** SAR Section 3.2.2 - SAR should address thru-puts.

Response: Thru-puts of the Facilities in Section 3.2.2 will be made available if required, but were not included to avoid access limitations (i.e., classification).

7. **Comment:** SAR Section 3.2.2 - SAR should include wind, tornado and earthquake hazard design considerations for the process systems and components.

Response: As noted in the response to Comment 3, Wind, Tornado and Earthquake design considerations are included in Sections 2.0 and 5.0. In addition, DuPont has established a task force that will, with the aid of a consultant, compare the design standards, orders and requirements of the DOE, DuPont and NRC with the existing design and make appropriate recommendations. This work will be completed in approximately one year. The results of that study will be provided to DOE once available and the results will also be included in the next update of the facility SAR to reflect the wind, tornado and earthquake hazard design considerations for the process systems and components.

8. **Comment:** F-Canyon SAR Section 3.3.1.2.1 - SAR indicates that both the 294-1F and the 294-F sandfilters were constructed in 1976.

Response: Agreed, F-Canyon SAR Section 3.3.1.2.1 will be revised version to reflect clearly that the two sand filters (294-F and 294-1F) were constructed sequentially but normally operate in parallel.

9. **Comment:** F-Canyon SAR Section 3.2.2.6.3 and Section 3.3 - Sections indicated that things will happen in 1986. These should be updated.

Response: Agreed, F-Canyon SAR Sections 3.2.2.6.3 and 3.3 will be revised to reflect the status consistent with the date of the SAR.

10. **Comment:** SAR Section 4.4 - SARs should address training for Separations Technology personnel as well as the Separations and Separations Works Engineering personnel.

Response: Agreed, Section 4.4 will be expanded in the next updated version of the SARs to include the training provided Separations Technology Personnel.

11. **Comment:** SAR Section 5.0 - SAR should address and, if applicable, analyze the potential for fire or uncontrolled reactions to lead to a criticality event.

Response: No fire that would lead to a criticality event was identified. Based on an analysis of the second Pu cycle (in preparation), uncontrolled reaction that can lead to a criticality in the sump is approximately 2% of all criticalities that could potentially occur.

12. **Comment:** SAR and Canyon Systems Analyses (SYAs) - Inconsistencies exist in the frequencies for criticality events between Table AC-1, Chart C and text in the SYAs and Figure B-5 and Table 5-7 in the SARs. For example: Table 5-7 in SUP-4 indicates that the criticality frequency in Head End is $1.7E-9$ but Figure B-5 in SUP-4 indicates $2.0E-9$. These inconsistencies are not significant but detract from the quality of the document and should be corrected.

Response: As the analyses proceeded, no effort was made to update the SYAs. The SARs will be reviewed for consistency with the Systems Analyses and the inconsistencies removed before the next updated version of the SARs are issued.

13. **Comment:** SARs and SYAs - Documents should take into account common mode failures - Example: An earthquake causing an operator error and instrument failure (See Chart C2.2 and C3.2 in SYA for F-Canyon and C2 and C31 in SYA for H-Canyon) or earthquake causing ventilation failure.

Response: The documents account for common cause failure in several ways. For most of the releases associated with the canyons, maximum releases occur when some initiating event causes a direct or indirect release to the canyon sumps, the exhaust fans are functional, and the sand filter is in a partially degraded state (that is, the efficiency is less than average). The only mechanism identified that could have the common effect of releasing material to the sump and of disabling the fan system is a seismic event. Seismic events are assumed to fail the fan system which is conservative relative to assuming that the fans function and is therefore the proper assumption in that the primary motive force for removing the canyon atmosphere to the environment is non-existent.

Process events affording highest risk are primarily from the data bank which inherently includes those combinations that contribute to the initiator, both qualitative and quantitative. Although no common mechanism exists that could initiate a release and degrade the filter efficiency, it is always assumed that the sand filter is degraded to the 97th percentile of worst measure efficiencies for whatever reason.

In the special case cited by DOE-OSR that involved a nuclear criticality accident as a result of a loss of material to the sump, failure of the level detector, concentration by evaporation, and continued operation of the facility, the question was raised as to why an earthquake would not cause both the release and failure of the sump instruments. This is a plausible scenario; however, the dominant cut set for the fault tree branch in question is that of a leak at a frequency of 6×10^{-5} per hour, an instrument unavailability of 5×10^{-3} , a human error probability of 1×10^{-1} , and a probability of continuing to operate of 0.5. The frequency for this dominant cut set is 2×10^{-8} per hour. If the proposed scenario were specifically isolated, the frequency of the seismic event would be 2×10^{-8} per hour and the unavailability of the instrument would be 1.0. Because the recovery time is quite long relative to the stress level effects on human error performance, the error demand remains at 1×10^{-1} . The probability that canyon operations would continue, however, would be dramatically reduced and is assumed to be 1×10^{-3} per day. Overall, the sequence frequency would be 2×10^{-12} per hour. Because this proposed sequence is about four orders of magnitude less than the dominant cut set, it may be ignored without affecting the frequency for all viable cut sets combined.

