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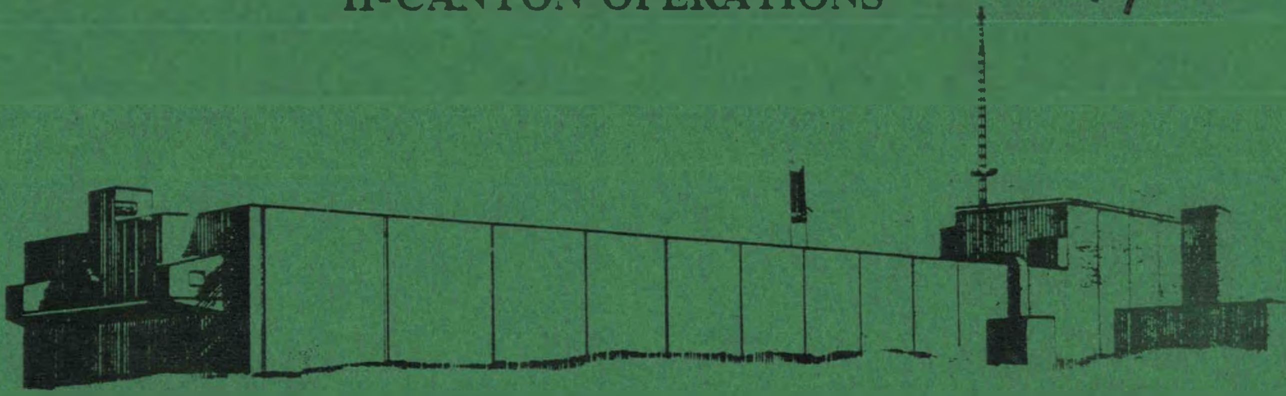

February 1986

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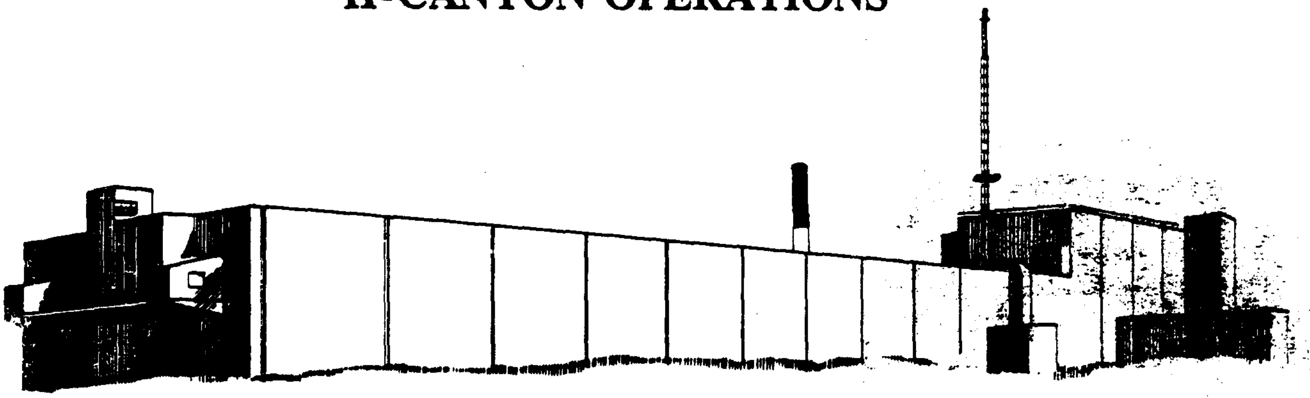

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TABLE OF CONTENTS

1.0 INTRODUCTION AND SUMMARY 1-1

1.1 INTRODUCTION 1-1

1.1.1 Historical Perspective 1-1

1.1.2 Approach 1-3

1.2 SUMMARY DISCUSSION 1-4

1.2.1 Facility Operations 1-4

1.2.2 Facility Risk 1-5

1.3 REFERENCES 1-8

2.0 SITE EVALUATION 2-1

2.1 SITE DESCRIPTION 2-1

2.1.1 Population Characteristics 2-1

2.1.2 Seismology 2-1

2.1.3 Geology 2-9

2.1.4 Hydrology 2-11

2.1.4.1 Surface Water 2-11

2.1.4.2 Groundwater 2-11

2.1.5 Meteorology 2-12

2.2 FACILITY DESCRIPTION 2-13

2.2.1 The H-Canyon Building 221-H 2-13

2.2.2 Uranium Cycle 2-15

2.2.3 Neptunium Cycle 2-15

2.2.4 Special Programs 2-15

2.3 REFERENCES 2-15

3.0 FACILITY AND PROCESS DESIGN 3-1

3.1 DESIGN CRITERIA 3-1

TABLE OF CONTENTS (Continued)

3.2 PROCESS AND FACILITY DESCRIPTION 3-1

3.2.1 Summary Description 3-1

3.2.2 Process Design Considerations 3-2

3.2.2.1 Receipt and Charging of Irradiated Material . . 3-8

3.2.2.2 Dissolution 3-13

3.2.2.3 Neptunium Aluminum Targets 3-15

3.2.2.4 Off-Gas 3-15

3.2.2.5 Head End 3-17

3.2.2.5.1 Concentration 3-17

3.2.2.5.2 Precipitation 3-18

3.2.2.5.3 Centrifugation 3-19

3.2.2.6 Solvent Extraction 3-21

3.2.2.6.1 First Cycle Solvent Extraction: Primary Fission Product Separation 3-25

3.2.2.6.2 Partitioning 3-25

3.2.2.6.3 Second Uranium Cycle 3-27

3.2.2.6.4 Second Product Cycle: Neptunium Purification 3-29

3.2.2.6.5 Second Product Cycle: Plutonium Purification 3-30

3.2.2.6.6 Thorium Purification 3-30

3.2.2.6.7 Solvent Extraction: Safety Considerations 3-30

3.2.2.6.8 Solvent Flammability 3-31

3.2.2.7 Anion Exchange: H Frames Process 3-32

3.2.2.7.1 First Column, RC-1 3-32

3.2.2.7.2 Second Column, RC-2 3-35

TABLE OF CONTENTS (Continued)

3.2.2.7.3 Third Column, RC-4 3-35

3.2.2.7.4 Np and Pu Recovery From Anion Raffinates 3-35

3.2.2.7.5 Safety Considerations 3-36

3.2.2.8 Evaporation 3-36

3.2.2.8.1 General 3-36

3.2.2.8.2 Low Acidity Waste Evaporation . . . 3-37

3.2.2.8.3 High Activity Waste Evaporation . . 3-37

3.2.2.8.4 Batch Evaporator Operation 3-38

3.2.2.8.5 Safety Considerations - Evaporators 3-38

3.2.3 Instrumentation and Controls 3-40

3.2.3.1 Dissolver 3-40

3.2.3.2 Dissolver Off-Gas 3-41

3.2.3.3 Head End 3-41

3.2.3.4 Solvent Extraction 3-42

3.2.3.5 Evaporation 3-43

3.2.3.6 Anion Exchange 3-44

3.2.3.7 Radiation Monitoring 3-44

3.2.3.8 Fire Detection 3-44

3.2.3.9 Nuclear Incident Monitor System 3-44

3.2.3.10 Data Logger System 3-45

3.2.4 Electrical Power Distribution 3-45

3.2.4.1 Normal Power 3-45

3.2.4.2 Emergency Power 3-46

3.2.5 Auxiliary Systems and Support Facilities 3-47

3.2.5.1 Water Systems 3-47

TABLE OF CONTENTS (Continued)

3.2.5.1.1 Cooling Water 3-47

3.2.5.1.2 Fire Water 3-48

3.2.5.1.3 Domestic Water 3-49

3.2.5.2 Steam Supply 3-49

3.2.5.2.1 Generation 3-49

3.2.5.2.2 Distribution 3-50

3.2.5.3 Compressed Air 3-51

3.2.5.4 Fire Protection Systems 3-51

3.3 ENGINEERED SAFETY FEATURES 3-51

3.3.1 Ventilation Systems 3-51

3.3.1.1 Air Supply 3-53

3.3.1.1.1 Canyon Air Supply System 3-53

3.3.1.1.2 Center Section Air Supply System . . 3-55

3.3.1.1.3 Gang Valve Corridor Air System . . . 3-58

3.3.1.2 Air Exhaust 3-58

3.3.1.2.1 Canyon Air Exhaust 3-58

3.3.1.2.2 Center Section Exhaust 3-60

3.3.1.2.3 Process Vessel Vent System 3-60

3.3.1.2.4 Recycle Vent System 3-63

3.3.2 Diversion of Segregated or Circulated Cooling Water . . 3-63

3.3.3 Secondary Confinement 3-64

3.4 DECOMMISSIONING CONSIDERATIONS 3-66

3.4.1 Remote Operation 3-66

3.4.2 Partitioning of Process Functions 3-66

3.4.3 Protection Coatings and Liners 3-66

3.5 REFERENCES 3-67

TABLE OF CONTENTS (Continued)

4.0 DESCRIPTION OF OPERATIONS 4-1

4.1 ORGANIZATIONAL STRUCTURE 4-1

4.2 OPERATIONAL DESCRIPTION 4-1

4.3 PROCEDURES 4-4

4.3.1 Administrative Procedures and Control 4-4

4.3.1.1 Notification and Reporting of Occurrences to DOE 4-4

4.3.1.2 Special Incident Reports 4-5

4.3.1.3 Process Incident Investigations and Reports . . 4-5

4.3.2 Operating and Maintenance Procedures 4-6

4.3.2.1 Operational Control Documents 4-7

4.3.2.2 Operating Procedures 4-11

4.3.3 Emergency Procedures and Plans 4-12

4.4 TRAINING 4-14

4.5 REVIEW AND AUDIT 4-16

4.6 INSPECTION AND TESTING 4-17

4.6.1 Ventilation Systems 4-17

4.6.2 Cooling Water Diversion 4-17

4.6.3 Confinement Barriers 4-18

4.6.4 Radiation Shielding 4-18

4.7 UNIQUE HAZARDS 4-18

4.8 CONTROL AND MANAGEMENT OF EFFLUENTS 4-20

4.8.1 Effluent Radioisotopes 4-20

4.8.2 Effluent Treatment and Control 4-22

4.8.3 Effluent Monitoring Program 4-22

TABLE OF CONTENTS (Continued)

4.8.4 Offsite Consequences of Radionuclides Released During Normal Operations (200-Area Operations) 4-29

4.9 SAFETY MANAGEMENT SYSTEMS 4-29

4.9.1 Radiation and Contamination Control 4-29

4.9.2 Occupational Doses from Normal Operations 4-34

4.9.3 Industrial Safety 4-38

4.9.4 Industrial Hygiene 4-38

4.9.5 Fire Protection 4-38

4.9.6 Criticality Control 4-41

4.10 REFERENCES 4-44

5.0 ACCIDENT ANALYSIS 5-1

5.1 INITIATORS AND DESCRIPTION OF ACCIDENTS 5-2

5.1.1 Natural Phenomena 5-2

5.1.1.1 Winds 5-2

5.1.1.2 Earthquake 5-3

5.1.1.3 Meteorite Impact 5-3

5.1.1.4 Floods 5-3

5.1.1.5 Other Natural Phenomena Related Events 5-3

5.1.2 Externally Induced Events 5-5

5.1.2.1 Aircraft Crash 5-5

5.1.2.2 Adjacent Explosion 5-5

5.1.2.3 Adjacent Fire 5-6

5.1.3 High Energetic Events 5-6

5.1.4 Medium Energetic Events 5-6

5.1.4.1 Internal Fire 5-6

5.1.4.2 Uncontrolled Reaction 5-7

TABLE OF CONTENTS (Continued)

5.1.4.3 Criticality 5-10

5.1.5 Low Energetic Events 5-13

5.1.5.1 Transfer Error 5-13

5.1.5.2 Overflow 5-14

5.1.5.3 Leak 5-14

5.1.5.4 Chemical Addition Error 5-14

5.1.5.5 Pluggage 5-14

5.1.5.6 Siphoning 5-15

5.1.5.7 Processing Short-Cooled Fuels 5-15

5.1.5.8 Ruthenium Volatilization 5-15

5.1.5.9 Iodine Reactor Failure 5-15

5.1.5.10 Coil and Tube Failure 5-16

5.1.5.11 Waste Header Failure 5-16

5.1.6 Residual Release Events 5-16

5.1.6.1 Suckback 5-16

5.1.6.2 Air Reversal 5-16

5.1.7 Chemical Hazards 5-17

5.1.8 Support Systems 5-17

5.1.8.1 Shops and Decontamination Facilities 5-17

5.1.8.2 Cell Covers 5-20

5.1.8.3 Gang Valve Corridors 5-20

5.1.8.4 Cranes 5-21

5.1.8.5 Sample Aisles 5-21

5.1.8.6 Steam Distribution 5-22

5.1.8.7 Compressed Air 5-22

5.1.8.8 Electrical Distribution 5-22

TABLE OF CONTENTS (Continued)

5.1.8.9 Cooling Water Distribution 5-22

5.1.9 Engineered Safety Features 5-22

5.1.9.1 Ventilation System 5-22

5.1.9.2 Water Return and Diversion 5-24

5.1.9.3 Seepage Basins 5-25

5.1.9.4 Secondary Confinement for Liquids 5-25

5.1.9.5 Pressure Regulators 5-26

5.1.9.6 Neutron Monitors 5-26

5.2 ANALYSIS METHODOLOGY 5-26

5.2.1 Data Base 5-27

5.2.1.1 Information Sources 5-27

5.2.1.2 Data Storage and Retrieval 5-27

5.2.2 Radionuclide Distribution 5-28

5.2.3 Event Trees 5-28

5.2.4 Fault Trees 5-31

5.2.5 Dose Models 5-31

5.2.5.1 Models for Calculating Doses from Atmospheric Releases for Accidents 5-31

5.2.5.2 Models for Calculating Doses from Liquid Releases 5-33

5.3 ACCIDENT FREQUENCIES 5-34

5.3.1 Natural Phenomena 5-34

5.3.1.1 Winds 5-34

5.3.1.2 Earthquake 5-34

5.3.1.3 Meteorite Impact 5-34

5.3.1.4 Floods 5-36

5.3.1.5 Other Natural Phenomena Related Events 5-36

TABLE OF CONTENTS (Continued)

5.3.2 Externally Induced Failures 5-36

5.3.2.1 Aircraft Crash 5-36

5.3.2.2 Adjacent Explosion 5-37

5.3.2.3 Adjacent Fire 5-37

5.3.3 High Energetic Events 5-37

5.3.4 Medium Energetic Events 5-37

5.3.4.1 Fire 5-37

5.3.4.2 Uncontrolled Reaction 5-39

5.3.4.3 Criticality 5-39

5.3.5 Low Energetic Events 5-39

5.3.5.1 Transfer Error 5-39

5.3.5.2 Overflow 5-42

5.3.5.3 Leak 5-42

5.3.5.4 Chemical Addition Error 5-42

5.3.5.5 Pluggage 5-50

5.3.5.6 Siphoning 5-50

5.3.5.7 Processing Short-Cooled Fuels 5-50

5.3.5.8 Ruthenium Volatilization 5-50

5.3.5.9 Iodine Reactor Failure 5-50

5.3.5.10 Coil and Tube Failure 5-50

5.3.5.11 Waste Header Failure 5-52

5.3.6 Residual Activity Release Events 5-52

5.3.6.1 Suckback 5-52

5.3.6.2 Air Reversal 5-53

5.3.7 Chemical Hazards 5-53

5.3.7.1 Frequency of Releases to Outside Areas 5-55

TABLE OF CONTENTS (Continued)

5.3.7.2 Frequency of Releases to Occupied Areas 5-55

5.3.8 Support Systems 5-55

5.3.8.1 Shops and Decontamination Facilities 5-55

5.3.8.2 Cell Covers 5-57

5.3.8.3 Gang Valve Corridors 5-57

5.3.8.4 Cranes 5-57

5.3.8.5 Sample Aisles 5-57

5.3.8.6 Steam Distribution 5-57

5.3.8.7 Compressed Air 5-60

5.3.8.8 Electrical Distribution 5-60

5.3.8.9 Cooling Water Distribution 5-60

5.3.9 Engineered Safety Features 5-60

5.3.9.1 Ventilation System 5-60

5.3.9.2 Water Return and Diversion 5-61

5.3.9.3 Seepage Basins 5-61

5.3.9.4 Secondary Confinement for Liquids 5-65

5.3.9.5 Pressure Regulators 5-65

5.3.9.6 Neutron Monitors 5-65

5.4 ACCIDENT CONSEQUENCES 5-65

5.4.1 Natural Phenomena 5-65

5.4.1.1 Winds 5-66

5.4.1.2 Earthquake 5-66

5.4.1.3 Meteorite Impact 5-67

5.4.1.4 Other Natural Phenomena Related Events 5-67

5.4.2 Externally Induced Failures 5-68

5.4.2.1 Aircraft Crash 5-69

TABLE OF CONTENTS (Continued)

5.4.2.2 Adjacent Explosion 5-69

5.4.2.3 Adjacent Fire 5-69

5.4.3 High Energetic Events 5-69

5.4.4 Medium Energetic Events 5-69

5.4.4.1 Process Fire 5-70

5.4.4.2 Uncontrolled Reaction 5-70

5.4.4.3 Criticality 5-71

5.4.5 Low Energetic Events 5-75

5.4.5.1 Transfer Error 5-75

5.4.5.2 Overflow 5-76

5.4.5.3 Leak 5-76

5.4.5.4 Chemical Addition Error 5-80

5.4.5.5 Pluggage 5-80

5.4.5.6 Siphoning 5-80

5.4.5.7 Processing Short-Cooled Fuels 5-81

5.4.5.8 Ruthenium Volatilization 5-81

5.4.5.9 Iodine Reactor Failure 5-81

5.4.5.10 Coil and Tube Failure 5-81

5.4.5.11 Waste Header Failure 5-83

5.4.6 Residual Release Events 5-83

5.4.6.1 Suckback 5-83

5.4.6.2 Air Reversal 5-85

5.4.7 Chemical Hazards 5-85

5.4.8 Support Systems 5-87

5.4.8.1 Shop and Decontamination Facilities 5-87

5.4.8.2 Cell Covers 5-87

TABLE OF CONTENTS (Continued)

5.4.8.3 Gang Valve Corridors 5-87

5.4.8.4 Cranes 5-88

5.4.8.5 Sample Aisles 5-89

5.4.8.6 Steam Distribution 5-89

5.4.8.7 Compressed Air 5-90

5.4.8.8 Electrical Distribution 5-90

5.4.8.9 Cooling Water Distribution 5-90

5.4.9 Engineered Safety Features 5-90

5.4.9.1 Ventilation System 5-91

5.4.9.2 Water Return and Diversion 5-91

5.4.9.3 Seepage Basins 5-91

5.4.9.4 Secondary Confinement for Liquids 5-91

5.4.9.5 Pressure Regulators 5-92

5.4.9.6 Neutron Monitors 5-92

5.5 RADIOLOGICAL RISKS 5-92

5.5.1 Risks Due to Natural Phenomena 5-93

5.5.2 Risks Due to Externally Induced Failures 5-93

5.5.3 Risks Due to Process Related Occurrences 5-93

5.5.4 Summary of Risks from Accidents 5-93

5.5.5 Risks to Operating Personnel 5-93

5.6 ACCIDENT MITIGATION 5-105

5.6.1 Natural Phenomena 5-105

5.6.1.1 Winds and Tornadoes 5-105

5.6.1.2 Earthquake 5-106

5.6.1.3 Meteorite 5-106

5.6.1.4 Floods 5-106

TABLE OF CONTENTS (Continued)

5.6.1.5 Other Natural Phenomena 5-106

5.6.2 Externally Induced Failures 5-106

5.6.3 High Energetic Events 5-107

5.6.4 Medium Energetic Events 5-107

5.6.4.1 Fire 5-107

5.6.4.2 Uncontrolled Reaction 5-107

5.6.4.3 Criticality 5-107

5.6.5 Low Energetic Events 5-108

5.6.6 Residual Release Events 5-109

5.6.6.1 Suckback 5-109

5.6.6.2 Air Reversal 5-109

5.6.7 Chemical Hazards 5-109

5.6.8 Support Systems 5-109

5.6.8.1 Shops and Decontamination Facilities 5-109

5.6.8.2 Cell Covers 5-109

5.6.8.3 Gang Valve Corridors 5-109

5.6.8.4 Cranes 5-110

5.6.8.5 Sample Aisles 5-110

5.6.8.6 Steam Distribution 5-110

5.6.8.7 Compressed Air 5-110

5.6.8.8 Electrical Distribution 5-110

5.6.8.9 Cooling Water Distribution 5-111

5.6.9 Engineered Safety Features 5-111

5.6.9.1 Ventilation System 5-111

5.6.9.2 Water Return Diversion 5-111

5.6.9.3 Seepage Basins 5-111

TABLE OF CONTENTS (Continued)

5.6.9.4 Secondary Confinement 5-112

5.7 REFERENCES 5-112

6.0 SAFETY-RELATED ITEMS 6-1

6.1 SAFETY-RELATED SYSTEMS, STRUCTURES, AND ITEMS 6-1

6.1.1 Fuel Receipt and Storage 6-1

6.1.2 Dissolving 6-4

6.1.3 Head End 6-7

6.1.4 First Solvent Extraction Cycle 6-8

6.1.5 Second Uranium Solvent Extraction Cycle 6-12

6.1.6 Second Neptunium Solvent Extraction Cycle 6-15

6.1.7 Frames and Frames Waste Recovery 6-16

6.1.8 Evaporation 6-16

6.1.9 Rerun 6-17

6.1.10 Waste Disposal 6-19

6.1.11 Chemical Handling and Storage 6-19

6.1.12 Ventilation System 6-20

6.1.13 Cooling Water System 6-20

6.1.14 Emergency Power Distribution 6-22

6.2 SURVEILLANCE REQUIREMENTS 6-23

6.3 REFERENCES 6-23

7.0 QUALITY ASSURANCE 7-1

7.1 QUALITY ASSURANCE PLAN AND MANUAL 7-1

7.2 PROCESS HAZARDS 7-1

7.3 REFERENCES 7-1

TABLE OF CONTENTS (Continued)