14. Comment: F-Canyon SYA Chart 2.2 - "Transfer Errors" and "Uncontrolled Reactions" should be included as potential contributors to the event "Dilute Fissiles in Sump".

Response: Agreed, F-Canyon System Analysis Chart C-2.2 (Volume 2, Page AC-20) will be revised in the next updated version to include "Uncontrolled Reactions" and "Transfer Errors" as potential contributors to the event "Dilute Fissiles in the Sump".

15. Comment: H-Canyon SYA Table AC-4 - The frequency for the event L01ASFSC is not listed in the Table.

Response: Agreed, H-Canyon System Analyses Table AC-4 will be revised in the next updated version to include the frequency of the event L01ASFSC (from Chart C-15).

16. Comment: SYAs Table AC-4 - The difference in events labeled HUMFAIL which indicate one event with a probability of 0.1 and another with a probability of 0.01 should be explained.

Response: The various events labelled "HUMFAIL-1, HUMFAIL-2, ... HUMFAIL-9 appearing in Tables AC-2 through A-4 of the H-Canyon SYA each refer to specific types of human errors that are defined in the tables under the column headed "Event Name"; as such, each has a unique frequency or probability of occurrence. The probability values for human error 1 and 3 through 8 were taken from the reference (WASH-1400) while those for errors 2 and 9 were derived from the 200-Area Fault Tree Data Base.

17. Comment: SAR Table 5-7, Section 6.1 and Section 5.3.4.3 - SAR Section 6.1 indicates that criticality can occur in many different operations in the Canyons. All of the operations identified should be included in the frequency determination for a criticality event and included in Table 5-7 and/or discussed in Section 5.3.4.3.

Response: The dominant events have been included in Table 5-7. If all other events need to be included, Table 5-7 will be reviewed and revised in the updated version of the SARs to include frequencies of all the operations identified in Section 6.1 as having potential for the occurrence of a criticality event.

18. Comment: F-Canyon SAR Section 5.3.4.3 - SAR should include a discussion and, if applicable an analysis of the criticality potentials in the Evaporators and in the product storage tanks.

Response: There is no specific analysis for criticality potential for the evaporators of the F-Canyon. Approximately 12% of all criticality events are calculated to occur in the storage tanks. The likelihood of accumulation of Pu and the potential for criticality in the evaporator feed tanks would be even lower. These tanks are routinely sampled and analyzed so that sample analysis would warn of any problems in the evaporator feed (see discussion on top of page 3-32 in F-Canyon SAR). Consistent with the requirements of the NRC format, Section 5.3.4.3 of the updated version of the F-Canyon SAR will be expanded to include discussion of the criticality potentials in the Evaporators and in the Product storage tanks.

19. Comment: H-Canyon SAR Section 5.3.4.3 - SAR should include a discussion and, if applicable, an analysis of the criticality potentials in the storage of Pu-239 in tanks 12.1 and 8.5 and in product storage tanks.

Response: Based on an analogy with the F-Canyon, the criticality potential is estimated to be 1 in 600,000 years. Section 5.3.4.3 of the updated version of the H-Canyon SAR will include a discussion and, if applicable, results of an analysis of the criticality potentials of canyon tanks 12.1 and 8.5 and the storage tanks all handling aqueous Pu-239 product solution.

20. Comment: H-Canyon SAR - Table 5-7 indicates that solvent extraction operations are the most significant when considering criticality. Reference 39 (SAIC 85/3075) states that the major contributor to criticality risk is in the evaporation process. The SYA indicates that the evaporation process is the major contributor to criticality risk. SAR addresses only the Head End evaporator in the criticality frequency determination. The rerun and lCU evaporators should also be considered. SYA indicates that the criticality frequency is approximately $5.8E-7$ and SAR indicates $3.7E-7$. These inconsistencies should be corrected.

Response: The criticality calculations in the H-Canyon are weak, but the numbers are conservative. A re-analysis of the sump is in progress. The differences between the criticality frequencies between the existing H-Canyon SYA and SAR will be resolved in the updated version of the SAR. Also criticality potentials of the rerun and ICU evaporators will be considered and if applicable an analysis made. Results of these will be included in the updated version of the SAR.

21. **Comment:** SAR Section 5.4.1.1 - SAR should address seismic and wind/tornado damage to stacks and to the covers over the air tunnels to the sand filters and if applicable calculate the risks involved.

Response: Agreed, consistent with the requirements of the NRC format, seismic and wind/tornado damage to the stacks and to the sand filter air tunnel covers will be addressed in SAR Section 5.4.1.1 and, if applicable, the risks calculated and included in the updated version of the SAR.

22. **Comment:** SAR Section 5.3.1.2 - SAR implies that all the material will be lost to the secondary confinement. Section 5.4.1.2 indicates that only half the material will be lost. SAR should be revised to clarify.

Response: Agreed, the discrepancy between SAR Sections 5.3.1.2 and 5.4.1.2 will be resolved by noting in 5.3.1.2 that a significant (~50%) fraction of the material will be released to the secondary confinement.