8.0 GLOSSARY OF TERMS 8-1

APPENDIX A - H-CANYON EMERGENCY POWER TO CRITICAL EQUIPMENT A-1

APPENDIX B - H-CANYON EVENT TREES AND FAULT TREE B-1

APPENDIX C - H-CANYON SOURCE TERM CALCULATIONS C-1

APPENDIX D - H-CANYON DOSE CALCULATIONS D-1

APPENDIX E - RESPONSE TO SROO COMMENTS AND NON-NUCLEAR HAZARDS ADDENDUM E-1

LIST OF FIGURES

1-1	Flow of Radioactive Materials in the 200 Separations Area	1-2
2-1	The Savannah River Plant Site	2-3
2-2	Plot Plan, 200 H-Area	2-4
2-3	Location of SRP Relative to Surrounding Population Centers	2-7
2-4	Profile of Geologic Formations Beneath the SRP	2-10
2-5	H-Canyon Flowsheet	2-14
3-1	Building Isometric Sections	3-3
3-2	Building 221-H	3-4
3-3	Hanford Connector	3-7
3-4	Dissolver Insert for SRP Fuels	3-10
3-5	HFIR Dissolver Insert Probe	3-11
3-6	6.3D Electrolytic Dissolver	3-12
3-7	Liquid Contact Principle	3-14
3-8	Head End Centrifuge and Centrifugation Steps	3-20
3-9	HM Process Solvent Extraction Flowsheet	3-22
3-10	Approximate Location of H-Canyon Equipment	3-23
3-11	Isometric View of Mixer-Settlers	3-26
3-12	Tank Decanter	3-28
3-13	Target Processing in Building 221-H	3-33
3-14	Frame Waste Recovery Process	3-34
3-15	Standard Coil Batch Evaporator and Column	3-39
3-16	Canyon Spray System - Typical Section	3-52
3-17	221-H Ventilation Air Flow	3-54
3-18	Exhaust Routes for Ventilation Air	3-61
3-19	Vessel Vent Scrubber	3-62
4-1	Atomic Energy Division Management	4-2

LIST OF FIGURES (Continued)

4-2	Organization of Savannah River Plant	4-3
4-3	Document Requirements and Procedures Flow Chart	4-8
4-4	200 H-Area Liquid Effluents	4-26
4-5	200 H-Area Airborne Effluents	4-27
4-6	Health Protection Department Organization	4-32
4-7	SRP Safety Committees	4-39
5-1	Example Event Tree for Airborne Releases	5-30
5-2	Example Fault Tree for Low Energetic Event Release of Liquid	5-32
5-3	Earthquake Frequency at the Savannah River Plant	5-35
5-4	Cumulative Distribution Functions for Criticality in the Canyons	5-74

LIST OF TABLES

1-1	Summary of Risks for H-Canyon Accidents for All Unit Operations	1-6
1-2	Comparison of Risk for Accident and Normal Operation	1-7
2-1	Approximate Distances to Locations of Interest from SRP	2-2
2-2	Onsite Personnel at SRP, November 1985	2-5
2-3	1980 Population Counts	2-6
3-1	HM Process - Composition and Relative Process Flow	3-24
3-2	Differential Air Pressure	3-57
3-3	Canyon Vessel Radiation Shields	3-65
4-1	Inspections and Tests on Confinement Barriers	4-19
4-2	Significant Radionuclides Present in Gases and Vapors from 200-Area Operations	4-21
4-3	Significant Radionuclides That May be Released as Particulates From the 200-Area	4-23
4-4	Radionuclide Releases to the Atmosphere From the 200-Area	4-24
4-5	Measured Liquid Releases to Surface Streams from 200-Area Operations	4-25
4-6	Annual Dose Due to Release of Radionuclides in Gaseous Effluents	4-30
4-7	Annual Dose Due to Release of Radionuclides in Liquid Effluents	4-31
4-8	Health Protection Department Organizational Responsibilities	4-33
4-9	Plant Radiation Dose Guide Values	4-35
4-10	Average Exposures to Employee Groups Assigned to H-Canyon Facility During 1984	4-36
4-11	Airborne Radioactivity Concentrations Requiring Filter or Air-Supplied Respiratory Protective Equipment	4-37
4-12	Subcommittee Primary Functions	4-40
4-13	Technical Standards Containing Limits for Isotopes Handled in 200-H Area Facility	4-42

LIST OF TABLES (Continued)

5-1 An Abbreviated Modified Mercalli Scale for Earthquake Description 5-4

5-2 Chemicals in Cold Feed Preparation and Storage 5-18

5-3 Hazardous Chemicals 5-19

5-4 Mix of Radionuclides for Various Process Solutions 5-29

5-5 Causes of Ignition - 200-H Area and 221-H Canyon Fires 5-38

5-6 Frequency and Canyon Systems Affected by Uncontrolled Reactions 5-40

5-7 H-Canyon Criticality Potential 5-41

5-8 Canyon System in Which Transfer Error was Initiated 5-43

5-9 Canyon System into Which Erroneous Transfer was Made 5-44

5-10 Canyon Systems Affected by Overflows 5-45

5-11 Canyon Systems in Which Leaks Occur 5-46

5-12 Frequencies of Contaminated Process Solution Leaks 5-47

5-13 Canyon System from Which Chemical Addition Error was Initiated 5-48

5-14 Canyon System into Which Chemical Addition Error was Made . . . 5-49

5-15 Canyon Systems Affected by Coil and Tube Failures 5-51

5-16 Frequencies of Chemical Releases from Cold Feed Preparation Area 5-54

5-17 Frequencies of Chemical Releases to Occupied Areas 5-56

5-18 Cause and Frequencies of Crane Malfunctions 5-58

5-19 Causes and Frequencies of Equipment Malfunctions in Sample Aisles 5-59

5-20 Frequencies of Local Air Reversals 5-62

5-21 Ventilation System Single Blower Failure Rates 5-63

5-22 Frequencies for Release of Activity into Plant Streams 5-64

5-23 Density Functions for Criticality in Canyon Operations 5-72

LIST OF TABLES (Continued)

5-24 Curies of Volatile Fission Products Available Following 2 x 10^{18} Fissions in Solution 5-73

5-25 Masses of Materials Involved in Transfer Errors 5-77

5-26 Liquid Release to 211-H Due to Transfer Errors 5-78

5-27 Masses of Materials Involved in Overflows 5-79

5-28 Total Activity Release per Coil Leak (F- and H-Canyons) 5-82

5-29 Liquid Releases to Retention Basin and Four Mile Creek Due to Coil and Tube Failure 5-84

5-30 Consequences of Air Reversals 5-86

5-31 Risks Due to Natural Phenomena for H-Canyon Unit Operations . . 5-94

5-32 Risks Due to Externally Induced Failures for H-Canyon Unit Operations 5-95

5-33 Risks Due to Process Related Occurrences for H-Canyon Unit Operations 5-96

5-34 Summary of Risks for H-Canyon Accidents for All Unit Operations 5-100

5-35 Summary of Risks for H-Canyon Unit Operations for All Accidents 5-101

5-36 Summary of Risk for Accidents 5-103

6-1 Safety Related Systems, Structures, and Items 6-2

6-2 Safety Instrumentation, Monitors, and Controls 6-3

1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The H-Canyon facility is located in the 200 Separations Area and uses the HM process* to separate uranium, neptunium, plutonium, and fission products. Irradiated uranium fuels containing ^{235}U at enrichments from 1.1% to 94% are processed and recovered, along with neptunium and plutonium isotopes. 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 H-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 H-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 H-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 H-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 H-Canyon separation facility was completed in the mid-1950s. Processing of irradiated depleted uranium fuel using the Purex process began in July 1955. The introduction of enriched uranium fuels into Savannah River reactor operations to produce a greater variety of products required the development of a modified process to accommodate recovery of unburned uranium from spent enriched uranium fuels. This H-Modified (HM) process was introduced in the H-Canyon in May 1959. The process was modified in 1963 to permit recovery of neptunium as well as enriched uranium. The H-Canyon operation continues to employ the large equipment designed for low enrichment fuels. To minimize criticality risk, a dilute flowsheet utilizing administrative and procedural controls rather than geometry is utilized. In addition to SRP fuels, offsite fuels from research and test reactors can be dissolved and processed using the HM process. An electrolytic dissolver, used for processing fuels clad in stainless steel, and other metals resistant to nitric acid attack, was installed in 1969. Modifications and improvements have been made subsequently to allow the canyon to perform a wide variety of missions.

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*The HM process is used for irradiated uranium fuels that contain 1.1 to 94% ^{235}U. A Purex process was used in H-Canyon prior to the May 1959 startup of the HM process.

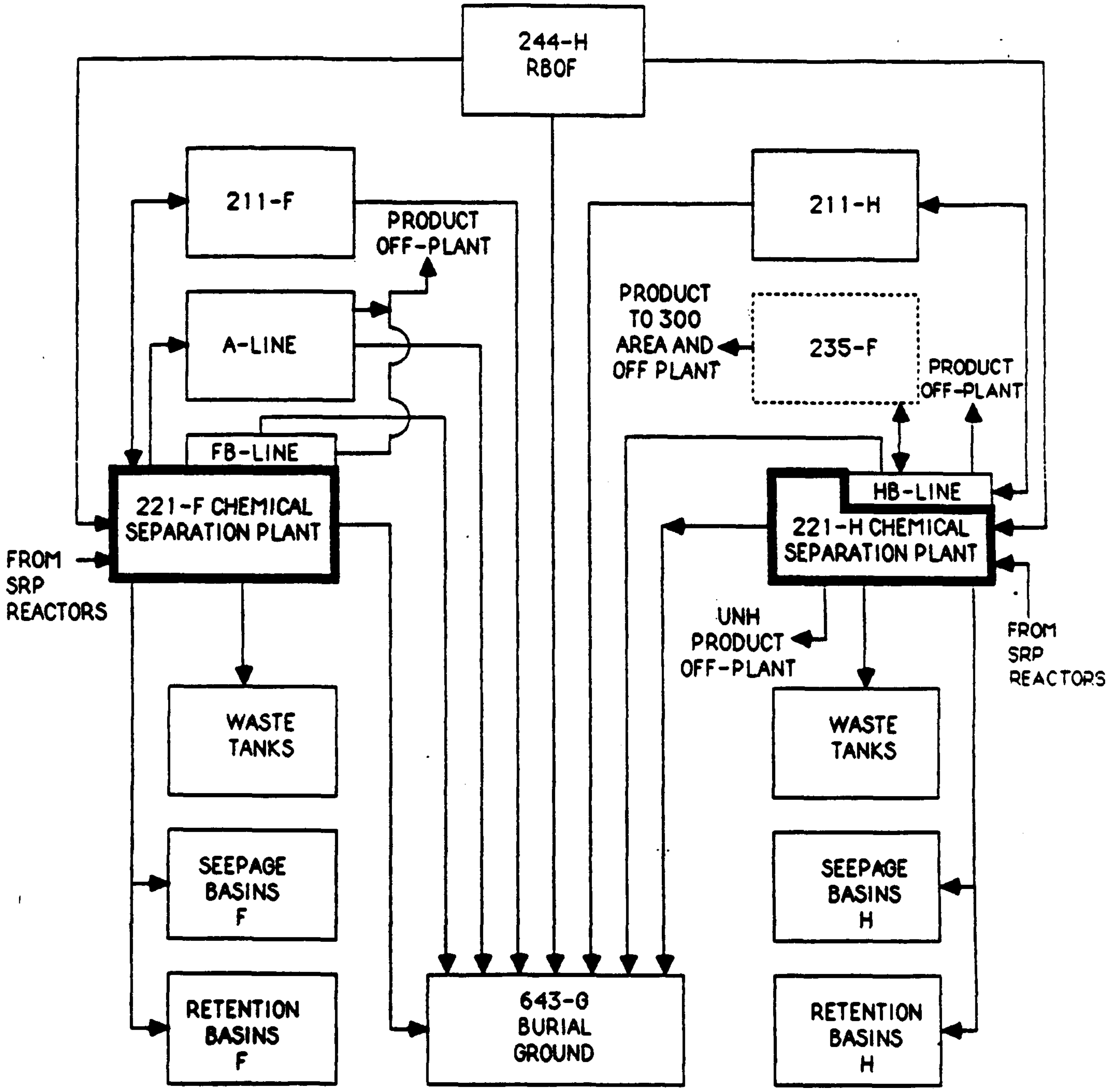

FIGURE 1-1. Flow of Radioactive Materials in the 200 Separations Area

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 to supersede the 1977 version, and several improvements have been made to the H-Canyon equipment.

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 H-Canyon. The H-Canyon facility and process design are described in Section 3 and a description of H-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 H-Canyon facilities. The H-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 H-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 H-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 H-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 H-Canyon operations important to safety is presented followed by a summary of H-Canyon safety analysis including tables of radiological consequences and risks for accidents and normal operation.

1.2.1 Facility Operations

The 221-H chemical separations plant has a large shielded "canyon" building for processing irradiated materials. Irradiated fuel and target materials are dissolved in nitric acid. Product materials are separated by using solvent extraction and ion exchange processes. The resulting solutions contain the various products that have been decontaminated from fission products. Further processing is performed in unshielded facilities where some products are 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 H-Canyon structure and supporting engineered safety features constitute an effective radiological confinement system designed to control and limit radionuclide release to the environment. For the case of criticality, canyon equipment by and large is not designed to geometrically preclude criticality. Hence, administrative and procedural control is the principle mechanism for controlling criticality. Such an approach has historically been effective (no criticalities have been experienced during operations of the H-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 H-Canyon are summarized in Table 1-1. The total calculated risk to the offsite maximum individual is only 4.0 x 10^{-4} 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 being 3.8 person-rem/yr for the offsite population group to a radius of 50 miles and 0.71 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 200 Separations 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 H-Canyon operations during 1984 was 44 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 H-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 H-Canyon operations indicate that the facility can be operated without undue risk to the public or operating personnel.

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

Accident		Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
Natural Phenomena		3.6E-05	1.3E-05	7.5E-09
Externally Induced Failures		1.2E-02	2.2E-03	1.5E-06
Process Related Occurrences				
Medium Energetic (Airborne)	Fire	7.7E-01	1.4E-01	8.0E-05
	Uncontrolled Reaction	4.7E-01	8.7E-02	5.0E-05
	Explosion	8.2E-03	1.5E-03	8.5E-07
	Criticality	4.8E-03	1.1E-02	4.3E-06
	Total (Airborne)	1.3E+00	2.4E-01	1.3E-04
Low Energetic (Airborne)	Transfer Error to 211-H	2.6E-02	9.8E-03	4.7E-06
	Transfer Error to Sump	1.9E-02	3.5E-03	2.0E-06
	Overflow to Sump	3.7E-01	6.9E-02	3.9E-05
	Leak to Sump	2.1E+00	3.8E-01	2.1E-04
	Processing Short-Cooled Fuels	3.6E-03	6.1E-04	5.5E-07
	Ruthenium Volatilization	1.2E-03	1.7E-04	2.3E-07
	Total (Airborne)	2.5E+00	4.6E-01	2.6E-04
Coil Failure (Liquid)		7.0E-02		5.9E-06
Total (Airborne)		3.7E+00	7.1E-01	4.0E-04
Total (Process Occurrences)		3.8E+00	7.1E-01	4.0E-04
Total for all accidents	Airborne	3.7E+00	7.1E-01	4.0E-04
	Liquid	7.0E-02		5.9E-06
	Total	3.8E+00	7.1E-01	4.0E-04

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

Pathways	Risks Due to Accidents: Natural Phenomena	Externally Induced Failures	Process Operations	Total	Risks Due to Normal Operation, person-rem/yr*: Annual Dose, 1984
Airborne Release to Offsite Population, Person-rem/yr	3.6×10^{-5}	1.2×10^{-2}	3.7×10^{0}	3.7×10^{0}	75
Airborne Release to Onsite Population Person-rem/yr	1.3×10^{-5}	2.2×10^{-3}	7.1×10^{-1}	7.1×10^{-1}	--
Airborne Release to Offsite Maximum Individual, rem/yr	7.5×10^{-9}	1.5×10^{-6}	4.0×10^{-4}	4.0×10^{-4}	--
Liquid Releases to Offsite Population, Person-rem/yr**	--	--	7.0×10^{-2}	7.0×10^{-2}	1.1
Liquid Releases to Offsite Maximum Individual, rem/yr**	--	--	5.9×10^{-6}	5.9×10^{-6}	--

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

**Direct release to Four Mile Creek.

1.3 REFERENCES

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2. SR Implementation Plans for DOE Order 5480.1A. Department of Energy, Savannah River Operations Office, Aiken, SC, July 1982.

3. Durant, W. S. and Perkins, W. C. Systems Analysis -200 Area Savannah River Plant H-Canyon Operations. Internal Report DPSTSY-200-1H, Volume 1, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, SC, October 1983.

4. Durant, W. S. and Perkins, W. C. Systems Analysis -200 Area Savannah River Plant H-Canyon Operations. Internal Report DPSTSY-200-1H, Volume 2, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, SC, October 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 H-Canyon is located in H-Area and noted as Building 221-H, 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 H-Canyon specifically.

2.1.1 Population Characteristics

The potentially affected population for any H-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
Savannah, GA	100
Charleston, SC	100
Columbia, SC	60
Augusta, GA	25
Aiken, SC	20
Williston, SC	15
Barnwell, SC	15
Jackson, SC	12