23. **Comment:** SAR Section 5.4.9.6 - Sections should refer to the appropriate references (Ref. 40 in SUP-5 and Ref. 38 in SUP-4).

Response: Agreed, Section 5.4.9.6 will be revised to include the Reference 38 in the F-Canyon SAR and Reference 40 in the H-Canyon SAR.

RADIATION SAFETY

Design Features

1. Comment: Provide specific examples illustrating how the ALARA concept was applied to the facility design.

Response: Radiation and Contamination control are discussed in Section 4.9.1. Consistent with the requirements of the NRC format, Section 3.0 text of the updated version of the SARs will be revised to show specific examples of when the ALARA concept was applied to the facility design as it exists today.

2. Comment: Include radiation zone maps.

Response: Radiation zones are discussed in Section 4.9.1. Consistent with the requirements of the NRC format, radiation zone maps of the facility will be included in the updated version of the SARs.

3. Comment: Describe the methods employed to determine shielding arrangements and thicknesses.

Response: Consistent with the requirements of the NRC format, the methods to determine shielding arrangements and thicknesses will be summarized in the updated version of the SARs and include a reference to the appropriate document detailing the methods.

4. Comment: Deleted.

Response: Not required.

5. Comment: Show the location of radiation safety-related facilities such as decontamination facilities and health physics laboratories and offices.

Response: The description of the radiation-safety related facilities in the Canyon SARs will be expanded in the updated version of Section 3 to identify these and include maps to show the locations and functions where appropriate.

6. Comment: Provide information on the types, location range, and number of installed radiation monitors.

Response: Tables and location maps will be included in Section 3.0 of the updated version of the Canyon SARs that list types, ranges and location of the radiation monitors installed in the canyon facilities.

7. **Comment:** Describe specific contamination control and decontamination facilities for areas with the facilities other than the hot or warm canyons.

Response: Consistent with the requirements of the NRC format, the canyon facilities will be reexamined to identify areas dedicated to contamination control and decontamination activities conducted outside the hot and warm canyons. Descriptions of these will be included in Section 3 of the updated version of the SARs.

Operating Features

1. **Comment:** Describe the radiation safety training provided to radiation workers and to Health Protection personnel.

Response: The NRC format requires Radiation Safety to be addressed in a more complete fashion. Section 4.4 of the updated version of the SARs will be expanded to include a description of the training of Health Protection personnel.

2. **Comment:** Provide information on the location, number and types of health protection radiation safety instrumentation available at these facilities.

Response: Consistent with the requirements of the NRC format, the updated version of the SARs will include Tables and in maps the numbers, types, and locations of health protection radiation safety instrumentation that is available in the canyon facilities. (Note that this effort includes/parallels that of item 5 of the Design Features above).

3. **Comment:** Describe methods to control access to radiation areas and methods for contamination control.

Response: Consistent with the requirements of the NRC format, the updated version of the Canyon Facility SARs will be revised to include the program of controlling personnel access to radiation areas and the method of contamination control.

4. **Comment:** Identify applicable radiation safety procedures.

Response: Radiation control procedures as applied in the canyon facilities will be summarized consistent with the requirements of the NRC format for the updated version of the SARs. It has not been DuPont's practice to reference the specific procedure numbers from the Safety Manual, DPSOP/DPSOL and Health Protection Manual.

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5. **Comment:** Describe the methods and facilities for the testing and calibration for all radiation monitoring instruments.

Response: The testing/calibration program for all radiation monitoring instrumentation will be summarized consistent with the requirements of the NRC format for the updated version of the Canyon Facility SARs.

6. **Comment:** Provide information on the occupancy factors, dose rates, and airborne concentrations, for normal or routine operations.

Response: Consistent with the requirements of the NRC format, the updated version of the Canyon Facility SARs will be expanded to include more detail regarding occupancy, dose rates and concentrations of airborne contaminants.

7. **Comment:** Provide information on the number and type of internal assimilations of radionuclides.

Response: It appears that the number and type of radionuclides relevant to the Canyon Facility SARs are currently identified and the critically affected organs pointed out. Principal pathways are also described and discussed. The assimilation will be addressed in references to several documents dedicated to this subject.

8. **Comment:** Indicate the availability of emergency power to those installed radiation monitoring instruments which are determined to be key radiation monitors.

Response: Table A-1 will be expanded in the updated version of the SARs to include the specific sources of emergency power for those installed radiation monitoring instruments having a key radiation monitoring function.

9. **Comment:** The frequency of contamination and high exposure incidents (once per week and once per two to three weeks, respectively) in the sample aisles appears excessive. A review of the sample system arrangement of sampling procedures is recommended (Reference Section 5.3.8.5 of each SAR). Crane and gang valve corridor failures and potential contamination events also appear excessive.

Response: Reviews of the sampling system in the canyon facilities and the radiation exposure experience with the current systems are continuing. There are current development programs aimed at converting the existing sampling system into a remotely operated system.

10. **Comment:** There appears to be no automatic isolation of clean areas ventilation exhaust system or "hot" areas supply system in the event of loss of "hot" areas ventilation exhaust. However, pressure differential

sensors are provided and it is assumed that these would alarm in the control room where appropriate prompt action would be taken by operators to prevent significant airborne hazards.