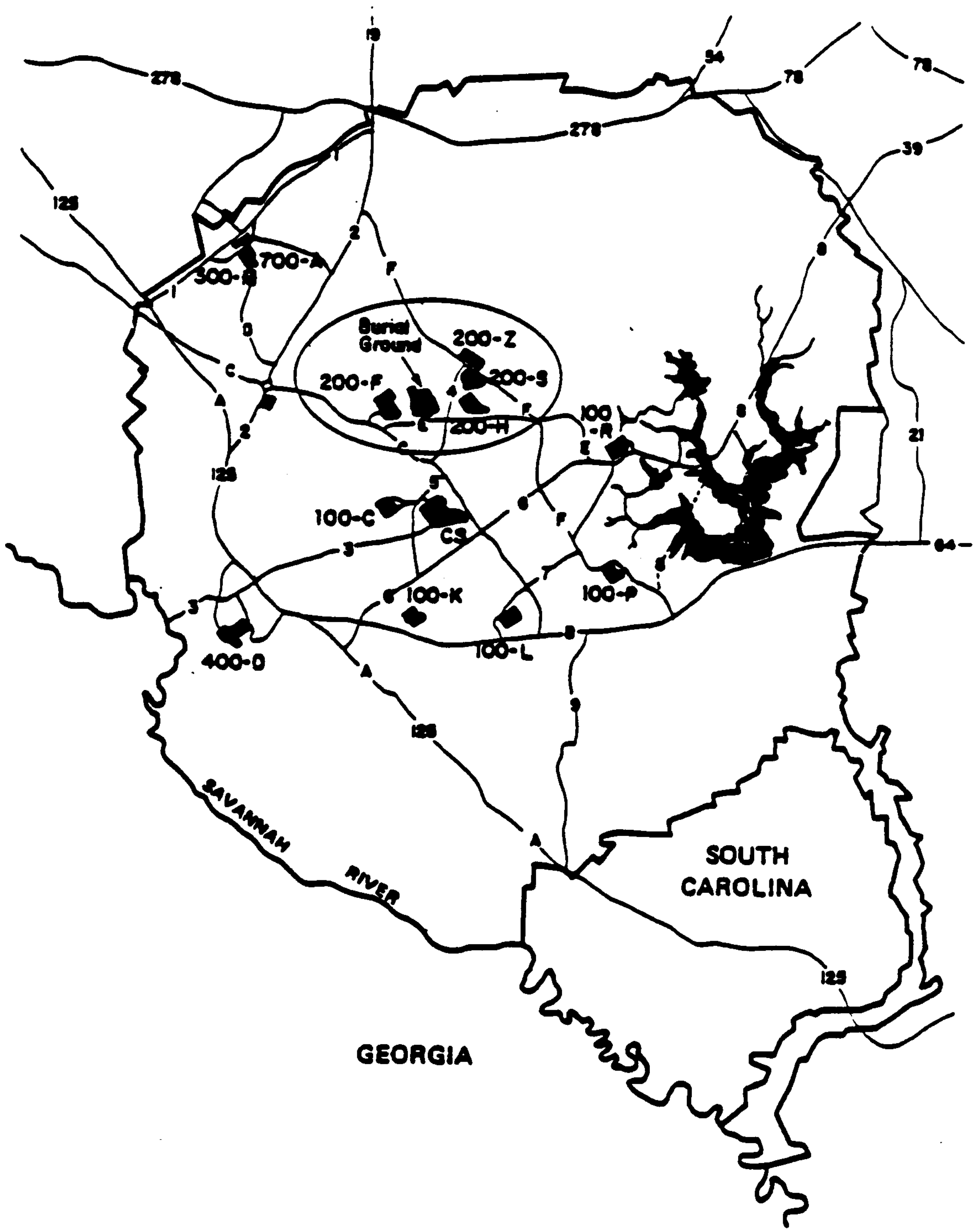


FIGURE 2-1. The Savannah River Plant Site

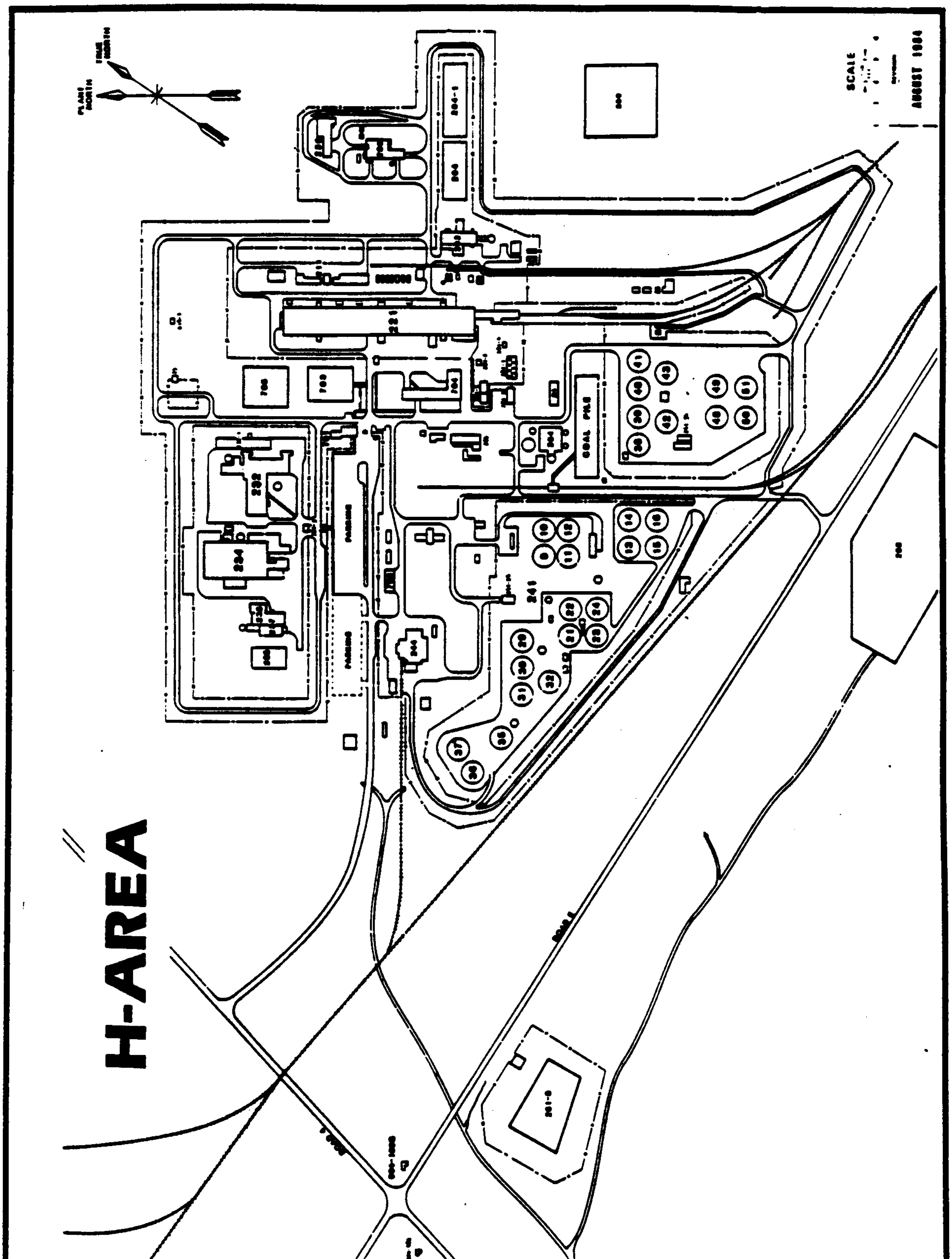


FIGURE 2-2. Plot Plan, 200 H-Area

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	848
Total	15,737

TABLE 2-3. 1980 Population Counts

Counties			
South Carolina		Georgia	
County	1980 Population	County	1980 Population
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

City	Distance (miles)	Direction from Plant	1980 Population
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

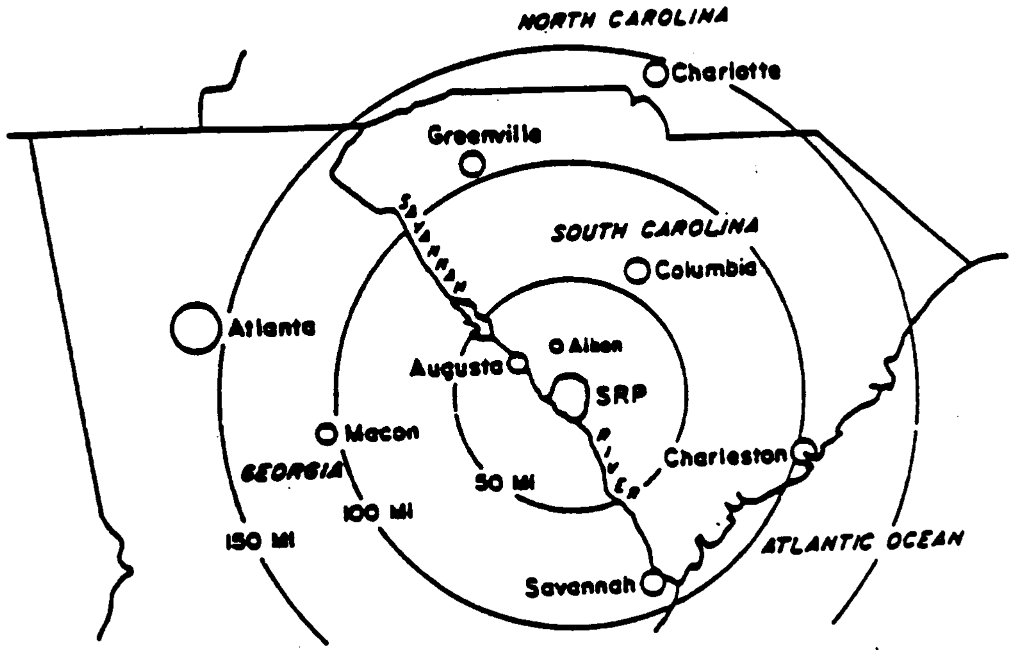


FIGURE 2-3. Location of SRP Relative to Surrounding Population Centers

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.

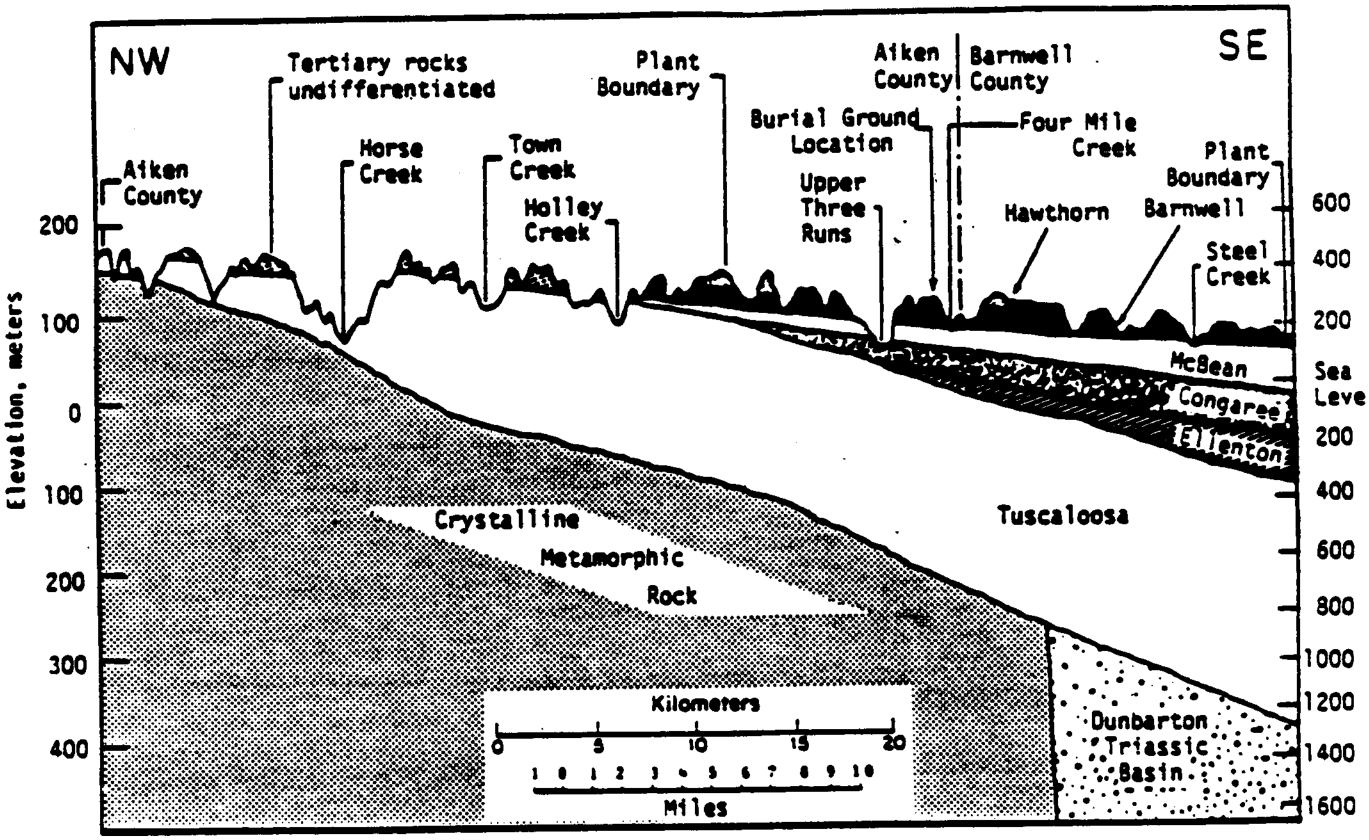


FIGURE 2-4. Profile of Geologic Formations Beneath the SRP

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 H-Canyon building uranium and ^{237}Np are recovered by a modified Purex solvent extraction process. ^{237}Np and ^{238}Pu are recovered from irradiated Np targets by ion exchange. Irradiated targets and fuel tubes are received from the reactors areas, and aqueous streams of enriched uranium, neptunium, and ^{238}Pu are transferred offsite or to the HB-Line.

The unit operations in the H-Canyon for reactor fuel are shown in Figure 2-5 and include receipt of reactor fuel, dissolving, head end, solvent extraction, evaporation, and waste disposal. A more complete discussion of H-Canyon process operations is included in Section 3 and the H-Canyon System Analysis (2).

2.2.1 <u>The H-Canyon Building 221-H</u>

The 221-H 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 materials and low activity materials and are designated the hot canyon and the warm canyon, respectively.

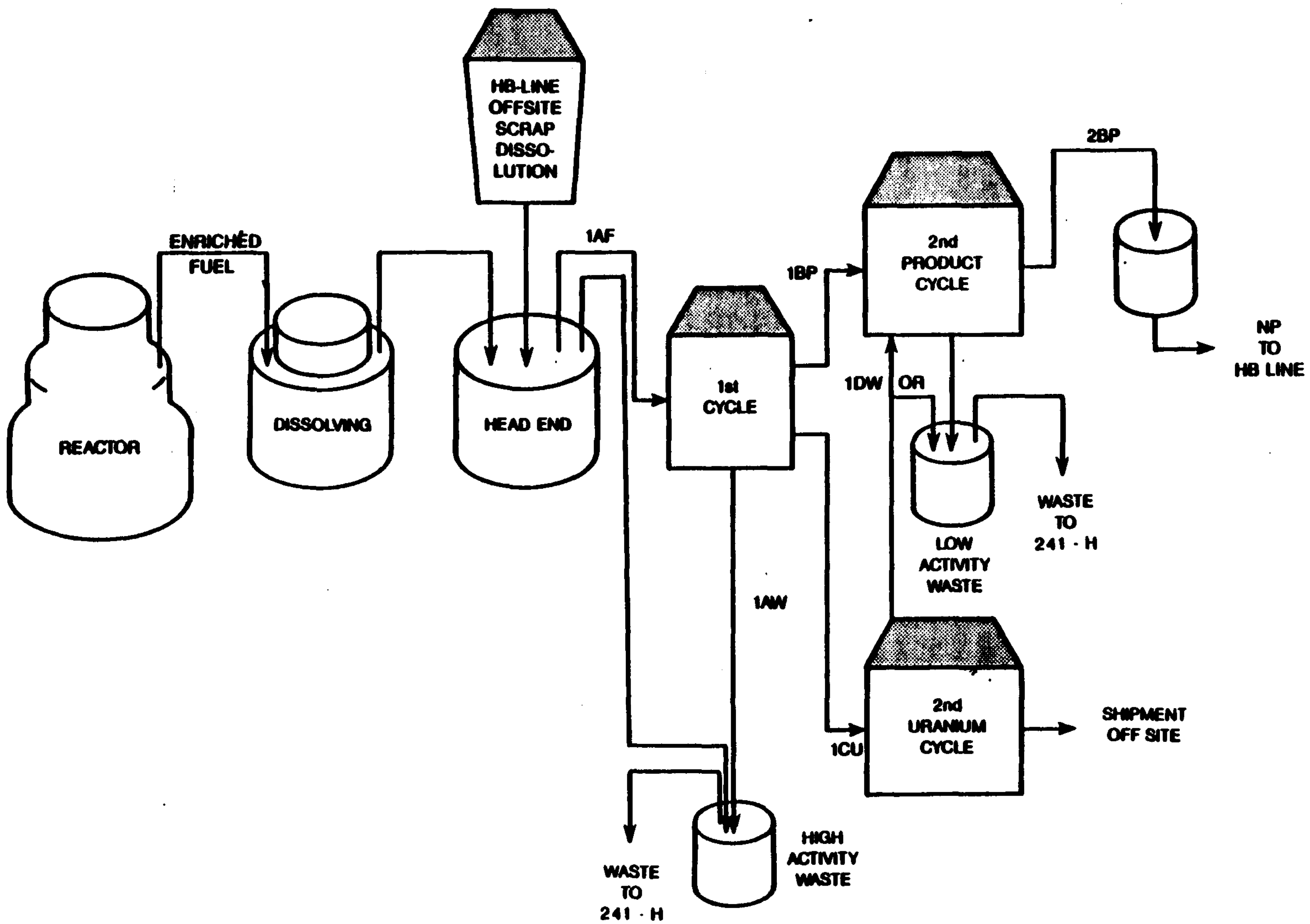


FIGURE 2-5. H-Canyon Flowsheet

2.2.2 Uranium Cycle

The standard enriched uranium fuel is a uranium-aluminum alloy, clad in aluminum, and the whole fuel piece is dissolved in nitric acid catalyzed by mercuric nitrate. The off-gases go through a silver nitrate reactor to reduce iodine emissions. The dissolved fuel is clarified in head end, and the adjusted solution is fed to the solvent extraction system. Uranium and neptunium are extracted from the fission products and separated from each other in the first cycle; then each goes through two separate cycles of solvent extraction for further purification. The uranium product solution is not concentrated further, but is transferred out of the building, loaded into tank trucks, and sent offsite for further processing. The neptunium product solution is transferred to the HB-Line for conversion to oxide as explained below.

2.2.3 Neptunium Cycle

In a separate process system in the shielded canyon, irradiated neptunium targets are dissolved in nitric acid, and ^{238}Pu and neptunium are separated from fission products and each other by a series of anion exchange resin columns. The product solutions of plutonium and neptunium are transferred to the finishing area where the two are concentrated, precipitated as oxalate, and calcined to oxides. The plutonium oxide is packaged for shipment offsite or sent to the metallurgical building (235-F) for formation into heat sources. The neptunium oxide is sent to the metallurgical building (235-F.) for refabrication into a billet to be made into reactor target elements.

2.2.4 Special Programs

In addition to these main process efforts, various parts of the system have been used for special programs. For example, highly irradiated plutonium has been processed in both the solvent extraction and canyon anion exchange systems and finally produced as oxide from the finishing area. A plutonium isotope mixture containing appreciable ^{238}Pu has been recovered by solvent extraction from the fission product waste stream from enriched uranium processing. Irradiated thorium has been processed in the solvent extraction system to recover ^{233}U. A purified, concentrated thorium nitrate solution has been shipped offsite for reuse, and the uranium fraction from solvent extraction has been concentrated by cation exchange in the finishing area, precipitated as ammonium diuranate, calcined to the oxide, and shipped offsite.

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. and Perkins, W. C. Systems Analysis - 200 Area Savannah River Plant H-Canyon Operations. Internal Report DPSTSY-200-1H, Volumes 1 and 2, E. I. du Pont de Nemours and Company, Savannah River Laboratory, Aiken, SC, October 1983.

3.0 FACILITY AND PROCESS DESIGN

3.1 DESIGN CRITERIA

The 221-H Canyon and processes were designed in accordance with applicable Du Pont Engineering Standards in effect at the time. The process has been modified since the facilities were originally constructed but the structure and process equipment remain the same.

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

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 5480.1.A and 6430.1.

3.2 PROCESS AND FACILITY DESCRIPTION

3.2.1 Summary Description

The H-Canyon building is a radiochemical plant originally designed for the separation and recovery of ^{239}Pu and ^{238}U from irradiated natural uranium by the Purex process. The plant mission was changed in 1959 to the processing of irradiated enriched uranium to recover uranium with ^{235}U content of 1.1% to 94% ^{235}U. Subsequently, additional equipment was installed to process irradiated Np targets to separate and recover ^{238}Pu and ^{237}Np. From 1964 to 1970, the plant periodically processed irradiated thorium to recover ^{233}U. 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 by a central operating and service section that is divided into four levels as shown in Figure 3-1:

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, maintenance facilities, and cold feed preparation equipment.

The more highly radioactive processing operations such as dissolution of irradiated materials, 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, 122 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 West 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 H-Canyon Systems Analysis (2,3).

3.2.2 Process Design Considerations

The 221-H Building isolates the radiological hazards 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 H-Canyon processing cells by the radiological shielding walls and by engineered features that minimize releases through ventilation air service, instrument lines and cooling water, and steam condensate discharge lines. These features are designed to remain intact in the event of any forseeable accident or natural phenomena such as earthquakes or tornadoes. The building (Figure 3-2) is a Class I explosion-proof Maximum Resistance Construction (MRC) structure designed to withstand pressures 1000 lb/ft^2. 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 34 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.

FIGURE 3-1. Building Isometric Sections

REVIEWING OFFICIAL J M Bauer

"DELETED VERSION"

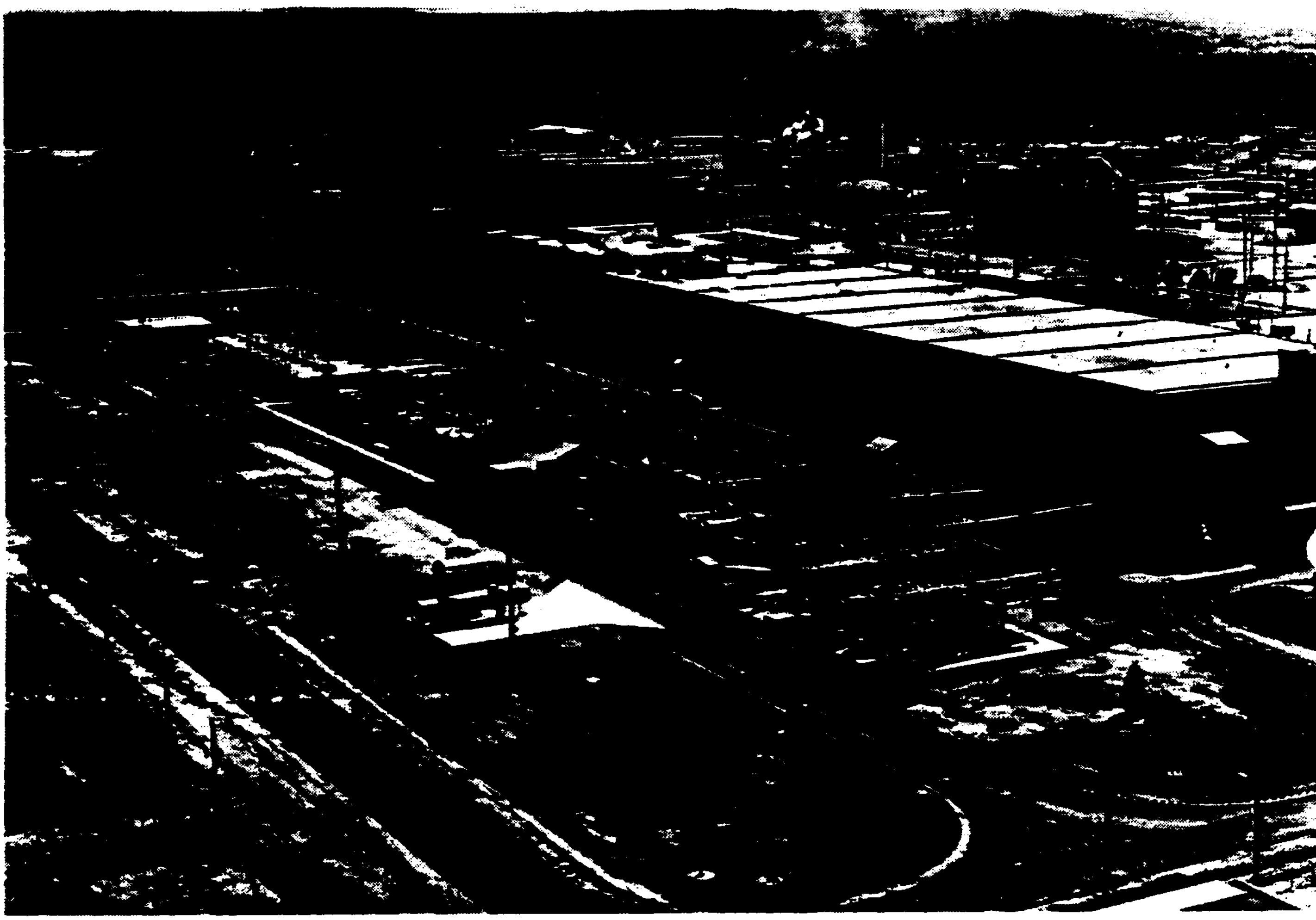


FIGURE 3-2. Building 221-H

The H-Canyon building is assumed to be insignificantly affected by an earthquake of less than VIII intensity on the Modified Mercalli (MM) Intensity Scale* of 1931. This assumption has been independently supported by studies 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 been 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 roll up 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 H-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 in a shielded cask on a railcar. The cask of irradiated material is prepared for unloading in the airlock, then pushed into the 221-H 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 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. Nonradioactive services, with the exception of steam supply for solution transfer jets, are supplied through the inside shielding walls from the central operating area. Steam for the motive force for the liquid transfer eductor (jet) is 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 cranes to make connections between the process equipment in the canyons with services or with radioactive transfer lines that are connected to 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,

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

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 the nuclear service. The connector consists of:

- The tappered male nozzle with flat sealing surface, is part of the fixed embedded piping in the wall, or stubbed-off on the vessel.
- 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 on 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 which is carried by the crane is positioned by the crane on the clamp drive nut. Positioning is accomplished remotely by the crane operator viewing the operation through a periscope type optic system 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 vertical or 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 sidewall nozzles for cold feed, cooling water, electricity, lubrication, steam, and instrumentation, 2) the horizontal nozzles on the hot side access cooling water outlet, the waste and vent headers, 3) low vertical connectors on the hot side that access steam supply from the jet gang valve corridor for the transfer jet and steam condensate from evaporators, etc. to steam traps, 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 sections.

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

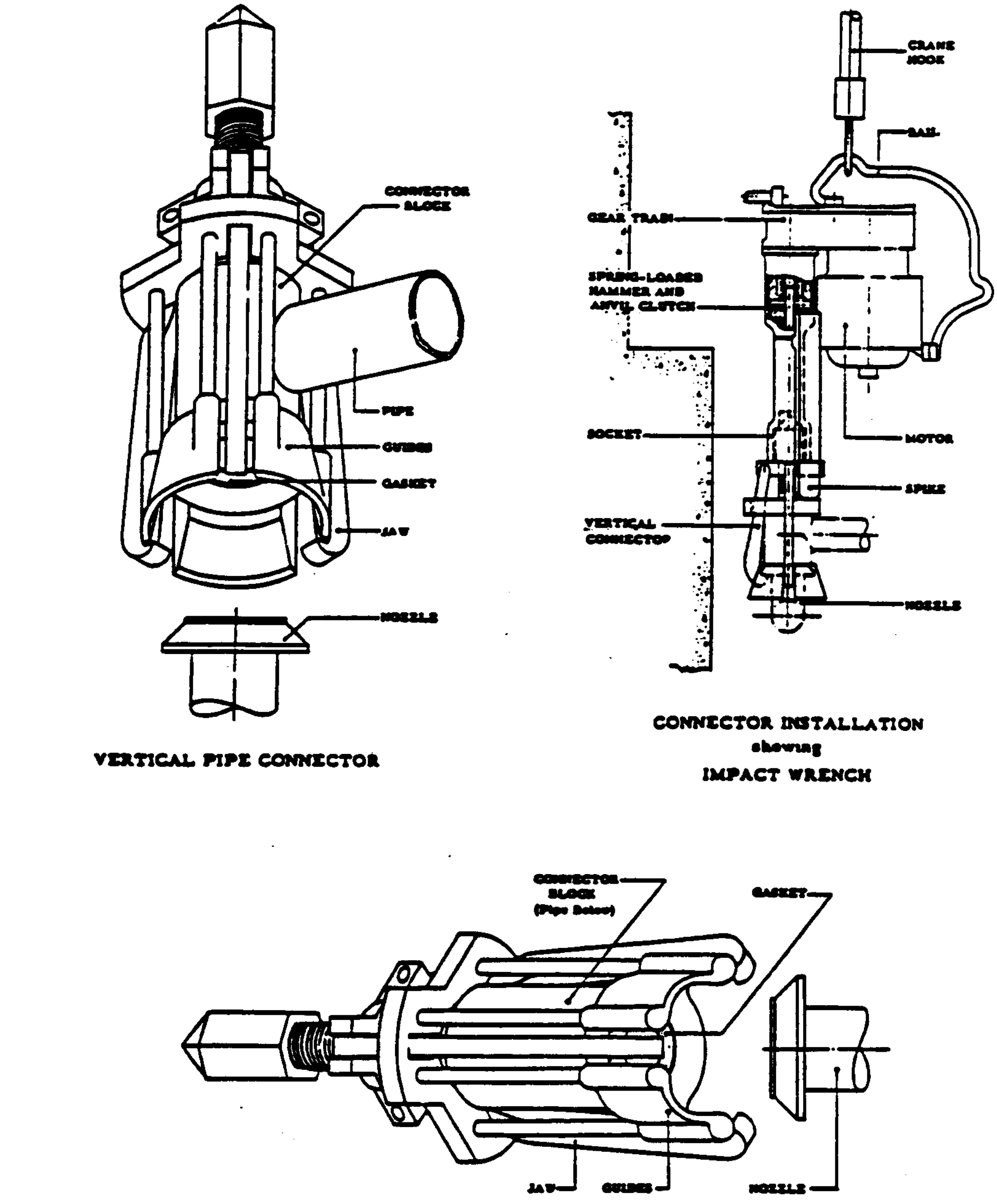


FIGURE 3-3. Hanford Connector

3.2.2.1 Receipt and Charging of Irradiated Material

Irradiated materials in the form of Np target tubes or spent reactor fuel tubes are received in bundles at Building 221-H in transfer casks mounted on special railroad cars.

Irradiated fuel is self-heating due to the fission products it contains. To maintain the integrity of the cladding, the bundles and targets are shipped in water filled cask wells. Bundles and targets received at H-Canyon are submerged in water in the target and bundle storage section or charged into a dissolver with sufficient solution to cover the irradiated material preventing excessive self-heating. Time out of water during transfer of bundles by the crane, including time to drain, is specified administratively. Fuel is kept under water at the reactors for a period of time to allow some fission product decay to prevent melting during the transfer in air from the cask to the dissolver.

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 45 days is required for Mark 53 neptunium targets. Other irradiated assemblies are required to have a minimum cooling time specified as a function of assembly exposure and exposure time. The timing of fuel shipments to H-Canyon is regulated at the reactor. Radiation monitors set to alarm upon the detection of abnormal radiation from short-lived radioisotopes, indicating the presence of fuel insufficiently aged, are part of the preparation of the irradiated fuel for shipment.

Correct identification of material to be charged is the responsibility of both the control room supervisor and the crane process operator (CPO). The supervisor verifies that the serial numbers and the bundle identification listed in the shipping papers agree with the information listed in the batching memorandum. The CPO verifies the bundle identification stenciled on the side of each bundle before he engages the hoist hook in the bundle bail. After each dissolver slot is charged, the CPO records the time charged, the type of fuel, the bundle identification number, and the storage position from which the bundle was removed.

A nuclear safety control requirement allows only one bundle be moved at any given time. All other bundles must be in either a cask car, bundle storage, or the dissolver. Bundles are carefully engaged, one at a time, and lifted from the cask car water well with a special transport hook.

Fuel requiring dissolution in the electrolytic dissolver because of the variety has no standard fuel container (bundle) in which these fuels (or scrap) are shipped to the 221-H Building. The type of shipping container is dictated by fuel type and dissolver chute geometry. The shipping cask inventory and container arrangement are dictated by a nuclear safety analysis for each type of fuel to be processed. Each container must be properly identified as to number, fuel type (or scrap type), stainless steel or Zircaloy and aluminum content, total uranium, and ^{235}U content before loading in the shipping cask. The fuel is transported to 221-H from Receiving Basin for Off-Site Fuels (RBOF) and charged directly to the dissolver in a manner analogous to the charging of uranium-aluminum alloy fuel to the chemical dissolver (Reference 2, Section 1.4.2.1.1). Fuel types may be mixed in a fuel shipment or dissolver charge if rules and limits of operation are based on the type of fuel that is most restrictive.

Double batching a dissolver, an error that compromises critical mass safety, is avoided by the use of dissolver inserts with space for only one bundle per compartment and by administrative controls. The two chemical dissolvers are prepared for receipt of fuel by fitting an insert in the dissolver. The insert (Figure 3-4) is perforated at the bottom and sides to permit free flow of solution around the tubes, discharge of foam to the dissolver proper, and the discharge of off-gas to the column condensers. For nuclear safety, one or more of the insert compartments may be blocked to limit the number of bundles that may be loaded for a given charge. Some fuel fragments may remain in the dissolver insert after the dissolution of all fuel is normally complete. Before the dissolved fuel is transferred from the dissolver, the crane operator verifies that an excessive quantity of fuel is absent from the dissolver insert by probing. The absence of excess metal heel ensures that the fresh charge can be inserted without mechanical difficulty, and without exceeding permissible fissile material loading. The fragment height is measured with a probe (Figure 3-5) before subsequent charging and must not exceed a predetermined height in any compartment.

The dissolver column insert limits the maximum number of fuel tubes that can be charged. This fixes the quantity of fissile material present for each dissolution, and with the minimum possible volume of dissolver solution, defines the maximum fissile material concentration in the dissolver solution. The dissolver column insert confines the undissolved fuel tubes to a fixed configuration.

Fuels containing components that cannot be dissolved chemically in nitric acid are charged in the electrolytic dissolver (6.3D) shown in Figure 3-6. The electrolytic dissolver vessel is an 8-ft high by 8-ft in diameter, jacketed pot.

Since the electrolytic dissolver, like the chemical dissolver, is not geometrically safe for enriched fuel, rigid administrative controls limit the mass of fissile material in the dissolver chute and in the dissolver. A dissolver insert is also used to maintain a safe geometry of whole or partially dissolved fuel. The electrolytic dissolver has a trash basket to maintain critically safe geometry throughout dissolution, to retain loose segments of fuel at the end of dissolution, and to retain insoluble foreign material present in the fuel.

In addition, a nuclear safety analysis is prepared for every fuel type prior to dissolving the fuel. Credible accidents that may occur during loading or unloading operations as well as during shipping, are considered in the nuclear safety analysis.

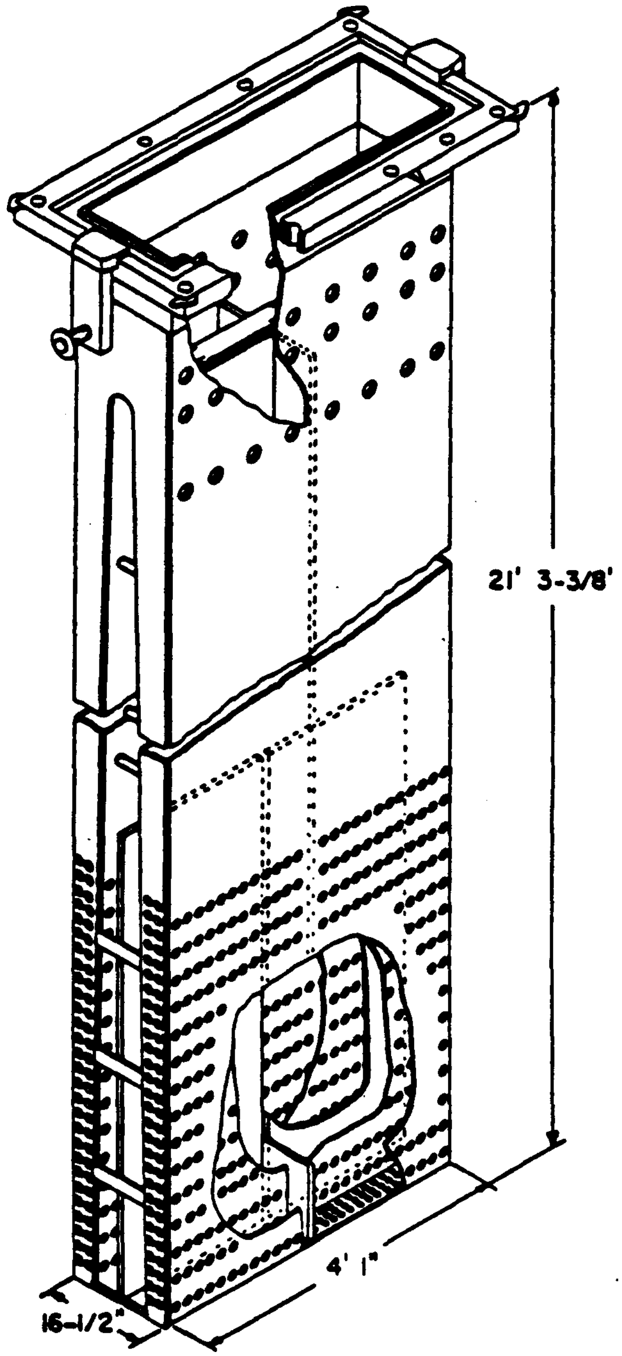


FIGURE 3-4. Dissolver Insert for SRP Fuels

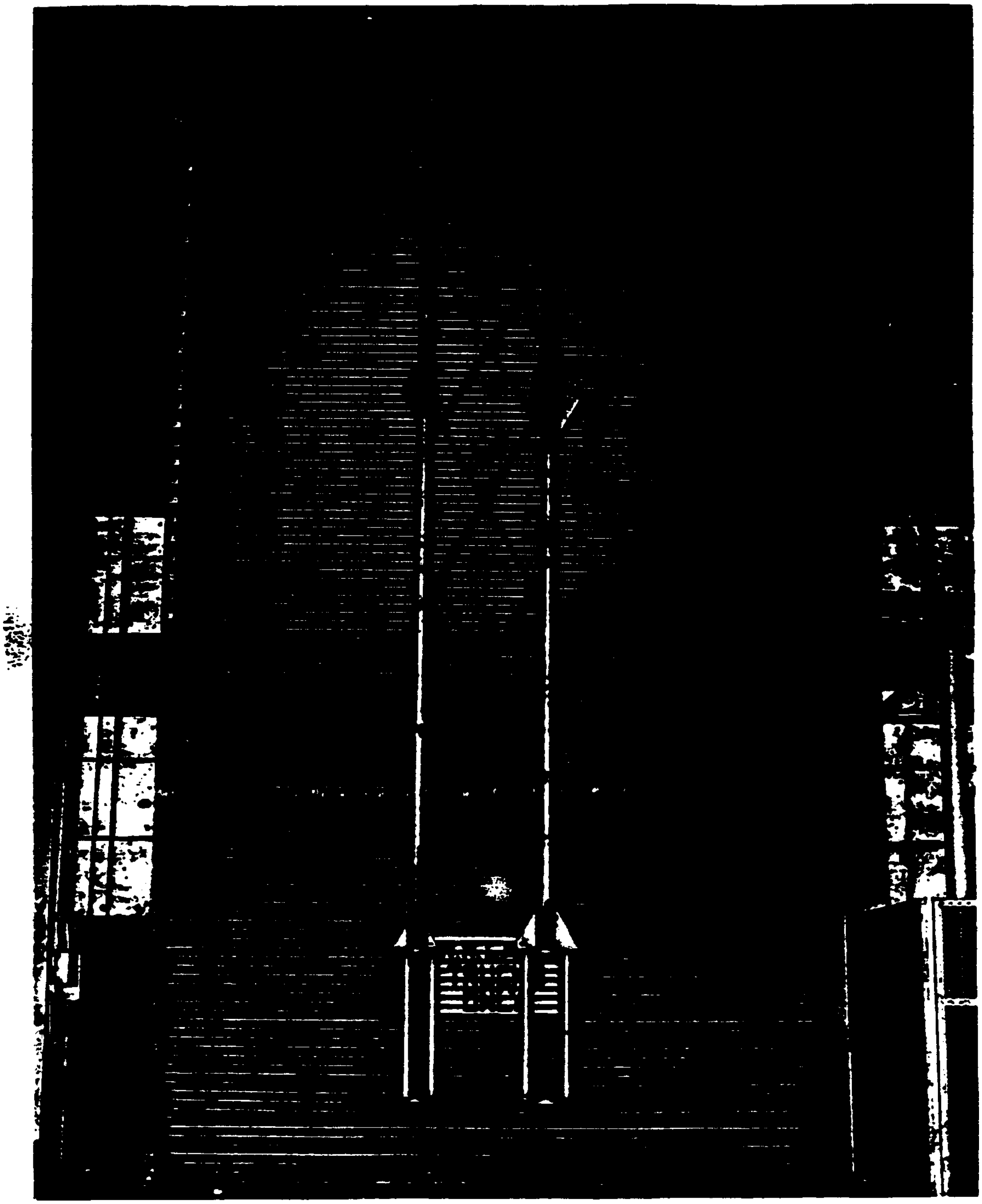


FIGURE 3-5. HFIR Dissolver Insert Probe

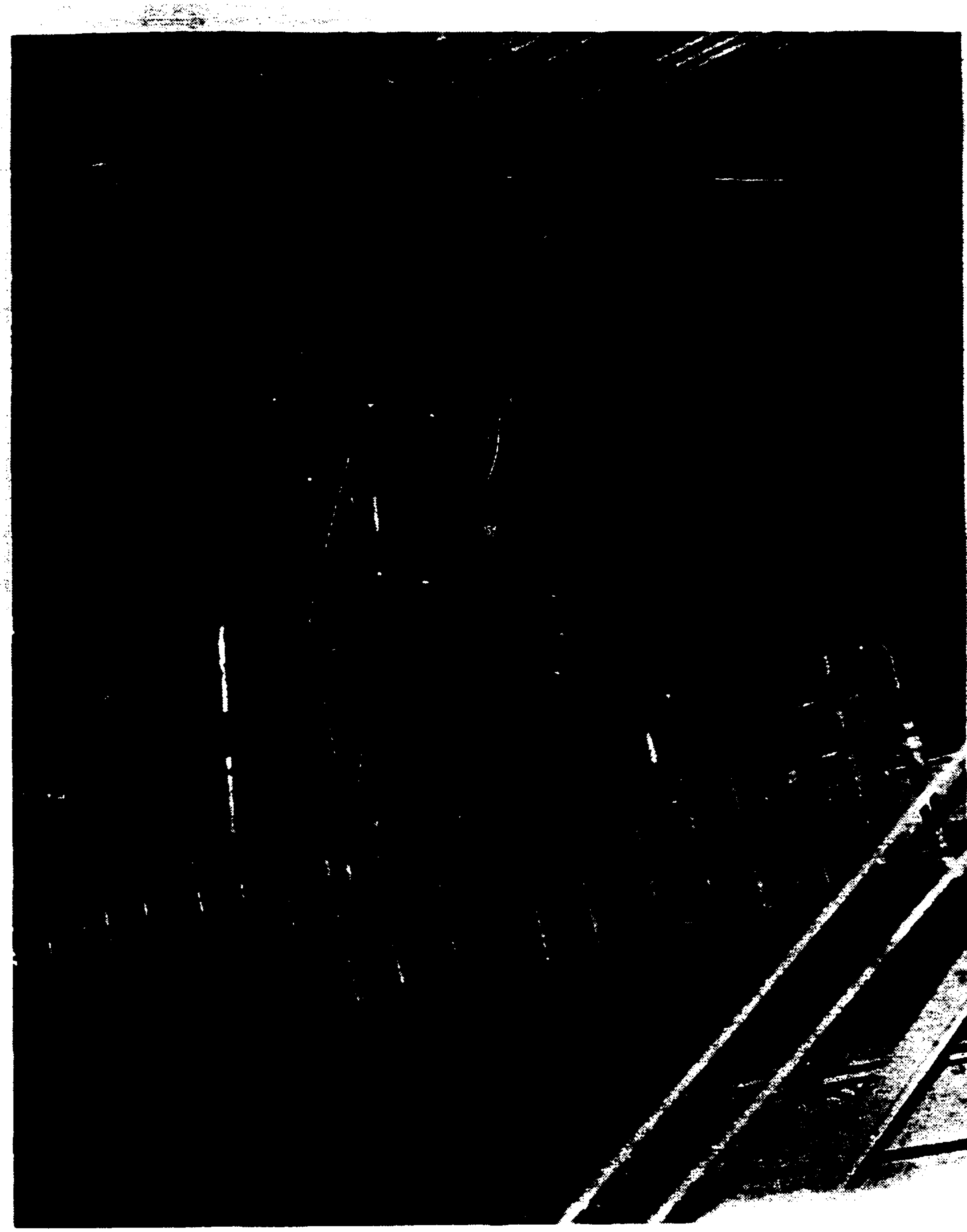


FIGURE 3-6. 6.3D Electrolytic Dissolver

3.2.2.2 Dissolution

The enriched fuel elements and aluminum components of a bundle are dissolved in the chemical dissolvers in a solution of boiling nitric acid, catalyzed by mercuric nitrate. The two chemical dissolvers are similar in construction. Currently dissolver 6.1D is 8 ft in diameter by 8 ft high, and dissolver 6.4D is 11.5 ft in diameter by 8 ft high. Each dissolver is equipped with a rectangular column and two parallel downdraft condensers. The rectangular column accepts a perforated insert that has slots or wells into which fuel bundles are charged. The solution is heated to the boiling point by applying steam to the upper coils in the dissolver pot. Mercuric nitrate catalyst is added and as the dissolution proceeds, the concentration of aluminum nitrate in solution increases, and the rate of reaction decreases.