Response: This comment is an observation that no automatic isolation of "clean" area ventilation exhaust systems or "hot" area in the event of a loss of "hot" area exhaust system, but pressure difference control instrumentation are provided.. The description of the ventilation system and the instrumentation that controls the magnitude and direction of flow of air thru the facility is given in Section 3.3.1. It should be noted that the redundancy of blowers and the availability of emergency backup power to all exhaust system blowers provides assurance that the hot and warm areas are maintained at subatmospheric levels regardless. Further, augmenting maintenance of the flow direction from the outside to the clean areas to potentially warm/contaminated areas to hot/contaminated areas to the cleanup system, the supply air system blowers are cut off automatically if the pressure differential sensors detect a lower threshold value. The ducting system also features automatic dampers that act to minimize and/or prevent reverse flows in the ducts. To emphasize these safety related items, a summary paragraph covering the system design safety features and the implementing instrumentation will be included in the updated version of the Canyon Facility SARs.

INDUSTRIAL HYGIENE

1. **Comment:** 1.1 Introduction. "In this report, risk is defined as the expected frequency of an accident, multiplied by the radiological consequence in person-rem." This limits the SAR to radiological hazards only and as defined would not be applicable to safety and industrial hygiene hazards. This section should be revised to reflect risk from all types of hazards.

Response: The definition of risk will be revised in the updated SARs to reflect non-radiological hazards.

2. **Comment:** "Routine conditions were not considered in this analysis other than for comparative purposes." Merely because a hazard exists on a routine daily or periodic basis does not mean that it can be disregarded. The SAR is to identify all hazards and risks either as a result of an accident or routine operating conditions. Additional facts are needed to support the assumption that routine conditions are actually without hazard and risk.

Response: As noted in the April 8, 1987 letter from R.L. Morgan to J.T. Granaghan, the updated SARs will identify all hazards and risks under both normal and accident conditions in arriving at a conclusion as to whether it can be operated without hazard and risk.

3. **Comment:** 1.2.2 Facility Risk. "The very low values for radiological consequences and for risk as a result of F-Canyon operations indicate that the facility can be operated without undue risk to the public or operating personnel." This statement incorrectly assumes that radiological risks are the only undue risks for which an SAR is necessary. Section 1.2.2 Approach stated "The F-Canyon operations were evaluated on the basis of three considerations: 1) potential radiological hazards, 2) potential chemical toxicity hazards, and 3) potential conditions uniquely different from normal industrial practice." This section is 1/3 of what is needed.

Response: As stated in the April 8, 1987 letter from R.L. Morgan to J.T. Granaghan, updated F- and H-Canyon Facilities SARs will include analysis for all hazards and risks to operating personnel and the public at large including radiological and nonradiological.

4. **Comment:** 3.2.2 Process Design Consideration. Although receipt, storage and handling of radioactive materials is discussed, no mention is made of how nonradioactive materials are received, stored and handled. As most canyon processes are closed and shielded, process chemical preparation and addition are the most likely sources of potential exposures. This section needs to be revised to correct this deficiency.

Response: The Process Design Consideration Section 3.2.2 of the updated

version of the F- and H-Canyon Facility SARs will include consideration of receipt, storage and handling of nonradioactive materials.

5. **Comment:** 3.3.1.1.2 Central Section Air Supply System. No information on the number of air changes per hour is given in this section. As this is the personnel section of the facility, this is necessary information.

Response: The number of air changes per hour in Section 3.3.1.1.2 of the updated version of the F- and H-Canyon Facility SARs will be revised to include number of volume changes of air.

6. **Comment:** 3.3.1.2 Air Exhaust. No information was given on exhaust systems for nonradioactive process chemical preparation equipment. This information needs to be added.

Response: Agreed. Consistent with the requirements of the NRC format, Section 3.3.1.2 of the updated version of the F- and H-Canyon Facility SARs will be revised to include a description and capacity of the exhaust systems serving the nonradioactive process chemical preparation equipment and area.

7. **Comment:** 4.7 Unique Hazards. DOE Orders require identification and evaluation of risks not routinely encountered and accepted by the vast majority of the public. This section should adopt the words of the Orders rather than "uniquely different from normal industrial practice".

Response: Agreed. Consistent with the requirements of the NRC format, Section 4.7 of the updated F- and H-Canyon Facility SARs will be revised to use the wording of the DOE Orders in defining and identify "Unique Hazards".

8. **Comment:** 4.8 Control and Management of Effluents. The sections under this title did not discuss nonradioactive chemical effluents. This needs correcting.

Response: Consistent with the requirements of the NRC format, Section 4.8 of the updated version of the F- and H-Canyon Facility SARs will include discussions of nonradioactive chemical effluents in the section on "Control and Management of Effluents."

9. **Comment:** 5.0 Accident Analysis. "The F-Canyon operations have been evaluated on the basis of three considerations: 1) potential radiological hazards, 2) potential chemical hazards, and 3) potential conditions different from normal industrial practice." See comment above for 4.7.

Response: Section 5.0 Accident Analysis of the updated versions of the F- and H-Canyon Facilities SARs will evaluate "...Canyon Operations on

the basis of 1) potential radiological hazards, 2) potential chemical hazards, and 3) potentially hazardous conditions not normally encountered and accepted by the vast majority of the public." See the attached H- and F-Canyon Operations addendums.

10. Comment: "Risk in this analysis is defined as the product of the expected frequency that an event sequence will release radioactivity and the radiological consequences of that release." This definition only has application to 1/3 of the stated purpose quoted above. This section is incomplete.

Response: The definition of risk in the updated versions of the F- and H-Canyon Facilities SARs will be revised to include reference to both radiological and nonradiological events and consequences.

11. Comment: 5.1.1.5 Other Natural Phenomena Related Events. "Extremes in temperatures, snow, rain, and lightning, may adversely affect operations, but will not result in accident sequences that lead to direct releases of radioactive materials." This section is improperly limited to radiological consequences. Adverse effects which could lead to nonradioactive chemical exposures must also be evaluated and documented.

Response: Section 5.1.1.5 Other Natural Phenomena Related Events will be revised in the updated versions of the F- and H-Canyon Facilities SARs to include effects leading to release of either or both nonradiological and radiological materials. See the attached H- and F-Canyon Operations addendums.

12. Comment: 5.1.5 Low Energetic Events. "A low energetic event is defined as an event that will not destroy the primary confinement barrier (the primary container), but activity may penetrate it." Use of the word activity improperly limits these events to radioactive materials. All events which allow nonradioactive chemicals to escape confinement must be addressed. These sections will need revision.

Response: Consistent with the requirements of the NRC format, Section 5.1.5 of the updated version of the F- and H-Canyon Facilities will include a redefinition of a "low energetic event" to eliminate the word "activity" and replace it with the words "confined chemicals".

13. Comment: 5.1.7 Chemical Hazards. "Methodology used in this analysis and the hazards for various chemicals are discussed in detail in Reference 23." A summary of the methodology used needs to be included as a minimum. A more detailed description of hazards from each chemical is also needed. The methodology appears appropriate but Reference 23 has other problems. Reference 23 is 6 years old and is flawed in that the documents on which it relies now range from 7 to 30 years old. The carcinogen reference is now 11 years old. The initial dependence on a 5 year old publication for cancer data, without identified reasons

otherwise, was a poor decision even in 1981. In regard to toxicity the document is not current. With the availability of commercial and government data bases for toxicology information only a telephone call away, no reason exists for failing to provide the latest assessments of toxicity for any chemical compound.

Response: Agreed, Section 5.1.7 Chemical Hazards will be revised in the updated versions of the F- and H-Canyon Facilities SARs to incorporate the latest available data as of the issue date with respect to the toxicity, mutagenicity and carcinogenicity of chemicals used and or generated in the operation of the process. See the attached H- and F-Canyon Operations addendums.

14. Comment: Table 5-2. Identification of chemicals only from the Reference 23 listing of carcinogens is not consistent with DOE Standards and OSHA (29 CFR 1910.1200). Chemicals in use or produced which have been identified by either ACGIH, NTP or IARC as carcinogens must be so noted in the SAR. Toxicity should be a separate heading with the adversely affected organs and health effects identified.

Response: Table 5-2 will list all chemicals used or generated in the updated version of the F- and H-Canyon Facility SARs along with the data required to comply with DOE Standards and 29 CFR 1910.1200.

15. Comment: Table 5-3 is incomplete and contains errors. The 85-86 TLV booklet lists NO₂ as 6 mg/m³, not 9; Hydrazine TLV of 0.1 mg/m³, not 0.13. AELs may be used in addition to or the absence of the adopted DOE standard of TLVs. TLVs need to be added to the table. The issue date will be needed for the TLVs used. OSHA PELs should also be listed for comparative purposes. If TLVs are unavailable, TLVs from similar compounds may be used with an explanation of the reasons for selection. LD50s and LDLo should be listed for comparison with well known hazards such as Pb, Hg, PCBs.

Response: Table 5-3 will be revised and corrected in the updated versions of the F- and H-Canyon Facility SARs to comply with the DOE and OSHA requirements and incorporate the most recent data available at the time of issue relative to effects of chemicals used or generated in the operations of processes in the facilities.

16. Comment: 5.2 Analysis Methodology. "The purpose of the accident analysis is to determine and evaluate the risk of operating the F-Canyon facilities... Risk in this analysis is defined as the product of the expected frequency of an event that will release radioactivity and the radiological consequence." The stated risk definition improperly limits the stated purpose and fail to meet the requirements of DOE SAR Orders.

Response: Section 5.2 Analysis Methodology of the updated F- and H-Canyon Facility SARs will restate the "purpose by deleting references to

radioactivity" and to "...radiological..." and replacing with "...hazardous materials" and "...harmful..."

17. **Comment:** 5.3 Accident Frequencies. This section is improperly limited to radioactive concerns only.