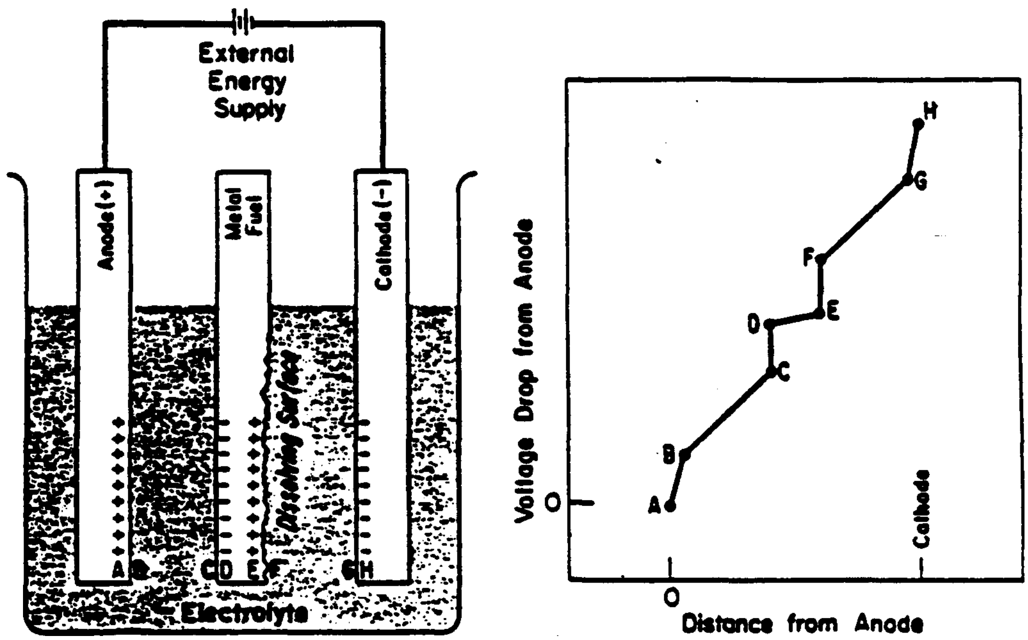
Stainless steel, zirconium, and aluminum cladding are dissolved anodically in the electrolytic dissolver; the core material dissolves chemically, or in the case of cermet cores, both anodically and chemically. The use of nitric acid as the electrolyte permits recovery of fissile materials by conventional solvent extraction flowsheets using existing stainless steel equipment.

Thermodynamically, nitric acid should dissolve many of the candidate fuel claddings, but in most cases, the metal surface is protected by oxide films. When the electrochemical potential at a metal-solution interface is increased by applying an external source of electrical energy, passive films can be destroyed. When directly connected to the electrical supply as the anode, stainless steel electrochemically dissolves in nitric acid at the theoretical rate of 0.59 g/(amp-hr); zirconium and aluminum are dissolved at 0.85 and 0.33 g/(amp-hr), respectively.

Because direct electrical connections with the fuel are difficult to maintain in a remotely operated production facility, contact is established through the electrolyte with the dissolving fuel suspended between two electrodes. The liquid-contact principle is shown schematically in Figure 3-7. As it passes out of the anode into the electrolyte and enters the adjacent surfaces of the fuel, the current produces oxygen by one of two competing cathodic reactions: either the reduction of nitrate ion, or the formation of hydrogen gas from water. The surface of the fuel adjacent to the cathode becomes anodic by association and dissolves as the current passes out of it and back into the electrolyte. The current is accepted at the cathode, and nitrogen oxides and/or hydrogen are produced.

Almost all of the power required to operate the cell is dissipated as heat in the nitric acid electrolyte. If the heat is not removed, the acid will boil and blanket the electrode with vapor, which causes a rapid increase in resistance. Also, heating will accelerate corrosion of electrodes.

For criticality control, mechanisms that concentrate the fissile material in either the chemical or electrolytic dissolver must be precluded. Normally, all water vapor is condensed and returned to the pot. If the condenser should fail, the minimum volume is controlled by the design of the heating coil. Steam is supplied only to the upper pot coils, the lower coil of which is 24 in above the bottom of the dissolver pot. The minimum volume of dissolver solution possible by evaporation with the steam coils is defined by the level at which the upper coils come out of solution. With a normal dissolver



VOLTAGE DISTRIBUTION

A to B - Reaction potential at the surface of the anode
B to C - Voltage drop across the electrolyte
C to D - Reaction potential at the cathode surface of the fuel
D to E - Voltage drop across metal fuel
E to F - Reaction potential at the anodic surface of the fuel
F to G - Voltage drop across the electrolyte
G to H - Reaction potential at the surface of the cathode

FIGURE 3-7. Liquid Contact Principle

charge, it is impossible to reduce the volume of the solution to a point where the fissile material concentration becomes critical, since the maximum concentration in this volume is less than the critical concentration (8). This minimum volume could represent a threefold concentration of the dissolver solution. In addition, the dissolver column insert (which contains the undissolved fuel tubes) extends into the dissolver pot to within 24 in of the bottom of the pot. In the event the dissolver solution was evaporated to its minimum volume, the fuel tubes would no longer be submerged in the liquid, and no further dissolving would take place.

For the electrolytic dissolver, the volume of the dissolvent is selected to limit the final concentration of fissile material in solution.

The other mechanism for concentration is precipitation. Administrative controls for the dissolution of enriched uranium require that a minimum of 0.3M HNO_3 be present to avoid hydrolysis and precipitation of fissile material. Accidental addition of a precipitating chemical such as NaOH is prevented by a combination of specially designed nuclear safety blanks (NSB) in process lines from certain cold feed tanks and by administrative control. No NaOH is piped directly to a dissolver.

3.2.2.3 Neptunium Aluminum Targets

The procedure for charging NpO_2-Al target bundles to the 4-ft 2-in diameter by 15-ft 6-in tall frame dissolver is similar to the procedure used for charging uranium fuel. Each bundle contains one to four targets.

The dissolver is supplied with 25 psig steam for heating and with air sparge for agitating the dissolver contents.

The usual dissolver instrumentation has been provided, including temperature measurements in the dissolver pot and the condenser off-gas (both of which are remotely removable), liquid level and specific gravity probes in the dissolver pot, and condenser pressure drop taps. Off-gases are vented through the process vessel vent system.

The bundle is dissolved in boiling nitric acid catalyzed by mercuric nitrate for the dissolution of the aluminum and aided by fluoride for the dissolution of the actinide oxides.

Off-gases are vented through the process vent system. Iodine releases are limited by cooling the targets a minimum of 45 days which reduces I_2 by a factor of 1.73×10^{-4}.

3.2.2.4 Off-Gas

Each chemical dissolver is equipped with two parallel downdraft condensers. Ports are provided in the top of the dissolver insert to allow off-gas to flow into the rectangular annulus between the insert and the column wall. The off-gases liberated from dissolution containing NO, N_2, N_2O and H_2, plus the fission products iodine, krypton, and xenon, flow from the column into the two downdraft condensers operated in parallel. The off-gas and water condensate flow downward concurrently, the coolest condensate in contact with the coolest gas for the most efficient absorption and conversion of NO_2 to HNO_3 for maximum utilization of acid during the dissolution.

The condensate of weak nitric acid is returned to the dissolver pot. The non-condensable and uncondensable gases, N_2, NO, N_2O, H_2, and rare gases are vented.

The temperature of the off-gas leaving the condenser is maintained below 60°C so that all condensable vapors are condensed. Iodine escaping the dissolver is removed by absorption in the iodine reactor, which contains Berl saddles coated with silver nitrate. Iodine absorption would be incomplete at temperatures below 170°C, and silver nitrate melts at 212°C, so the iodine reactor is maintained in the temperature range of 170°C to 200°C by heating with steam at 140 psig (185°C).

A "Fiberglas" filter downstream of the silver reactor is designed to collect any particulate matter. This filter is heated with 15 psig steam to avoid pluggage of the filter by water condensate.

The off-gas from the electrolytic dissolver is primarily air from the air spargers that are used for solution agitation and circulation in the electrode cavity.

The off-gas passes through a single reflux condenser and is vented through one of the chemical dissolvers to the heated iodine reactor and "Fiberglas" filter of the chemical dissolver off-gas system before discharge to the 291-H stack. The iodine reactor and filter are common to both the chemical dissolver and the electrolytic dissolver.

In the dissolution of fuel bundles by mercury catalyzed dissolution of uranium and aluminum at low acid (0.1M HNO_3 vs. the minimum limit of 0.3M) hydrogen concentration of up to 10% can be produced from the following reaction:

$$2Al + 6HNO_3 \dashrightarrow 2Al(NO_3)_3 + 3H_2 \tag{9}$$

Hydrogen-air mixtures are flammable in the range of 4.1 to 75 volume percent in dry air, if an ignition source were present, and can explode at room temperature over a more limited concentration range (9). An external source of ignition is required, such as sparks from electrical equipment, overheated bearings, or an open flame.

Hydrogen evolution at concentrations in the flammable range could be ignited in the dissolver or in the silver reactor. The latter incident could release a large fraction of the ^{131}I and ^{129}I absorbed as AgI and $AgIO_3$ (9). Maintaining sufficient acid above 0.1M and an adequate air purge rate to dilute the hydrogen prevents this from occurring.

The initial nitric acid concentration specified for the dissolution corresponds to a specific gravity that is checked by dissolver operator observing the specific gravity instrumentation on the dissolver as the acid is added. A minimum acidity of .3M at the end of the dissolution is required (9). A minimum air purge rate of 37 cfm is also required to dilute the hydrogen concentration to the acceptable level (9).

3.2.2.5 Head End

The head end process clarifies, concentrates, and decontaminates the raw metal solution from some of the Zr and Nb for subsequent solvent extraction processing. The head end process may include a concentration step, a "strike" or precipitation step, and centrifugation to remove the precipitate, cake washing, and cake disposal. Several types of strikes can be performed:

- Gelatin Strike. Treatment of the solution with gelatin to coagulate and remove silica impurities. Used by itself only for those cases where the fuel has been aged a long period of time.

- Combined Strike. Simultaneous formation of gelatin and MnO_2 precipitates for removal of silica and fission products used on concentrated dissolver solution with Zr and Nb concentration at normal level.

- Double Strike. Sequential treatment of dilute dissolver solution (prior to evaporation in the head end evaporator) with MnO_2 to precipitate higher than normal concentration of Zr and Nb followed by the formation of gelatin in the concentrated solution to remove silica.

- Dilute Combined Strike. Simultaneous formation of gelatin and MnO_2 precipitates in dilute dissolver solution for removal of high concentration of Zr, Nb, and silica.

3.2.2.5.1 Concentration

Dissolver solutions from the electrolytic dissolver containing a high concentration of iron, chromium, and nickel are processed without concentration since the hexavalent chromium present would cause severe corrosion of stainless steel evaporators. Aluminum solutions are normally evaporated to a concentration of 1.8M in the head end evaporator, which is subsequently diluted by jet transfers and strike solutions before solvent extraction to about 1.5M Al.

The concentration of fissile materials to unsafe levels is limited by temperature controls and by the presence of aluminum nitrate which raises the boiling point as the solution is concentrated. The mass limit of fissile material in the head end evaporator and its feed tank are based upon a precipitate covering but not exceeding the safe floor loading value so that even though the evaporator was concentrated to dryness, a criticality could not occur.

The evaporator instrumentation limits the evaporator temperature to prevent a possible explosive reaction between solutions possibly contaminated with solvent recycled from solvent extraction (see Section 3.2.3). Solvent can undergo vigorous exothermic reaction between nitric acid or heavy metal nitrates. The explosion or rapid reaction has involved the evolution of nitrogen oxides. Two incidents occurred only after concentration had proceeded to the point of incipient calcination of uranyl nitrate (>135°C).

The amount of entrained solvent in evaporator feeds is limited to 0.5% volume. In addition, the solution in the evaporator feed tank is agitated while the evaporator is being fed, and the evaporator feed tank is emptied periodically to prevent gradual accumulation of organic material.

3.2.2.5.2 Precipitation

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 the silica to form large molecules that agglomerate and precipitate from the solution.

Routinely, the raw metal solution is concentrated and a combined strike of gelatin to remove the silica and MnO_2 to remove both silica and fission products is used in preparation of the dissolver solution for solvent extraction.

Freshly precipitated manganese dioxide, by virtue of its large surface area, sorbs a major portion of the zirconium and niobium radioactivity that is present, and agglomerates or occludes suspended matter. The precipitate can then be flocculated by digestion so that 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°C to 75°C. Next, enough gelatin is added as a 1% solution to give 50mg gelatin per liter and then a 3% potassium permanganate solution is added at a rate of 10 to 15 pounds 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 1.5 to 2.5M.

In order to obtain zirconium and niobium decontamination factors up to 16 to 20, a dilute MnO_2 strike on pre-evaporated dissolver solution and a post-evaporation concentrated gelatin strike is used. Manganous nitrate and potassium permanganate are added to the unevaporated feed as described previously. The MnO_2 is removed by centrifugation; and the clarified liquor is evaporated and gelatin added to effect a second precipitation and a re-centrifugation.

Safety considerations, in the operation of the precipitation or strike processes, include the potential for unsafe concentration of fissile material by evaporation, the volatilization of ruthenium and an overflow of foam during the dissolution of the centrifuge cake.

Unsafe concentration during digestion is possible but less likely than during evaporation because the temperature is below boiling. Instrumentation with interlocks and alarms prevent excessive concentration from occurring (see Section 3.2.3).

Ruthenium is highly radioactive and in the volatile tetroxide form will pass through the filters. To prevent this from occurring manganous nitrate, a reductant, is added to the dissolver solution followed by the addition of potassium permanganate. Adding the reagents in this order avoids oxidizing the ruthenium to the volatile tetroxide.

The principal zones of evolution of ruthenium are the head end strike tank, the centrifuge, and the cake disposal tank. The head end strike tank would evolve relatively large amounts of ruthenium tetroxide if manganous nitrate was absent, if the rate of addition of potassium permanganate was excessive, or if the temperature was higher than about 80°C. To minimize such evolution, the addition of potassium permanganate is made at a controlled rate with maximum agitation, and the temperature of the precipitation and simmer is kept at about 70°C. Off-gases from the strike tank are routed through a ruthenium absorber that typically removes most of the ruthenium before the gas enters the process vessel vent system. In the main exhaust system, additional ruthenium may be removed by the vessel vent scrubber.

3.2.2.5.3 Centrifugation

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 3-8 shows the construction of the centrifuge and illustrates the various steps. The clarified solution overflows the centrifuge bowl and flows by gravity to the centrifuge run tank. After all the feed solution has been fed to the centrifuge, the product is moved from the run tank 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 run tank. Before each wash, the cake is dislodged by high-pressure sprays that inject about 20 gal of 0.75% nitric acid into the bowl compartments at 600 psig while the bowl is rotating at 10 rev/min. When the spray cycle is complete, the slurry is recentrifuged (1740 rev/min), and the wash solution is skimmed (870 rev/min) to the centrifuge run tank.

The washing procedure successively dilutes the uranium solution left with the cake, and decreases the uranium loss from about 2% to less than 0.1% of the total amount of uranium fed to the centrifuge. The skim solution and the cake washes (in the centrifuge run tank) are normally recycled to the strike tank to prevent any entrained cake from reaching the solvent extraction process.

After the final wash, the cake is broken up again 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 two additional washes, which are transferred to the high-activity waste disposal tank for neutralization.

Periodically, the centrifuge cake is dissolved with sodium nitrite and nitric acid and sampled for uranium content for accountability purposes. Normally the cake is slurried directly to Building 241 with the high level waste.

Nitrous acid (formed by the addition of sodium nitrite to dilute nitric acid) decomposes with evolution of nitrogen oxides. Too rapid an addition of sodium

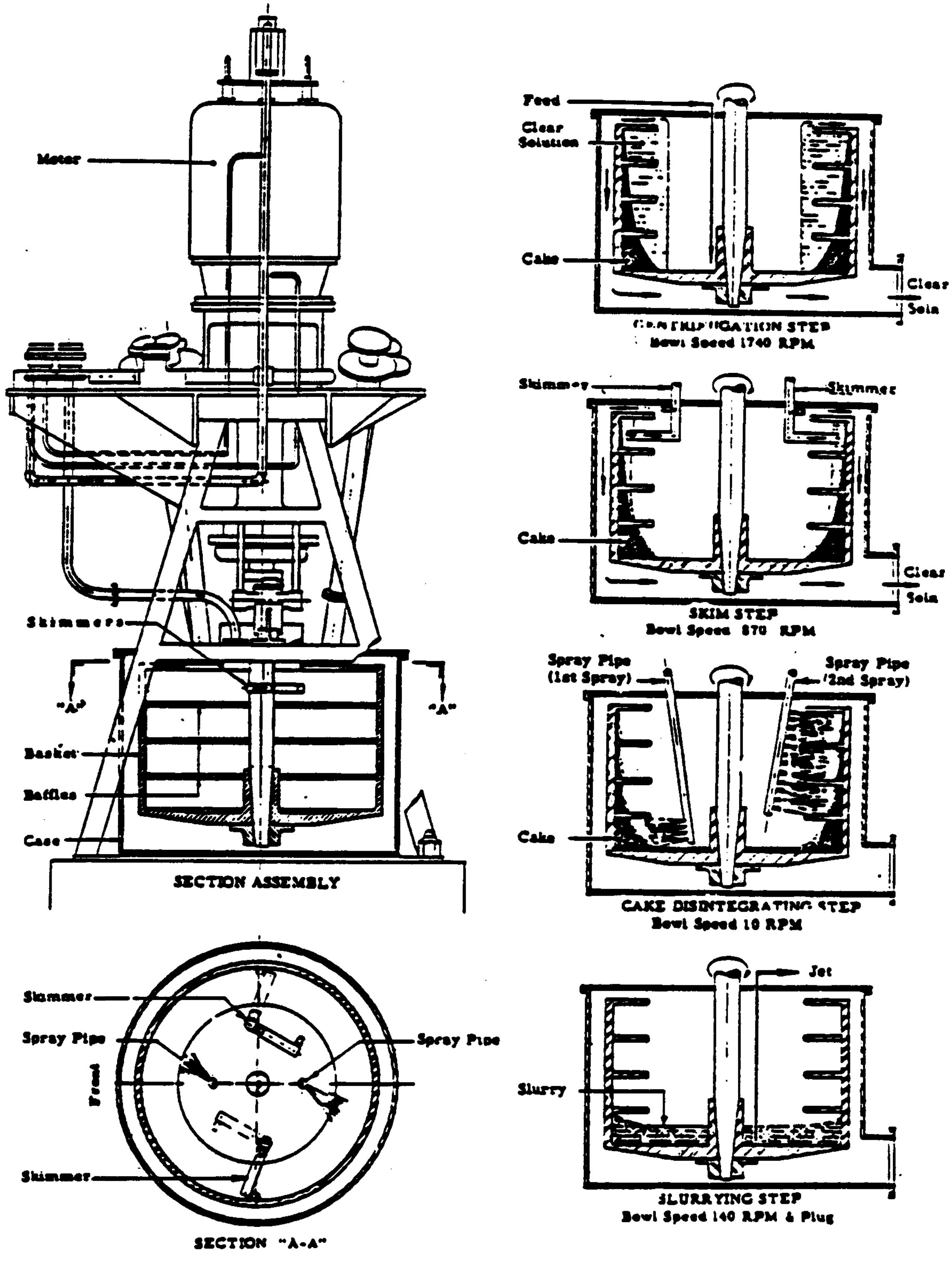


FIGURE 3-8. Head End Centrifuge and Centrifugation Steps

nitrite results in excessive gas evolution and foam-over 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 nitrite at a low rate.

3.2.2.6 Solvent Extraction

The HM Process is used for the recovery of enriched uranium and neptunium or plutonium from irradiated uranium-aluminum alloys and the Thorex Process is used for the separation of thorium and ^{233}U from protactinium and other fission products. The Thorex Process was run only periodically from 1964 to 1970. The primary mission of the plant is the recovery of enriched uranium and neptunium with the HM Process and the recovery of ^{238}Pu and Np with the Frame Process. The highly enriched uranium fuel processed in the HM Process contains insignificant quantities of ^{239}Pu that is routinely discarded. Significant amounts of Np are present in high enriched uranium and are recovered. Low enriched uranium contains ^{239}Pu which is recovered and stored for later purification.

H-Canyon is equipped with seven banks of mixer-settlers for the solvent extraction recovery and separation of the actinides. These seven banks are divided into first cycle processes consisting of three banks: one bank where most of the fission products are removed from the actinides of interest; a second bank where either ^{239}Pu or Np is partitioned into an aqueous stream; and the third bank where the enriched uranium in the organic is stripped into the aqueous.

There are two second cycles consisting of two banks each: one set of banks in the second uranium cycle, for the purification of enriched uranium; and a second set of banks in the second product cycle, for the purification of neptunium, plutonium, or thorium.

The extractability of the U, Np, and Pu are significantly affected by the concentration of $Al(NO_3)_3$ and HNO_3 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. Pu is extractable in the IV valence state and Np is extractable 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 of Pu with reductants is used to effect separation. Neptunium is maintained at valence IV and separated from uranium by utilizing the difference in distribution coefficients of U and Np.

A detailed description of the HM Process is found in the H-Canyon Systems Analysis (2). Figure 3-9 is a schematic of the HM Process and the Thorex Process and Figure 3-10 is a schematic of the canyon equipment. The composition and relative process flows are listed in Table 3-1 for the HM Process.

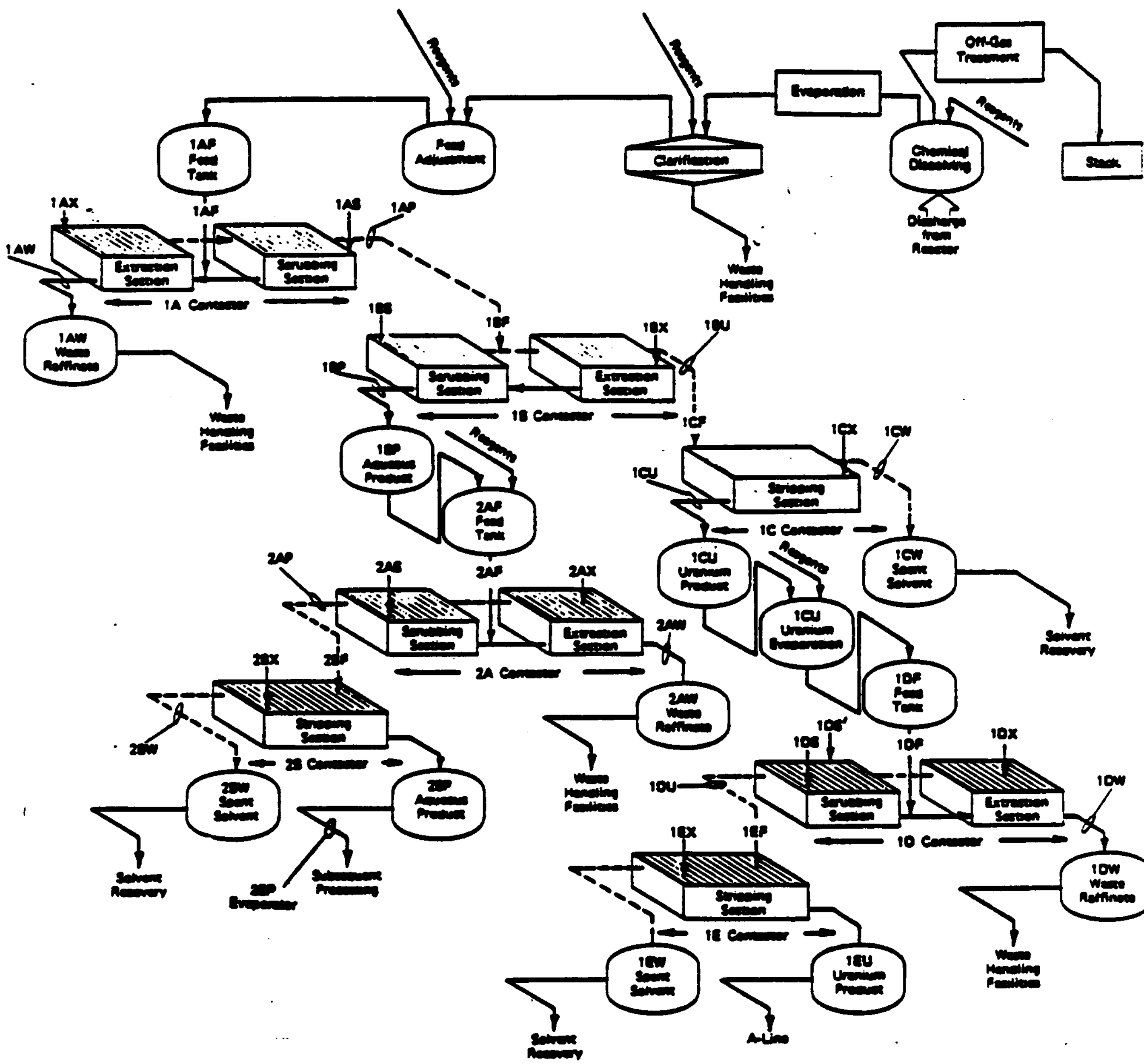


FIGURE 3-9. HM Process Solvent Extraction Flowsheet

WARM CANYON	SECTION NO.	HOT CANYON
Process Vessel Vent Fans Process Vessel Vent Dehumidifier and Heater Process Vessel Vent Filter Storage Module	5	Frame II-H Resin Digestion Tank Frame I-H Frame Dissolver and Frame Waste Hold Tank
Storage Module LAW Evaporator	6	Pot Dissolver 6.1D O.G. Filter and Iodine Reactor Electrolytic Dissolver Pot Dissolver
LAW Evaporator Overheads Condensate Tank LAW Evaporator LAW Evaporator Storage	7	6.4D O.G. Filter & I-Reactor Process Vessel Vent Filter Process Vessel Vent Scrubber Process Vessel Vent Scrubber Feed Tank
Pu-239 Storage Tank LAW Condensate Neutralization Tank LAW Evaporator Feed Tank B-Line Column Waste Receipt Tank	8	HAW Evaporator Feed Tank B-Line Solution Receipt Tank Dissolver Solution Accountability Tank HAW Concentrate Neutralization Tank
Low-Assay Pu-238 Storage Tank B-Line Column Waste Receipt Tank LAW Decanter 2BP Tank	9	HAW Evaporator 1st Stage HAW Evaporator 2nd Stage HAW Evaporator Overheads Condensate Tank 1AW Hold Tank
2BW Tank 2B Bank Mixer-Settler, 2nd Np Cycle 2BP Decanter	10	Head End Strike Tank Vent Filter Head End Strike Tank Head End Centrifuge Centrifuge Run Tank
2A Bank Mixer-Settler, 2nd Np Cycle 2AW Tank LAW Hold Tank	11	B-Line Scrap Recovery Solution Receipt Tank Head End Evaporator Feed Tank Head End Evaporator 11.3E Overheads Condensate Tank
2AF Tank First Cycle Solvent Washer, 1st Stage 2AF Adjustment Tank First Cycle Solvent Washer, 2nd Stage	12	Pu-239 Storage Tank 1AF Adjustment Tank 1AF Tank 1AW Decanter
1st Cycle Solvent Wash Hold Tank Two Decanters Decanted Aqueous Wash-Waste Tank 2nd Cycle Solvent Wash Hold Tank	13	1A Bank Mixer-Settler 1BP Tank 1B Bank Mixer-Settler
1EU Tank 2nd Np Cycle Solvent Washer 1BP Tank 2nd U Cycle Solvent Washer	14	1CU Decanter 1C Bank Mixer-Settler
1E Bank Mixer-Settler, 2nd U Cycle 1EU Decanter 1D Bank Mixer-Settler, 2nd U Cycle	15	1CW Run Tank Rerun Feed Adjustment Tank Rerun Batch Extraction Tank 17.2E Feed Tank
Warm Canyon Sump Solution Collection Tank 1DW Tank 1DF Tank	16	Resin Digestion/17.2E Neutralization Tank Decanter FWR Product Tank FWR Raffinate Hold/Decanter Run Tank FWR Resin Column FWR Raffinate Tank/FWR Feed Tank (Bicell)
1DF Adjustment Tank 1 CU/1DW Evaporator 17.6E & 17.8E Overheads Condensate Tank 2BP Evaporator	17	Sump Solution Hold Tank Rerun/FWR Batch Evaporator 17.2E Overhead Condensate Tank Storage Module
17.8E Feed Tank Storage 17.6E Feed Tank Storage	18	Rerun Product Wash Tank Hot Canyon Sump Solution Collection Tank 6.3D Hold Tank/1CU Tank (Divided Bicell)

T = Tank
D = Dissolver
E = Evaporator
H = Humidifier/Dehumidifier
F = Filter
R = Iodine Reactor
= Centrifuge

FIGURE 3-10. Approximate Location of H-Canyon Equipment

TABLE 3-1. HM-Process - Composition and Relative Process Flow

Stream	Relative Flow	Composition
1AF	100	1.5M $Al(NO_3)_3$, 1.5M HNO_3, U, Np
1AX	160	7.5% TBP in n-paraffin hydrocarbon
1AS	21	4M HNO_3, 0.12M $Fe(NH_2SO_3)_2$
1AW	116	1.2M $Al(NO_3)_3$, 1.6M HNO_3, 0.007M $Fe(NH_2SO_3)_2$
1AU	165	0.2M HNO_3, U, Np
1BF	165	0.2M HNO_3, U, Np
1BX	96	1.5M HNO_3, 0.08M $Fe(NH_2SO_3)_2$
1BS	288	7.5% TBP in n-paraffin hydrocarbon
1BP	97	1.4 HNO_3, 0.08M $Fe(NH_2SO_3)_2$, Np
1BU	452	0.1M HNO_3, U
1CF	452	0.1M HNO_3, U
1CX	114	0.01M HNO_3
1CW	448	7.5% TBP in n-paraffin hydrocarbon
1CU	118	0.4M HNO_3, U
1DF	90	4.1M HNO_3, U
1DX	193	7.5% TBP in n-paraffin hydrocarbon
1DS	29	0.9M HNO_3
1DS'	1	40% FS
1DW	127	3.4M HNO_3, 0.006M $Fe(NH_2SO_3)_2$
1DU	195	0.03M HNO_3, U
1EF	195	0.03M HNO_3, U
1EW	193	7.5% TBP in n-paraffin hydrocarbon
1EX	59	0.01M HNO_3
1EU	59	0.24M HNO_3, U
2AF	135	4.8M HNO_3
2AS	47	0.9M HNO_3, 0.02M $Fe(NH_2SO_3)_2$
2AX	135	30% TBP in n-paraffin hydrocarbon
2AW	198	3.7M HNO_3, 0.005M $Fe(NH_2SO_3)_2$
2AU	137	0.2M HNO_3, Np
2BF	137	0.2M HNO_3, Np
2BW	135	30% TBP in n-paraffin hydrocarbon
2BX	70	0.01M HNO_3
2BP	71	0.36M HNO_3, Np

3.2.2.6.1 First Cycle Solvent Extraction: Primary Fission Product Separation

The dissolver solution, after clarification, is transferred to the first cycle feed adjustment tank where the solution is adjusted to the proper nitrate concentration. A valence adjustment can be made to enhance or inhibit extractability of radionuclides by adding suitable reagents. This adjusted, clarified feed (1AF), is pumped to the mixing section of the center stage (stage 8) of the 1A mixer-settler bank. This is a 16 stage counter-current liquid-liquid contactor. Each stage consists of a mixing section and a settling section.

The aqueous and organic phases flow concurrently from the mixing section through the settling section where the organic and aqueous phases are disengaged by gravity, and flow counter-currently between adjacent stages (Figure 3-11).

In the HM process an extractant of 7.5 volume percent tri-n-butyl phosphate (TBP) diluted with a hydrocarbon diluent is pumped to stage 16; the aqueous scrub solution of 4M nitric acid enters at stage 1. In the extraction section of the 1A bank (stages 8 through 16), the actinides U, and Np are selectively extracted into the organic phase while the bulk of the fission products and aluminum remain in the waste and exit the 1A bank as waste, 1AW, from stage 16. In the scrub section (stages 1 through 7), the organic bearing the actinides is washed by an acidic solution (1AS) to remove additional fission products entrained or extracted into the organic. The streams coming into the 1A bank (1AF, 1AS, and 1AX) are heated to improve decontamination from fission products, and to improve hydraulic performance, allowing the bank of mixer-settlers to run efficiently at the highest possible flow rate.

The composition of the process streams changes for the Thorex Process, but the function of primary decontamination of dissolver solution remains unchanged. The volume percent TBP in the extractant, 1AX, is increased to 30 volume percent and sodium phosphate is included in acid strip 1AS.

3.2.2.6.2 Partitioning

The scrubbed organic phase of enriched uranium, 1AU, flows by gravity to the center, stage 8, of the 16 stage 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 nitric acid extractant (1BX) solution contains a reductant such as ferrous sulfamate. The small quantity of plutonium is back-extracted, into the aqueous phase, where it is reduced to the near-inextractable valence III and exits the 1B mixer settler from stage 16. The neptunium remains in the extractable IV valence but the relatively low concentration of nitric acid in the 1BX results in the transfer of the neptunium from the organic phase to the aqueous phase. The acid is high enough to keep the U in the organic. This aqueous product solution is scrubbed with fresh organic in the scrub section (stages 8 through 16) to remove residual uranium.

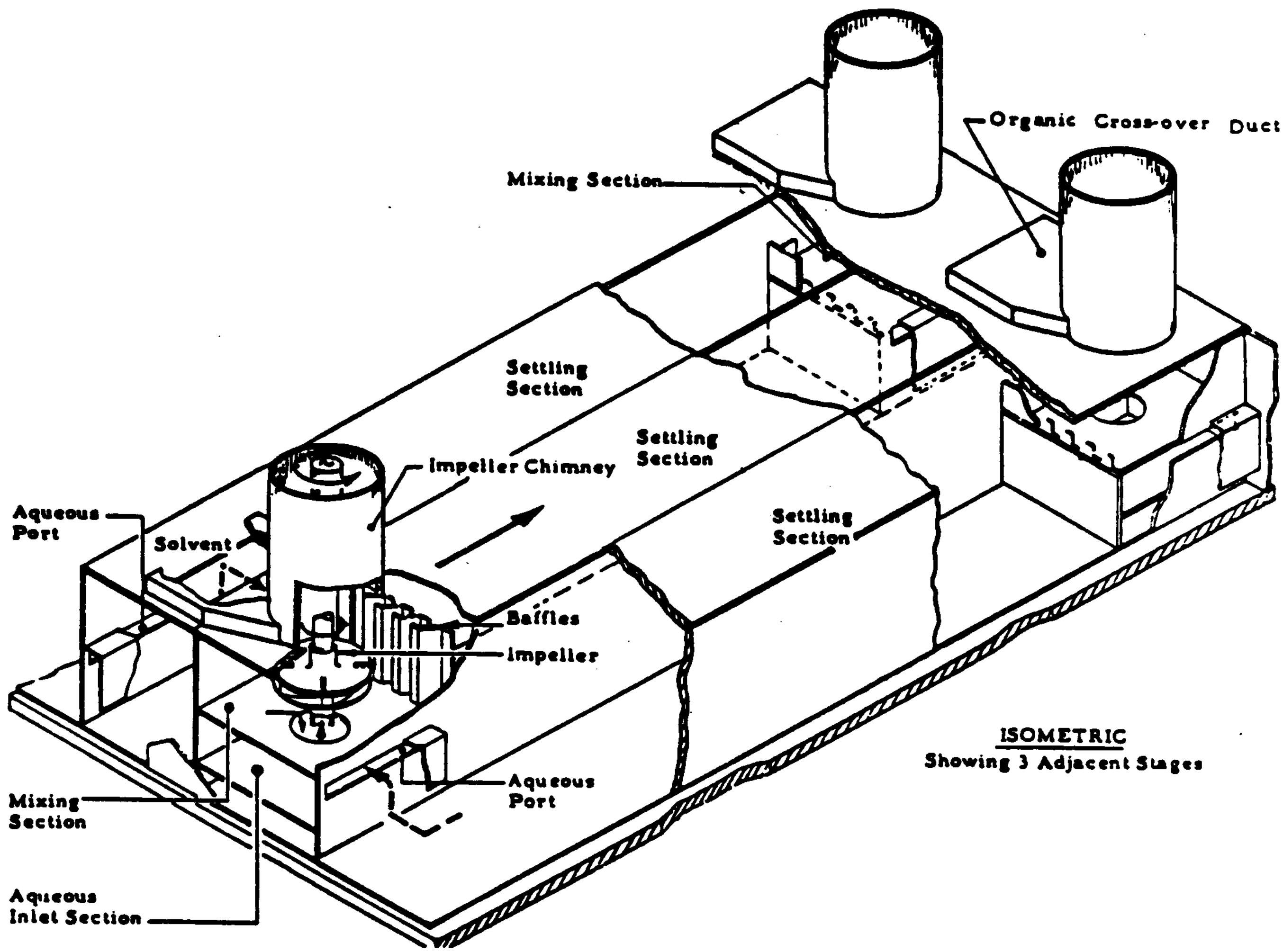


FIGURE 3-11. Isometric View of Mixer-Settlers

The scrubbed aqueous, of primary importance for the Np it contains, is sent to the second product cycle for further fission product decontamination. In the Thorex Process, the thorium is partitioned from the uranium in the 1B bank with a dilute nitric acid solution. The thorium, like the Np partitioned in the HM Process, is sent to the Second Product Cycle for further decontamination while the uranium is diverted to the Second Uranium Cycle.

The organic from the 1B bank, containing uranium, a small amount of 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 to remove degraded organic and is then recycled to the process. The uranium in the 1CU is evaporated to increase the uranium and nitric acid concentrations. Increasing the uranium by evaporation is also routine in the Thorex Process.

The evaporation of an aqueous stream containing gross amounts of organic from an abnormal mixer-settler bank could create a potential explosion hazard. Tank-type decanters (Figure 3-12) are used to remove the solvent and the solvent is periodically transferred to rerun or returned, by means of an air lift, to the banks.

3.2.2.6.3 Second Uranium Cycle

The uranium-bearing aqueous solution (1CU) from the 1C contactor is concentrated in the 1CU evaporator where acid is added and the solution concentrated to ~4.5M to enhance the extractability of the uranium in the second cycle of solvent extraction. If more acid is needed, it is added to the 1DF adjustment tank.

The adjusted feed (1DF) is next transferred to the 1DF feed tank from which it is delivered continuously at a controlled rate to the mixing chamber of the center stage of the 16 stage 1D contactor. Organic extractant (1DX) is pumped to the end stage (stage 16). The uranium is extracted into the organic phase; the fission products, any plutonium reduced by the FS in the scrub to Pu (III) and 60 to 95% of any neptunium, remain in the aqueous phase and leave the contactor at stage 16 in the aqueous waste stream (1DW).

The organic solution of uranium, while flowing toward stage 1, is scrubbed with a solution of dilute nitric acid containing ferrous sulfamate to strip any contaminants out of the organic. The nitric acid scrub (1DS) enters stage 1 while the 1DS prime stream containing 40% ferrous sulfamate enters the contactor in the fourth stage. The ferrous sulfamate is added to stage 4 to avoid iron contamination in the small amount of entrained aqueous in the uranium effluent (1DU).

The acidity of the scrub (1DS) and the flow ratios of the extract-to-scrub streams are adjusted to reject the Np (IV) to the aqueous waste (1DW) while retaining extracted uranium in the organic.

The organic product stream (1DU) leaves from the contactor at stage 1 and flows by gravity to stage 12 of the 1E bank. In the 1E contactor, the uranium

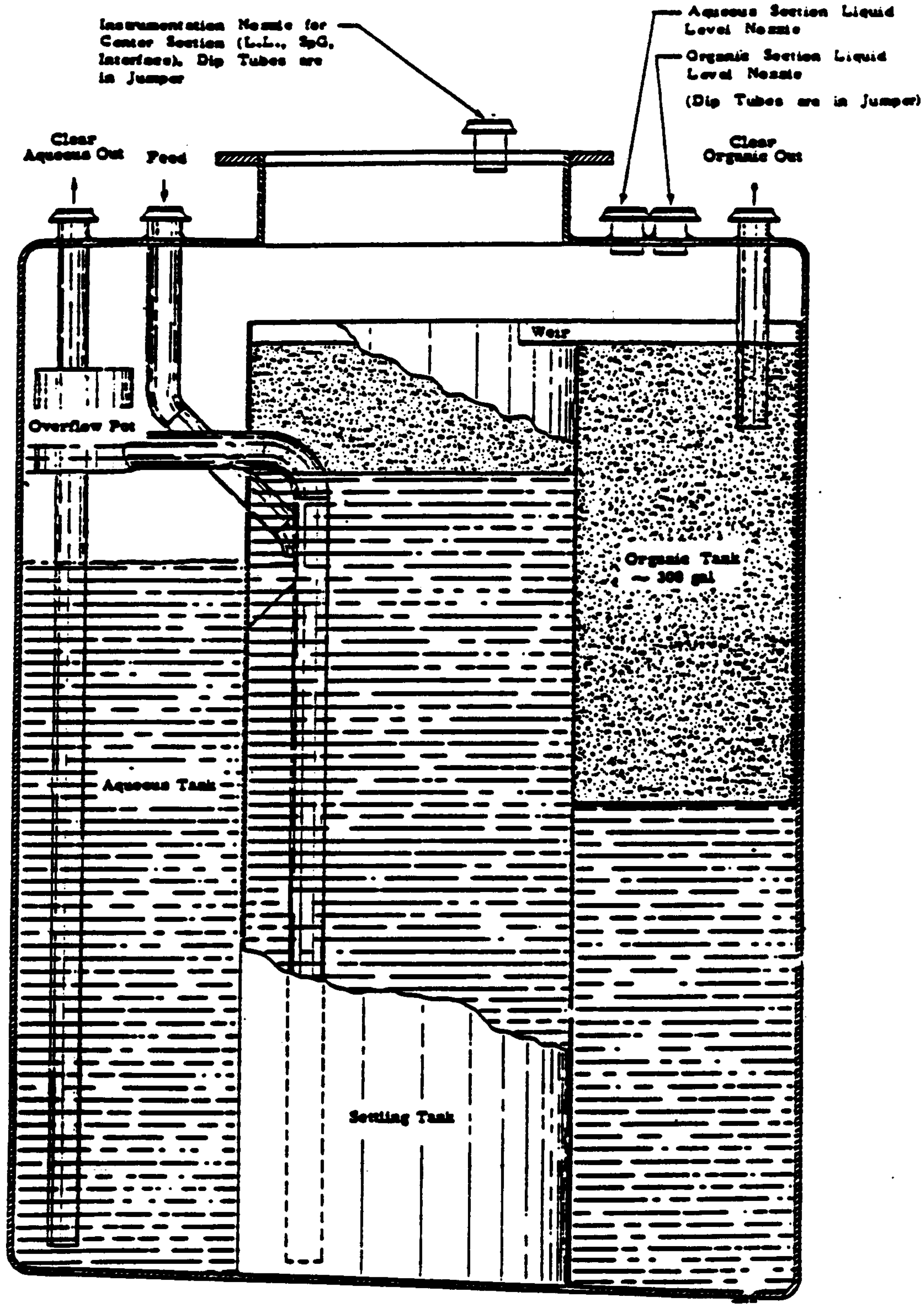


FIGURE 3-12. Tank Decanter

is stripped from the organic phase into a counter-current stream of water slightly acidified with nitric aid (1EX). The 1EU is shipped offsite. The spent organic stream (1EW) is discharged from stage 1, and pumped to hold tanks in solvent recovery facilities where it is washed and stored for reuse. If the uranium product 1EU is low-enriched in ^{235}U, it may be evaporated before shipment.