Response: Consistent with the requirements of the NRC format, Section 5.3 Accident Frequencies will be expanded to include nonradiological concerns also in the updated F- and H-Canyon Facility SARs.

18. **Comment:** 5.4.4.2 Uncontrolled Reactions. No consequences are listed for 2580 lbs. of nonradioactive material or air concentrations of 100 mg/m³. This needs correction.

Response: Section 5.4.4.2 Uncontrolled Reactions will be expanded in the updated F- and H-Canyon Facility SARs to consider consequences of nonradioactive materials as well as radioactive materials in the operation of the process. See the attached H- and F-Canyon Operations addendums.

19. **Comment:** 5.4.5 Low Energetic Events. No consequences are listed for nonradioactive material when this is the most likely source of an exposure. This needs correction on an event by event basis.

Response: Section 5.4.5 Low Energetic Events will be revised in the updated version of the F- and H-Canyon SARs to include those involving a release of nonradiological materials. See the attached H- and F-Canyon Operations addendums.

20. **Comment:** 5.4.6.2 Air Reversal. In light of the conditions possible from over flows, uncontrolled reactions, leaks and spills, an analysis of the nonradioactive consequences of an air reversal is needed. It is recalled that 2 evacuations occurred in 12/86 because of nonradioactive chemical leaks and spills.

Response: Section 5.4.6.2 Air Reversals will be revised and expanded in the updated versions of the F- and H-Canyon Facilities SARs to consider consequences of potential releases of nonradioactive materials in air reversals. See the attached H- and F-Canyon Operations addendums.

21. **Comment:** 5.4.7 Chemical Hazards. DOE Orders, Federal, and State law as well as DuPont corporate policy require elimination of known hazards from the workplace. It is improper to mitigate the existence of a hazard by equivocation that only a few injuries will take place. Adjusting risk by factoring reported injuries is not acceptable. As an exposure is assumed to be at or over acceptable limits, the frequency rate is the risk rate. Methods used to reduce the consequence of an exposure may be discussed

and injury data may be used to show the success to these methods. This provides a basis for accepting the risk but it does not reduce the risk.

Response: Risk is the product of consequence in appropriate terms and frequency. Methods used to reduce consequences, if effective, do reduce risk even though the frequency part of the risk equation is not altered. The basis for "accepting risk" can be based on any of the three components of the risk equation (frequency, consequences or the product (risk)).

22. **Comment:** 5.4.8.7 Compressed Air. The consequences for a cross connection to helium, argon, carbon dioxide with the breathing air system is death to all users without any warning. This is an arguable imminent danger situation and actions are necessary to correct this possibility. More is needed on why this condition is allowed to exist.

Response: The current F- and H-Canyon SARs describe the Breathing Air supply system as it exists and identifies the potential hazard of attaching cylinders of nitrogen, CO₂ argon, helium, etc. to the manifold of the breathing air supply. The possibility of a cross connection has been accepted by the Compressed Gas Association.

Several factors act to reduce the probability of a cross connection. First, the breathing air cylinders are painted yellow, a distinctive color different from other gas cylinders. Second, the breathing air cylinders and manifold hookup are isolated by an appreciable distance from other gas cylinders. Third, the breathing air cylinders are labeled and easily readable. Fourth, operating procedures are used to make cylinder connections to the manifold and require that supervision make the breathing air connections. On the basis of the above factors, a breathing air cross connection with another gas is assessed as ID (catastrophic but remote) as per MIL-STD-882B.

23. **Comment:** 5.6.4.2 Uncontrolled Reaction. This section did not address mitigation of material being expelled through sample line and other piping back into occupied areas.

Response: Agreed, consistent with the requirements of the NRC format, Section 5.6.4.2 Uncontrolled Reaction in the next update of the F- and H-Canyon SARs will address mitigation of process materials being pressured into occupied areas such as the sample line and others as a result of uncontrolled reactions. Since this is controlled by the radiation detection system, the worker does not get a large dose.

24. **Comment:** 5.6.5 Low Energetic Events. This section did not address how fumes will not enter into occupied areas as happened in the two NO_x fume outs and evacuations in 12/86.

Response: Agreed, consistent with the requirements of the NRC format, Section 5.6.5 Low Energetic Events of the next update of the F- and H-

Canyon SARs will include a discussion of how the potential for release of fumes from canyon equipment is minimized.

25. **Comment:** 5.6.7 Chemical Hazards. More detail is needed in this section to describe an adequate protection program. Also see 5.6.5.

Response: Consistent with the requirements of the NRC format, Section 5.6.7 Chemical Hazards of the next updated F- and H-Canyon SARs will be expanded to include a detailed description of the program aimed at providing an adequate protection from all of the identified chemical hazards present in the operation of the processes in the F- and H-Canyon Facilities. See the attached H- and F-Canyon Operations addendums.

26. **Comment:** 5.6.8.7 Compressed Air. Nothing in this section address the concerns raised in 5.4.8.7 and how DOE standards for respiratory protection are adequately implemented for this facility.

Response: Consistent with the requirements of the NRC format, Section 5.6.8.7 Compressed Air of the next update of the F- and H-Canyon SARs will be expanded to include discussion of the program and equipment developed and installed that implements compliance with DOE Standards for respiratory protection of facility personnel.

27. **Comment:** 6.1.12 Chemical Handling and Storage. No risks were addressed specific to the identified areas in this document. As leaks were described as giving exposures at or above acceptable levels, this section did not provide enough detail on how this is controlled.

Response: Consistent with the requirements of the NRC format, Section 6.1.12 Chemical Handling and Storage will be expanded and/or revised in the next update of the F- and H-Canyon SARs to include consideration of all of the risks arising in the handling and storage of chemicals in the Third Level Cold Feed Preparation and Storage area of the Canyon Facility and in the storage facilities where level quantities of chemicals are handled.

INDUSTRIAL SAFETYGeneral Comment

1. **Comment:** The document fails to address the risk of non-radiological hazards in F- and H-Canyon. Non-radiological hazards identified in this document must be analyzed and a qualitative risk evaluation should be provided. A guideline which outlines an acceptable approach to this type of analyses has previously been provided to Du Pont. A copy of the guidance is attached for use, as appropriate.

Response: As stated in the April 8, 1987 letter from R.L. Morgan to J.T. Grahaghan, the next updated versions of F- and H-Canyon SARs will address nonradiological hazards and the associated risks evaluated in accordance with the guidelines which were identified in the above comment as being acceptable and in compliance with DOE Order 5481.1A. See the attached H- and F-Canyon Operations addendums.

Specific Comments

1. **Comment:** Section 1.1 Introduction - Risk is defined only in terms of radiological consequence. Suggest this be expanded to include nonradiological consequences.

Response: The definition of risk in Section 1.1 Introduction of the next updated version of the F- and H-Canyon Facilities SAR will not be restricted to radiological consequences.

2. **Comment:** Section 1.1.2 - Approach - This section does not discuss the use of Process Hazards Reviews. I suggest the PHR be utilized as a method for risk analysis of nonradiological hazards.

Response: Process Hazards Reviews are one method of risk analysis of nonradiological hazards. Consistent with the requirements of the NRC format, PHRs will be used for the next update of the F- and H-Canyon Facility SAR.

3. **Comment:** Section 1.2.1 - Facility Operations - I recommend the release of toxic materials to the atmosphere be included as one of the principal safety concerns expressed in this section.

Response: Consistent with the requirements of the NRC format, Section 1.2.1 - Facility Operations of the next updated version of the F- and H-Canyon Facility SARs will include the release of toxic materials to the environment as one of the principal safety concerns relating to operations conducted in the facilities.

4. **Comment:** Section 1.2.2 - Facility Risk - A quantitative or qualitative nonradiological risk conclusion should be included in this section.

Response: Consideration of nonradiological hazards and risks as a result of Facility operations will include considerations of nonradiological hazards in the next updated version of the F- and H-Canyon Facility SARs as stated in the April 8, 1987 letter from R.L. Morgan to J.T. Granaghan.

5. Comment: Section 3.2 - Process and Facility Description - The operations described in this section of the SAR must be analyzed in terms of nonradiological risks. A quantitative or qualitative risk conclusion should be provided in Section 5.0 for each of the significant hazards associated with the operations described in this section.

Response: The next update of the F- and H-Canyon Facility SARs will include nonradiological hazards in Section 3.2 Process and Facility Description and will include an analyses of these in Section 5.0.

6. Comment: Section 5.0 - Accident Analysis - The canyon operations must be analyzed on the basis of nonradiological hazards also. See the attached (Ltr Morgan/Granaghan, dated April 8, 1987) guidance.

Response: Section 5.0 - Accident Analyses of the next update of the F- and H-Canyon SARs will include an analyses of the nonradiological hazards that are identified in each canyon facility.

7. Comment: Section 5.4 - Accident Consequences - The consequences of each identified accident initiator (Section 5.1) should be expressed in terms of personnel injury and severity.

Response: The manner in which consequences of accident-initiators are expressed is addressed in the attached F- and H-Canyon Operations addendum.

8. Comment: Section 5.5 - Risks - This section should include nonradiological risk expression for operations in F- and H-Canyons. The risks should be based on the probability of nonradiological hazards identified in Section 5.1 and 5.3, 5.4 of the SAR.

Response: Section 5.5 - Risks - of the next update of the F- and H-Canyon SARs will include consideration of nonradiological hazards in evaluation of the risks associated with conducting the operation of the separations processes in the canyon facility and its support activities.

FIRE PROTECTION

1. Comment: The SAR deals only with fires that will have potential radiological consequences (person-rem/yr). All possible fires should be addressed.

Response: Consistent with the requirements of the NRC format, updated versions of the F- and H-Canyon SARs will include consideration of all potentially hazardous fires that could occur in the facility operations or its support activities.

2. Comment: SAR should evaluate and provide estimates of the potential losses or damages to government property from any fire that could occur in these facilities.