The Second Uranium Cycle process used for the recovery of enriched uranium (HM Process) is very similar to the Second Uranium Cycle process used for the recovery of ^{233}U. The concentration of TBP in the solvent is 7.5 volume percent, and the acidity of the feed is adjusted initially. Contamination of the uranium with plutonium, a concern in the recovery of enriched uranium that necessitated the addition of ferrous sulfamate in the 1DS, is not a problem in the Thorex Process. The ^{233}U product from solvent extraction (1EU) is further purified in the HB-Line.

3.2.2.6.4 Second Product Cycle: Neptunium Purification

The second product cycle provides additional decontamination for the product being recovered; Np, Pu, or Th, with the purification of Np from the first cycle being the primary effort. Plutonium accumulated primarily from the processing of low enriched uranium in the HM Process is processed infrequently and Th from the Thorex Process is processed even less frequently.

The neptunium partitioned from the uranium in the 1B bank is combined with the raffinate from the anion exchange column in the HB line and, depending on the flowsheet, waste from the 1D bank which has been evaporated in 1CU/1DW evaporator to increase the HNO_3 acid concentration to 7.6 M HNO_3. The Np in the feed solution is acid adjusted and fed to the center stage of the 16 stage 2A mixer settler bank. The Np is extracted in the solvent phase which contains the extractant, TBP, at a concentration of 30 volume percent and then leaves the mixer-settler at stage 1. Fission products and other impurities are scrubbed from the Np bearing organic into the aqueous phase with 0.9 M HNO_3 and exits the contactor as waste (2AW) from stage 16.

The organic with neptunium (2AP) flows by gravity to the 16th stage of the 2B contactor, where neptunium is stripped from the organic into a slightly acidified aqueous stream (2BX) introduces at stage 1. The stripped solvent (2BW) is discharged to solvent recovery. The aqueous neptunium solution (2BP) is fed to the neptunium evaporator for concentration and temporary storage.

After the thorium campaigns in the late 1960s, the need for a flowsheet which could decontaminate Np from Th necessitated a thorium removal flowsheet. The oxidizing agent, ceric ammonium nitrate, was added to the 2BP evaporator concentrate to adjust the Np to most extractable VI valence and recycled to the 2nd cycle. By adjusting the relative flow ratios of feed (2AF) to extractant and the nitric acid concentration, the Np was extracted leaving the Th in the aqueous phase where it accumulates in the 2A bank and eventually leaves in the aqueous waste, 2AW. This flowsheet has not been used in many years.

3.2.2.6.5 Second Product Cycle: Plutonium Purification

The second product cycle is also used to further decontaminate the plutonium-bearing solution that has accumulated in previous campaigns or that has been partitioned in the 1B bank in a current operation while processing low enriched uranium.

The plutonium solution is collected in an accumulation tank, chemically adjusted by the addition of sodium nitrite to oxidize the plutonium to the extractable IV valence, and diluted, if necessary, to Pu concentration that is acceptable for processing in the second Pu Cycle. The acidity of each batch of feed is adjusted for optimum extractability of the Pu.

After adjustment to the proper concentration of nitric acid, the feed solution (2AF) is jetted at a controlled rate to the center stage of the 2A contactor. The plutonium and any uranium are extracted into the organic phase (2AX); the fission products, iron, and other inorganic contaminants remain in the aqueous phase, and leave the process in the aqueous waste stream.

Control of the 2A bank operation is directed to 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, since nitrite is readily extractabale into the organic.

The aqueous waste is combined with other wastes and transferred to the low activity waste evaporator.

The organic solution of plutonium is scrubbed with dilute nitric acid (2AS) for removal of fission products, and the emergent organic stream (2AP) flows by gravity to the 2B contactor. The 2B mixer-settler bank utilizes a stripping (2BX) solution consisting of hydroxylamine nitrate in dilute nitric acid to reduce the Pu to relatively inextractable III valence. The product stream (2BP) is to be transferred to the new HB-Line for conversion to the oxide. The stripped solvent (2BW) is sent to solvent recovery.

3.2.2.6.6 Thorium Purification

The thorium product stream from the first cycle of the Thorex Process is concentrated by evaporation and adjusted for extraction by 30 volume percent TBP in a hydrocarbon diluent. In this cycle, thorium and any uranium contaminant are extracted while the residual protactinium and fission products are discharged to the aqueous waste (2AW). The extracted thorium is scrubbed with HNO_3 solution containing Na_3PO_4, and is then stripped into acidified water in the 2B contactor; the aqueous thorium solution (2BT) is evaporated and stored.

3.2.2.6.7 Solvent Extraction: Safety Considerations

Nuclear safety is based on concentration control to maintain the concentration of fissile material well below the critical concentration (8). The solvent extraction equipment is geometrically unfavorable for fissile material exceeding the minimum safe solution concentration; therefore, safe concentration must be maintained by: 1) rigid administrative control of the operation of the process, 2) frequent monitoring, sampling, and material balance checks, and 3) the use of colorimeters, conductivity probes, and continuous nuclear monitors (see Section 3.2.3, Instrumentation and Controls). Concentrations within the solvent extraction systems in the first cycle and the second uranium cycle, where an accumulation of enriched uranium is a potential, are limited by the concentration of uranium in the feed, aqueous and organic flow ratios, concentration of nitrate in the aqueous streams, and by the extractant concentration at 7.5 volume percent TBP.

An accumulation of fissile material in the 1A and the 1D banks can occur if there is insufficient nitric acid or aluminum nitrate in the scrub or when the flow rates of the various input streams are off-standard (such as low solvent flow and high feed and scrub flow). However, in a reflux or internal recycle situation, some product would be lost to the waste raffinate before unsafe concentrations are exceeded. In the 1C and 1E banks, the concentration of fissile material in the banks is directly proportional to the flow ratio of organic to aqueous streams.

When ^{239}Pu is processed in the Second Product Cycle, there is an initial accounting of the Pu in the feed batch. This batch of feed is processed through the 2A and 2B banks and the 2BP is collected and sampled for Pu and an accounting performed to be sure that the processing has not left Pu accumulation in a mixer-settler bank. A flush often follows the processing of a batch of feed to assure that an accumulation in the system has not occurred and takes the place of the accounting system. Analysis of periodic samples of the Second Product Cycle, particularly the 2BP product stream, are used to detect off-standard conditions.

3.2.2.6.8 Solvent Flammability

The HM Process uses organic solvent consisting of 7.5 volume percent or 30 volume percent TBP in normal parafin hydrocarbon (NPH).

The TBP is considerably less volatile (flashpoint 146°C) than NPH (flashpoint 70°C). The properties of the NPH diluent are, therefore, considered to be controlling with respect to solvent flammability.

The NPH is a mixture of 5 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 C_{10} to C_{14} constituents and a flashpoint of $70° \pm 1°C$. 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 greater 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 for 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 resulting from the sampling system. The highest concentration of aerosol obtained in these experiments approached 2.0 mg NPH/liter of vapor at 60°C (10). The equilibrium vapor phase concentration of NPH was calcualted 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 C_8 and greater. Based on these studies, a flammable concentration safety margin exists (10).

3.2.2.7 Anion Exchange: H Frames Process

The NpO_2-Al targets are processed through anion exchange for the recovery of ^{237}Np and ^{238}Pu. The targets are dissolved in the frames dissolver, 5.4, in a solution of nitric acid, mercuric nitrate, and fluoride. There are three anion exchange columns used in the process: column RC-1 separates the ^{237}Np and ^{238}Pu from aluminum and fisson products; column RC-2 partitions the ^{238}Pu from the ^{237}Np and further decontaminates Np; column RC-4 decontaminates the ^{238}Pu from iron and from residual fission products (Figure 3-13). Losses in the waste from the frame operation are recovered by anion exchange column RC-16 (Figure 3-14).

3.2.2.7.1 First Column, RC-1

In the absorption of Np and Pu on RC-1, the nitrate ions are displaced from the resin by the anionic nitrate complexes of Np (IV) and Pu (IV). The resin effectively absorbs the Np and Pu only if the feed solution has sufficient concentration (6 to 9M) of nitrate ion and if the Np and Pu are in the IV valence.

Dissolver solution is moved to the first anion exchange column feed tank and adjusted to 8.1 molar total nitrate. The valence state of the dissolver solution is not uniformly IV for the Pu and Np; the Pu valence is split between the IV and VI valence states, and the Np is in the VI valence state. The neptunium is reduced to (IV), and the plutonium to (III) with ferrous sulfamate. Oxidation of Fe(II) to Fe(III), and Pu(III) to Pu(IV) is made by heating to 55°C for a controlled period of time. The adjusted feed contains the anionic nitrate complexes of Np(IV) and Pu(IV) in a solution of optimum nitrate concentration for absorption.

The adjusted feed solution downflows through a non-agitated bed (RC-1) of anion exchange resin, and the complex anions of ^{238}Pu and ^{237}Np are selectively absorbed. The rate of flow of the feed and the total quantity of actinides absorbed are limited to avoid excessive loss of Np and Pu. After the bed of resin is washed with 8M HNO_3 and 0.005M fluoride for fission product removal, Np and Pu are eluted with a dilute solution of nitric acid.

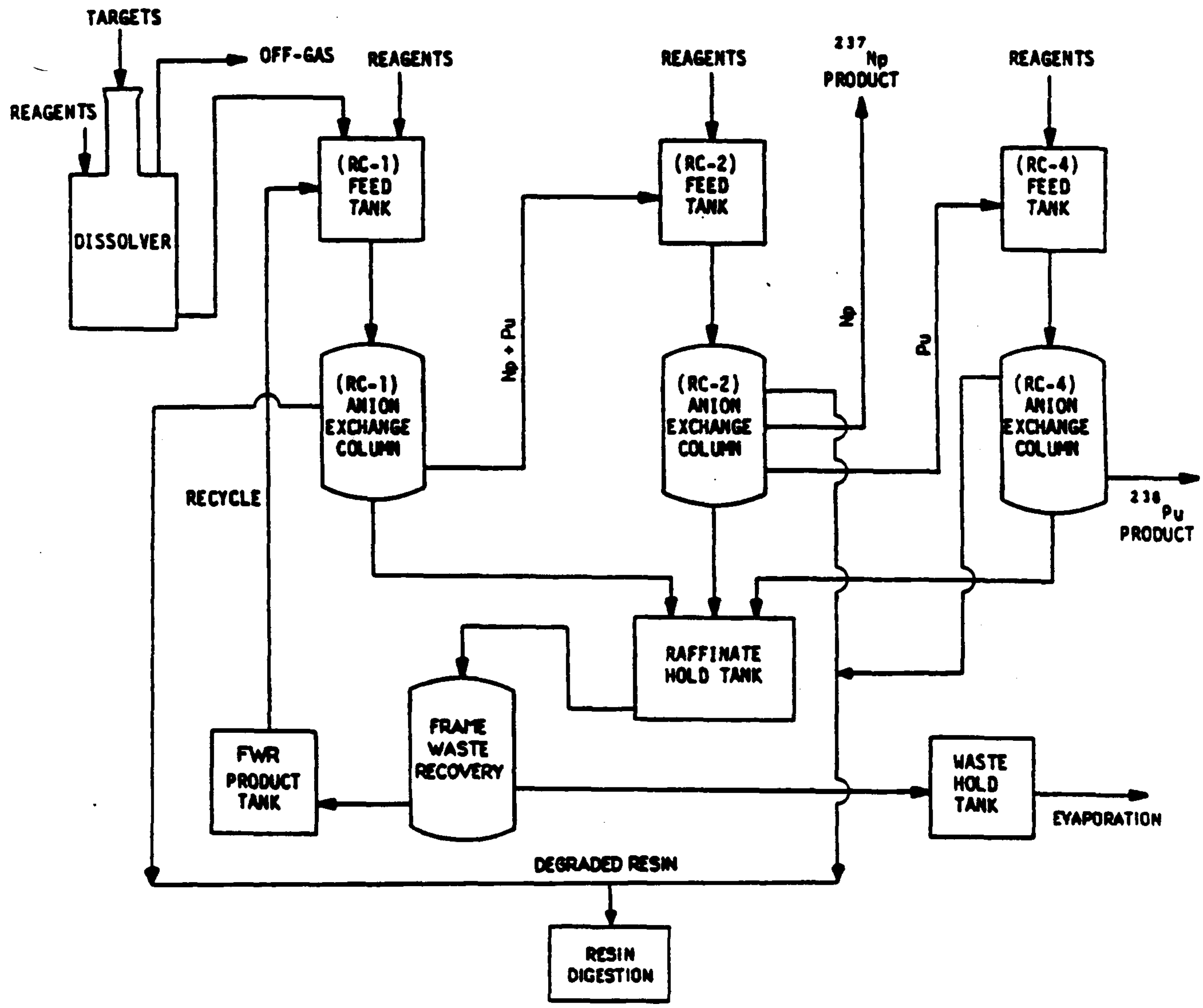


FIGURE 3-13. Target Processing in Building 221-H

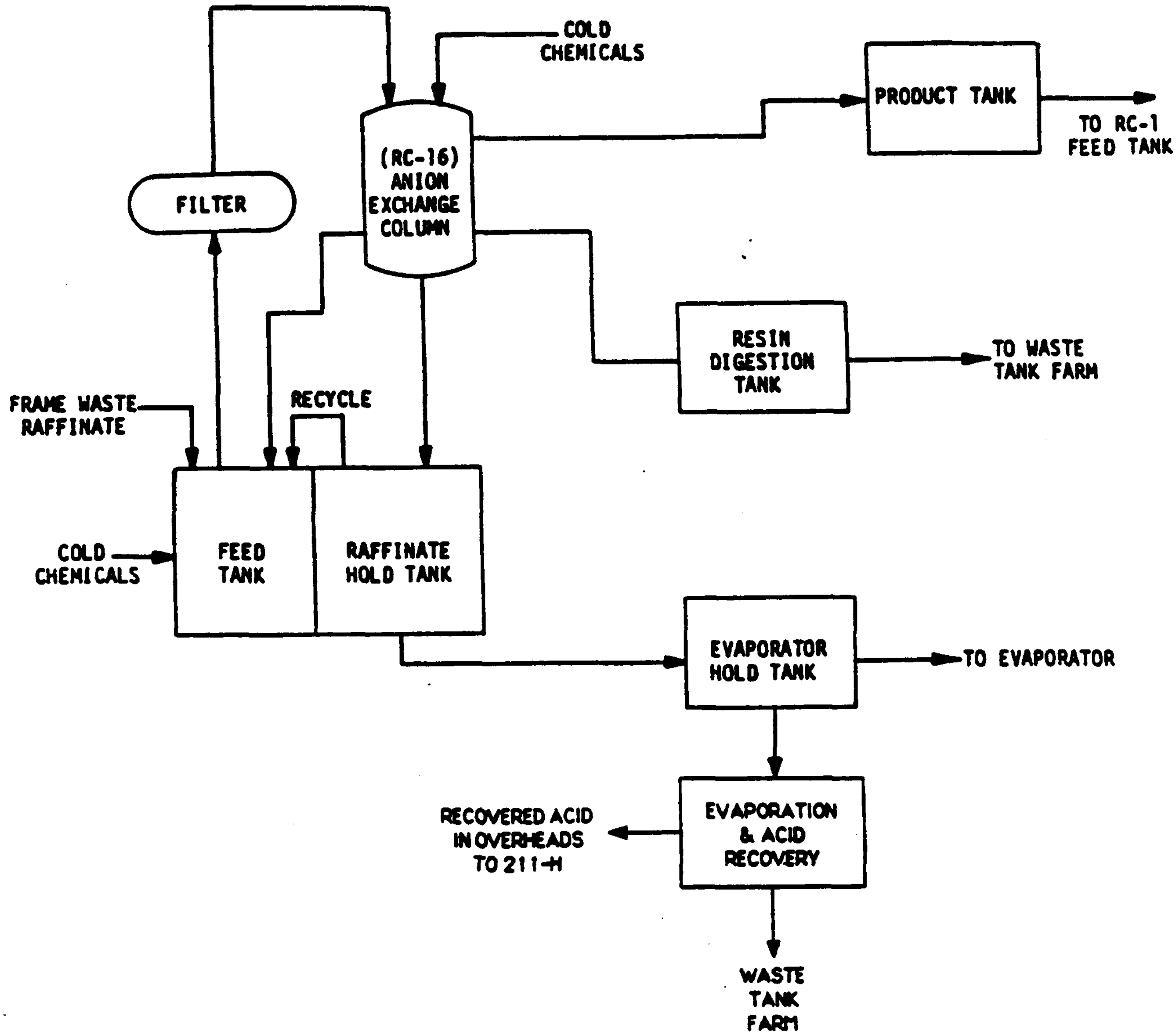


FIGURE 3-14. Frame Waste Recovery Process

3.2.2.7.2 Second Column, RC-2

The eluate from RC-1 contains Np(IV) and Np(V), and Pu(IV) and Pu(VI) in approximately 3M HNO_3. This solution is adjusted to 8M HNO_3, and the Np reduced to (IV) and the Pu to (III) with ferrous sulfamate. Oxidation of Fe(II) to Fe(III), and Pu(III) to Pu(IV) is done by heating to 55°C.

The adjusted feed solution, containing anionic nitrate complexes of Np(IV) and Pu(IV), flows downward through a fixed bed (RC-2) of anion exchange resin. After the absorption is complete, the plutonium is selectively eluted from the resin bed with a partitioning solution containing 5.7M HNO_3 -0.05M hydrazine-0.125M ferrous sulfamate. This solution reduces the Pu(IV) to Pu(III), causing the breakdown of the absorbed Pu(IV) complex and thus allows the removal of the less absorbable Pu(III). Nitrate concentration in the partition solution must be maintained above 5.4M to prevent the simultaneous elution of neptunium.

When the removal of plutonium is complete, the resin bed is washed with a solution of 8M HNO_3-0.05M fluoride to decontaminate the remaining neptunium. After the decontamination wash is complete, the Np is eluted with a dilute solution of nitric acid, and transferred to the HB-Line where it is converted to oxide.

3.2.2.7.3 Third Column, RC-4

The partition effluent from the second anion exchange column contains Pu(III), Fe(II) and hydrazine. This solution is adjusted to 8M HNO_3 and the Pu and Fe are oxidized to the IV and III valence states, respectively.

The adjusted feed solution containing the anionic nitrate complex of Pu flows downward through a fixed bed (RC-4) of anion exchange resin. The absorbed Pu is washed with a strong solution of nitric acid to remove fission products. The Pu is eluted with a dilute solution of nitric acid (0.35M). The eluate containing the Pu is transferred to the HB-Line, where the Pu is converted to oxide.

3.2.2.7.4 Np and Pu Recovery From Anion Raffinates

Waste streams from anion exchange columns RC-1, RC-2, and RC-4, and dissolved scrap and filtrates from HB-Line are processed through anion exchange column RC-16 to recover ^{237}Np and ^{238}Pu in this waste. (See Figure 3-14.) The acidity and valence of the Np and Pu are adjusted to the RC-1, RC-2 and RC-4 feed specifications by the addition of HNO_3 and ferrous sulfamate followed by heating. The Pu and Np in the adjusted feed is absorbed on the RC-16 column anion exchange bed.

The absorbed Pu and Np are eluted with dilute nitric acid. This eluate is blended back with normal feeds to the first anion exchange column (RC-1). The waste raffinate from RC-16 is evaporated in the re-run evaporator (17.2E) to recover acid, and the concentrate neutralized and transferred to Building 241-H for underground waste storage.

3.2.2.7.5 Safety Considerations

The processing of ^{238}Pu and Np concentrate with anionic resin 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 shows that the dried nitrate form of anion resin is explosive and could be detonated in the falling hammer test (3). The Operational Safety Requirement (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 a closed bomb test temperatures up to 200°C (3). 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 gas mixtures are excluded.

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). Diminished ion capacity necessitates an earlier resin replacement than resin replaced simply because of increased resin radiation damage. The intense alpha radiation from ^{238}Pu is particularly damaging, making rigorous accounting of the accumulated exposure from the actinides a most important control measure.

Design or control for the critical mass for ^{238}Pu and ^{237}Np are unnecessary because of the large masses required for these isotopes to become theoretically fissionable.