Response: Historically, estimates of potential losses or damage to radiochemical processing facilities due to potential fires has not been emphasized or even included in SARs. If it is to be included, the estimates of potential damages/losses from the potential fires identified in the SARs will be included in the next updated version of the F- and H-Canyon SARs.

3. Comment: The section on Instrumentation and Controls addresses the Fire Brigade System. This is erroneous. Area Fire Brigades do not exist anymore. The real question is: How does this affect the SARs? It is suggested that an assessment be provided.

Response: Consistent with the requirements of the NRC format, the description of Fire Protection in the Fire Protection Systems Section 3.2.5.4 will be reviewed to reflect the reorganized status and responsibilities of the Fire Fighting Organizations and the results incorporated in the next updated versions of the F- and H-Canyon Facility SARs.

4. Comment: The same section mentions the deluge sprinkler systems in the canyons and the fact that the deluge valves in the railroad tunnel (2nd level), the warm and hot decontamination cells (2nd level) and in the gang valve corridors have been "blanked". If this means that the systems have been disabled, then this should be stated clearly and an assessment of the situation must be provided: Do we, or don't we have adequate fire protection in the canyons to mitigate the effects of credible fires which could expose personnel unnecessarily?

Response: See item 3 above.

5. Comment: The subsection on Safety Management Systems dealing with Fire Protection must be expanded to state exactly what the systems are. It is not enough to say that a subcommittee "works closely with the Fire Prevention Group". Additional information in terms of "where they've

been, where they're going and how are they going to get there must be provided".

Response: Section 4.9.5 Fire Protection in the updated version of F- and H-Canyon SARs will be revised to emphasize that the function of the Fire Protection Subcommittee of the 200 Area Central Safety Committee is to continuously review and evaluate the currently existing fire protection systems and programs in terms of effectiveness and capability. It is also charged with anticipating needs and providing Facility Management with early information as to future needs as well as those that which relate to increasing existing systems effectiveness and capabilities.

ADDENDUM TO DPSTSA-200-10, SUP-4, SAFETY ANALYSIS - 200 AREA
-SAVANNAH RIVER PLANT, F-CANYON OPERATIONS

The F-Canyon structure and supporting engineered safety features that constitute an effective radiological confinement system also serve to protect personnel from non-nuclear hazards posed by accident initiating events such as internal fire and uncontrolled reactions. Chemical hazards associated with the gallery feed tanks on the third level cold feed preparation and storage area (CFP) and the rupture of these tanks are the only non-nuclear safety risks identified in the F-Canyon operations.

Chemical hazards due to leaks, overflows, transfer errors, and uncontrolled reactions are discussed in detail in the SAR (see Section 5.4.7). The evaluated frequency for a hazardous liquid injury in CFP is $1.3 \times 10^{-5}/\text{hr}$ (0.11/year) or about one per nine years, with the most likely injury being an acid burn to the skin. The evaluated frequency for hazardous vapor injury in CFP is $<5.6 \times 10^{-6}/\text{hr}$ (0.05/year) or about one per twenty years. There is no injury from hazardous vapor uptake in the SRP records. Considering that a chemical burn typically occurs within a short exposure period and using the Du Pont Safety and Fire Protection Guidelines¹ with the Military Standard 882B² (MIL-STD-882B), chemical hazards in the F-Canyon Operations Area are assessed as IIIIE (marginal and improbable).

Rupture of a tank containing corrosive chemicals in the CFP could cause burns to employees in the vicinity of the spill. Given sufficient concentrations of such a chemical a fatality might ensue if an employee received extensive and sudden contact with the material. There are approximately 26 tanks (2 to 8 feet in diameter) in CFP that are reported to contain Nitric Acid or Sodium Hydroxide at concentrations of 50% or more. It is estimated that in the daytime, there will be an average of 10 employees (operators and maintenance crew) working in the area.

The frequency of a single tank rupture as evaluated in Reference 3 is 3.5×10^{-5} years between occurrences ($2.8 \times 10^{-6}/\text{year}$). Based on an estimated radius of entrapment of 10 feet, and an overall area of 13,000 square feet, the conditional probability that an employee will be in the affected area when a tank ruptures is 0.24. Consequently, the expected frequency of a fatality due to rupture of a single tank is estimated to be $1.8 \times 10^{-5}/\text{year}$ or one per 55,000 years. The rupture of a tank in CFP can therefore be assessed as ID (catastrophic but remote) as per MIL-STD-882B. There is no tank rupture recorded in the SRP data bank.

At the Savannah River Plant, Du Pont Safety policies and commitment to safety provide a safe work environment and assist employees in avoiding injuries. On the basis of these practices and the analysis previously discussed, non-nuclear hazards are qualitatively judged to be in the remote to improbable range for the F-Canyon Operations.

REFERENCES

1. DuPont Safety and Fire Protection Guidelines, Special Volume: Process Hazards Management, Section 6.4, Page 7, August 1978.
2. System Safety Program Requirements, MIL-STD-882B, Department of Defense, March 30, 1984.
3. Renwick, A. J., Evaluation of Industrial Hazards. Building 321-M, Internal Report DPST-87-236, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, SC, March 12, 1987, 7/01/87.

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