3.2.2.8 Evaporation

3.2.2.8.1 General

Batch evaporators are used in 221-H to:

1. Increase uranium or neptunium concentration available in feed streams for operational efficiency. The function of the head end product evaporator, the 1CU/1DW concentrator, and the 2BP concentrator, have been discussed in the section on head end and solvent extraction.

Evaporator Service	Evaporator No.
Head End Product	11.3E
Rerun (Sump Solutions)	17.2E
1CU/1DW Concentrator	17.6E
2BP Concentrator	17.8E

2. Recover acid (primarily from evaporation of Low Activity Waste) and economize by reducing the waste volume through the evaporation of excess water.

Evaporator Service	Evaporator No.
Low Activity Waste	6.8E, 7.6E, 7.7E
High Activity Waste (1st Stage)	9.1E
Frame Waste (1st Stage)	17.2E
HAW, Head End, and Rerun (2nd Stage)	9.2E

The batch evaporators also decontaminate recoverable acid overheads from another evaporator. The overheads from the High Activity Waste Evaporator, 9.1E, the Head End Product Evaporator, 11.3E, and occasionally, the Rerun Evaporator, 17.2E, are re-evaporated in the 2nd stage High Activity Waste Evaporator 9.2E.

The operation of the batch evaporator as described in the following section and the principal safety consideration of a vigorous exothermic reaction from entrained organic pertains to all evaporators.

3.2.2.8.2 Low Acidity Waste Evaporation

The acidic waste from the second product cycle (2AW) (and depending on flowsheet, 1DW) is mixed with the sodium carbonate wash solution used to purify the solvent of degradation products. Carbon dioxide produced from the controlled addition of sodium carbonate and the acidic 2AW is vented. If there is any entrained organic in the waste, it is removed in the LAW decanter (Figure 3-12) where solvent disengages from the aqueous phase and is periodically transferred to rerun.

The acidic mixed waste is moved to the LAW feed tank where it is combined with any available decontamination or tank 805 solutions. Process water is added to dilute the feed to about 7% nitric acid so that the concentrator does not exceed 8M HNO_3; this minimizes corrosion. The diluted acidic waste is transferred to any or all of the three LAW evaporators that are operated in parallel. The feed is transferred to the evaporator until the concentration reaches about 12% dissolved solids, then three volumes of stripping water per volume of concentrate are added and evaporated at a constant specific gravity to remove additional acid. About one-third of the residual nitric acid is boiled off as condensate.

The dilute feed is concentrated 40 to 80 times, yielding 15% solids and 27% nitric acid. At the completion of evaporation, the concentrate is cooled to 60°C and transferred to the concentrate neutralizer tank. Condensate from the LAW evaporators flows to the low activity condensate tank, transferred to basin tank B-1-2, and pumped from the basin transfer tank to the acid recovery unit.

3.2.2.8.3 High Activity Waste Evaporation

Aqueous wastes (1AW) from the first extraction cycle, vessel vent scrubber solutions, and occasionally decontamination and tank 805 solutions are combined in the HAW feed tank and evaporated in the first-stage HAW evaporator (9.1E). Condensate from 9.1E flows by gravity to the second-stage (9.2E) evaporator where it is again evaporated. Condensate from the double evaporation is re-evaporated in the acid recovery unit in Building 211-H. High activity waste evaporation is essentially a concentration step to reduce waste volume. Minimal acid is recovered in the overheads.

The rerun evaporator, 17.2E, is used to prepare feeds for batch extraction or to recover acid and concentrate raffinate from the Frame Waste Recovery System (FWR). The FWR concentrate is transferred to the 16.1 tank, neutralized, and sent to Building 241-H. The condensate from 17.2E is evaporated in the 9.2E evaporator if the activity is too high for transfer to the acid recovery unit.

3.2.2.8.4 Batch Evaporator Operation

All the evaporators in H-Canyon are batch evaporators with a pot, containing steam coils, and a condenser in a column that can be attached and detached by the overhead crane (Figure 3-15). Feed solution is transferred to the evaporator at a controlled rate, by jets controlled by 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. A small portion of condensate is diverted to the top tray, and overflows 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.

3.2.2.8.5 Safety Considerations - Evaporators

Vigorous exothermic reactions can occur between nitric acid or heavy metal nitrates and TBP under conditions that might arise during the concentration of the waste and product solutions. Violent reactions resulting in equipment damage were experienced during early development work (11,12).

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°C).

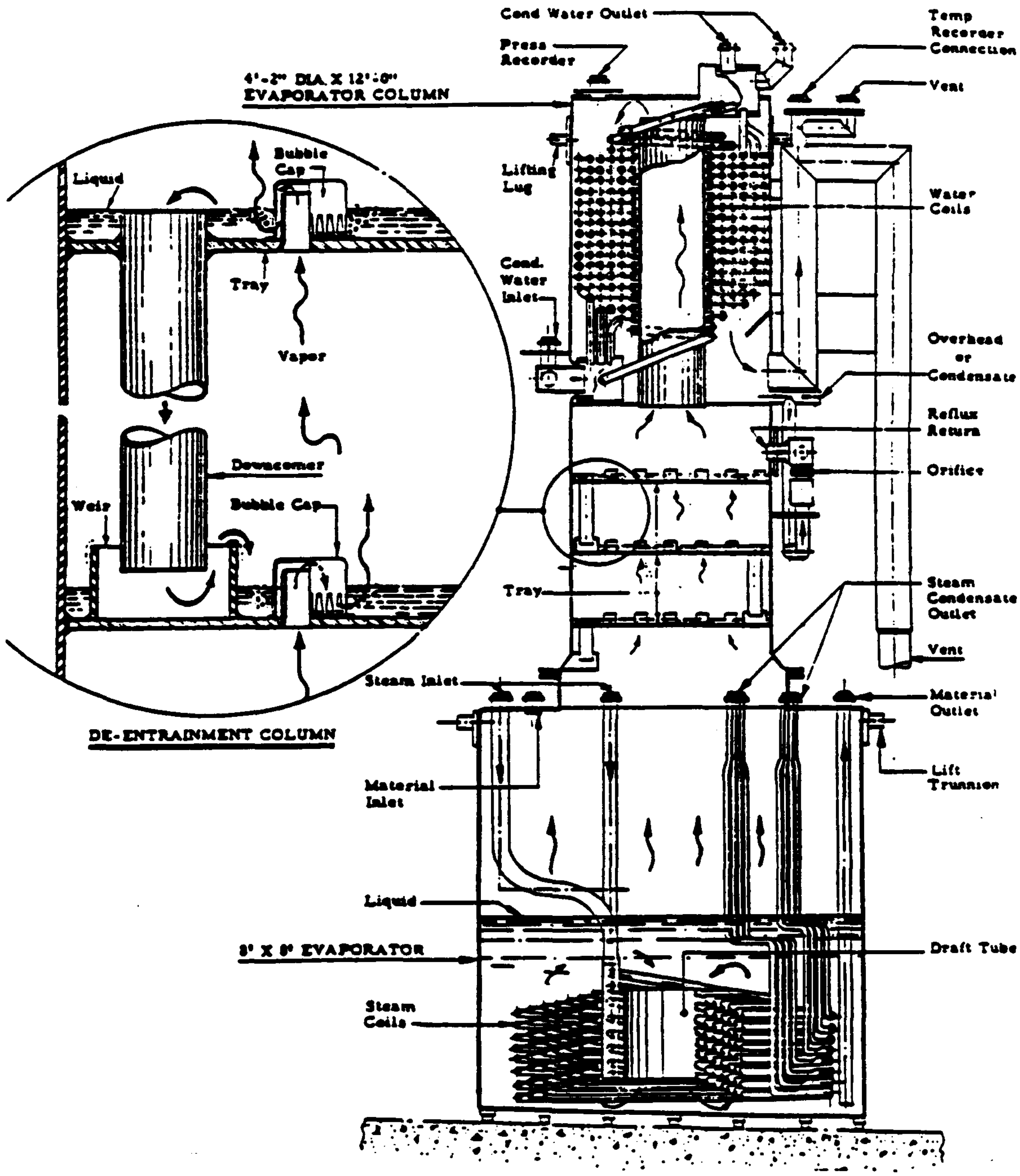


FIGURE 3-15. Standard Coil Batch Evaporator and Column

- In each incident, solvent was present in the concentrated product and a dense, second organic phase was present 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 H-Canyon batch evaporator operation is controlled to prevent the introduction of solvent and to prevent exceeding the safety limits. Entrained solvent in the evaporator feed is limited to 0.5 vol %. Under normal operating conditions the reboiler steam pressure is limited to 25 psig to limit the miximum temperature attainable. The evaporator temperature is generally limited to 118°C although the limit in the OSR is 130°C (8) (see Section 3.2.3 Instrumentation and Control). Evaporator feeds from solvent extraction are passed through a decanter prior to introduction into the evaporator.

Evaporators are geometrically unfavorable for concentrations of fissile material in excess of the OSRs (8). Uranium concentration less than the always safe solution concentration is maintained by control of the endpoint concentration (see Section 3.2.3.5). The evaporators are limited to a batch volume that contains insufficient fissile material to exceed critical mass if precipitated evenly across the bottom of the evaporator. This is known as the areal limit.

3.2.3 Instrumentation and Controls

Each process, dissolution, precipitation, extraction, ion exchange, and waste evaporation within the 221-H building, has control parameters unique to that process to maintain the process within a safe regime of operation with respect to personnel and equipment, and to efficiently produce a satisfactory product. The limits of operation are defined in the 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

After the nitric acid and water are added and the dissolver charged with fuel, the acid temperature is increased to boiling and $Hg(NO_3)_2$ catalysts are added producing a vigorous reaction. The reaction is controlled by regulating the steam flow and the $Hg(NO_3)_2$ addition rate. When the reaction subsides the steam to the coil is increased to produce vigorous boiling to provide good agitation. The vacuum in the dissolver is regulated to sweep off-gas from the dissolver by maintaining the dissolver vessel under a constant vacuum by a steam jet. Loss of vacuum is a concern because vessel pressurization could spread contamination outside the confines of the canyon by way of pneumatic instruments and service lines.

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. If a coil develops a leak, the air leaks out of the coil, rather than radioactive solution leaking into the coil where it would 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 H-Canyon by packed beds of silver-coated berl saddles. The reaction efficiency is affected by temperature, the primary control parameter.

In mercury catalyzed dissolution H_2 evolution can be a problem under certain dissolver conditions. To dilute the hydrogen, a controlled air purge is maintained at a minimum measured flow rate that is calculated to dilute the maximum hydrogen evolution rate.

3.2.3.3 Head End

Before solvent extraction, the dissolver solution is clarified with gelatin to remove impurities that form emulsions that cause upsets in mixer-settler operation (2). Agglomeration of the silica impurities by gelatin is temperature sensitive. The temperature of the dissolver 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. Excessive concentration of process solution in the strike tank is prevented by an interlock that shuts off steam to the tank coils if the liquid level is too low or the tank static pressure is high.

When zirconium and niobium activity are high a MnO_2 strike is made. The rates of permanganate addition and agitation are controlled to avoid evolution of ruthenium.

When the flocculation is complete, the dissolver solution is 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 dissolver solution remains in the centrifuge. A 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. The dissolution of the centrifuge cake by the addition of nitric acid and sodium nitrate, which is only done periodically, can cause foaming and the overflow of the cake slurry to the canyon sump. This is minimized by maintaining the temperature as low as possible during dissolution and by adding sodium nitrate at a low rate. The segregated cooling water and the steam condensate are protected from coil failure by the Cash regulator. The coils are automatically pressurized with air at pressure in excess of the hydraulic head to prevent solution from leaking into the coil.

3.2.3.4 Solvent Extraction

In the mixer-settler banks, 1A, 1B, 2A, 2B, 1C, 1D, and 1E, the TBP-diluent extractant 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 aqueous outlet interface 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 Evaporation 3.2.3.5). To prevent the temperature of the organic from exceeding the flash point of the diluent, limiting control settings have been established for the 1A feed tank and the solvent storage tank (8).

A primary use of instrumentation in solvent extraction is to ensure that fissile material entering a mixer-settler bank is leaving at the same mass rate that it is entering.

The three mixer-settler banks, 1A, 1B, and 1C, in the first cycle, and the 1D and 1E, are equipped with neutron monitors to detect an accumulation of fissile material caused by: 1) circumvention of the process chemistry of extraction or stripping rendering the operations inefficient or ineffective or, 2) off-standard flowrates of the various streams entering a bank such as low solvent flow and high feed and scrub flow. Because of the sensitivity of the mixer-settler to an internal recycle of fissile material as a result of imbalanced flow rates, duplicate flow recording devices are used on all streams except the 1AF, 1DF, 1CX, and 1EX. The flows of 1CX and 1EX are measured by extra meters on the aqueous discharge (1CU and 1EU) of the banks in the canyon as well. Off-standard process chemistry can occur as a result of a significant change in specified values of the acidity or aluminum nitrate concentration of the stream entering a bank. A specific gravity instrument indirectly monitors the HNO_3 acid concentration of the 1A bank scrub by measuring the solution density. The same function is performed by a conductivity probe in the scrub stream to the 1D bank.

Any event causing an increase in the fissile material is detected by neutron monitors that alert the operators to an off-standard condition. A neutron monitor is positioned at the second stage of the 1A, 1B, and 1D banks and at stage 12 of the 1C and 1E banks.

The 1C bank and 1E bank are equipped with a colorimeter to detect an off-standard uranium concentration in the 1CU and 1EU respectively. Conductivity probes are also located in the 1CU and 1EU streams to detect any changes in acidity. High acid or an abnormal ratio of aqueous to organic flows will be manifest as a change in the readings on these instruments.

Solution temperatures in the mixer-settler banks are important with respect to highly efficient mass transfer of desired radionuclides. The uranium bearing organic from the 1B bank, 1BU, is stripped of uranium in the 1C 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. An uncontrolled scrub temperature causes an alarm before the flash point of the diluent in the 1BU is reached.

3.2.3.5 Evaporation

Batch level in the evaporator is a primary control parameter. Evaporator bottoms are monitored for specific gravity, temperature, and liquid level; an interlock automatically shuts off the flow of steam to the evaporator when the preset minimum limit is reached. In 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 the HNO_3 acid. In the other evaporators, the contents are transferred after solids concentration, as determined by specific gravity, and when the boiling temperature has reached the desired level.

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 an upstream mixer-settler bank. This reaction is known as a "red oil" explosion. Aqueous streams emerging from mixer-settler banks or centrifugal contactor banks are piped to decanters to remove gross amounts of organic. Interlocks within the 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 coil. Additional temperature-related safety interlocks that shut down the batch evaporators preventing the conditions conducive to a "red oil" explosion are as follows:

- High condenser outlet cooling water temperature,
- High reboiler steam temperature,
- Low liquid level, and
- High evaporator bubble cap column differential pressure.

The cooling water supply valve to the condenser must be 1/4 open before the operation of the evaporator can be initiated.

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

Air flowing through a leak is detected by a monitor in the process air supply to the coil that reads-out on a data logger system (that is currently being installed) and alarms. 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 eight alpha and beta-gamma radiation monitors in the cooling water discharge lines. A leak from a coil that introduces radioactivity into the segregated or circulated cooling water can be traced and the failure isolated before a large volume of cooling water becomes contaminated.

3.2.3.6 Anion Exchange

The neptunium and ^{238}Pu in the dissolved neptunium oxide targets are recovered in a batch operation of anion exchange equipment in the I-H and II-H frames. The instrumentation is appropriate to supporting the operations. The anion resin is degraded by the radiation exposure particularly from the Np and ^{238}Pu. The accountability of the quantity of Np and ^{238}Pu loaded on the resin is provided by dip tube bubblers that measure the liquid level and specific gravity of dissolver solution, product solution, and raffinates.

The OSRs for anion exchange limit temperature of the operation to 60°C (8). The I-H frame has a resistance thermometer in each of three resin columns individually removable by the canyon crane; the II-H frame has a resistance thermometer in the one resin column, also removable by the canyon crane.

3.2.3.7 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-Building dispatcher's 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.

3.2.3.8 Fire Detection

The fire detection instrumentation is discussed in detail in Section 3.2.5.

3.2.3.9 Nuclear Incident Monitor System

Nuclear Incident Monitor (NIM) system activates an alarm warning to personnel to evacuate the vicinity because of a criticality incident. This system is provided wherever fissile materials are stored or processed in a 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 farther than 100 ft from a potential incident location (13). A steady radiation rate between the limits of 0.5 and 3R/hr will sound the alarm within five seconds (13). An alarm shall also sound if the total dose received at the detector within 1 min exceeds 50 mR. However, for instruments in normally unattended, shielded areas, alarm rates shall be set well above the prevailing radiation background.

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 (13).

3.2.3.10 Data Logger System

A Data Logger System processes the data from: 1) the eight radiation monitor points on the cooling water discharge headers from the canyon vessels and 211-Building, 2) the air flow monitor on the Cash system, 3) differential pressure in the ventilation system, and 4) independent cooling water flow to vessels not equipped with air pressurization.

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 independent 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. H-Canyon receives power from two of these sub-stations (for detailed description see the H-Canyon Systems Analysis (2). In the H-Area, two 7500-kVA transformers reduce the line voltage to 13.8-kVA and feed the area supply loop. All area supply lines are pole-mounted except buried lines for emergency use between Building 221-H and an emergency generator in Building 292-H. The 221 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.

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. The 750-kVA transformer in H Area is center-leg grounded on the 460-volt side to the area grounding loop. All electrical equipment in Building 221-H is connected to this same grounding loop.

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-H electrical equipment is supplied power from motor control center panels except for the instrument air compressors that receive power directly from the emergency bus. Some equipment uses remote-operated magnetic starters that 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 instrument panels, gang valves, lighting, air compressors, and ventilation equipment in Building 221-H. If normal power to Building 221-H 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. The critical equipment on emergency power is listed in Appendix A.

Generators. The main emergency power source for Building 221-H is a diesel generator located on the [redacted] This diesel drives a 1000 kw generator to furnish 1600 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-H is distributed by the emergency bus in electrical Control Room [redacted] This bus can be connected to two sources of power:

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

The emergency bus in electrical Control Room [redacted] 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-H through the 1500-kVA transformer in electrical Control Room [redacted] Failure of this source causes the Building 221-H diesel generator to start automatically, and connect to the emergency bus. Feeders from the Building 221-H emergency bus are connected through circuit breakers to various motor control panels. To permit maintenance work on the Building 221-H emergency bus, the motor control panels that receive power from electrical Control Room [redacted] 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-H Building with three separate service systems: cooling water, fire water, and domestic 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-H 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-H*, cooling water is pumped through the area by five pumps in parallel. An auxiliary steam-turbine-driven pump is also provided that 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 H-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-H reservoir.

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*H-Area is additionally equipped with a smaller cooling tower, 285-3, to supply tritium facilities.

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-Lines and Building 221-H 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 282-2H pump basin and returns to the Building 285-H 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 281-4H activity monitor, 281-1H delaying basin, and 281-2H pump basin, and then returns to the Building 285-H reservoir.

The segregated cooling water system accumulates from canyon vessel coils that are supplied by both cooling water and steam. The segregated water is not reused; it is discharged to the creek. 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. If there is a coil leak in one of these vessels, chances of getting contamination in the return stream are greater because of the possibility of a vacuum being formed within the coil that might draw in contamination through any coil leak. Segregated cooling water flows from warm canyon vessels to an underground header west of Building 221-H. Segregated water from hot canyon vessels discharge to a condensate return header in the hot gang valve corridor. This header leaves the gang valve corridor through the east wall of Building 221-H, and ties into an underground header. The two underground segregated cooling water headers join south of Building 221-H, and the water flows underground to the 281-6H activity monitor. From here, the water flows through the 281-5H 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-H 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 that 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 electrically operated isolation valves operated from the control room. On each of these electrical valves, there is a STOP button that 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 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 suplies 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 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 ft 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-H 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 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.

3.2.5.2 Steam Supply

3.2.5.2.1 Generation

Steam is generated by three coal-fired, stoker-fed boilers in Building 284-H. Each boiler has a design capacity of 60,000 lb/hr of saturated steam at 325 psig. Service water which is treated by anion and cation exchange resins is fed to the boilers by one electric-motor-driven pump and three steam-driven pumps. De-aerating heaters preheat boiler feedwater before it goes to the boiler feed pumps.

3.2.5.2.2 Distribution

The 325 psig steam is distributed through a 12-in header to pressure reducing stations for the buildings. For Building 221-H, 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. C-Area, H-Area, and F-Area are tied together on a common steam header. C-Area and D-Area, in particular, supply steam to H-Area.

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 pressure regulator valve into a 2-in header that enters the building and runs along the ceiling of second level to Section 7. There are cutoff valves at the reducing station and at Sections 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 solution 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 that 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.

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-H. Its only use in the canyons is on the dissolver off-gas filters in Sections 6 and 7 of the hot canyon and on the heat exchangers for cold feed streams to solvent extraction.

The 15 psig steam is reduced from a 150 psig supply at two reducing stations outside the south end of Building 221-H. 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 that 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-H.

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. For additional information, see Reference 2, Section 1.3.2.6.

3.2.5.4 Fire Protection Systems

The Savannah River Plant has a Fire Protection Division that 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.

Temperature measuring instruments, resistance temperature devices (RTD's), 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.

Heat sensed by these detectors will alert operating personnel by an alarm at the data logger in H-Area. Deluge valves (Figure 3-16) can be manually opened in the affected building section. Each deluge valve can be opened by pushing buttons in the control room.

3.3 ENGINEERED SAFETY FEATURES

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

3.3.1 Ventilation Systems

Building 221-H is supplied by three ventilation systems: the canyon air system, the center section air system (also known as central air), and the gang valve corridor air system. Air supplied by these systems is exhausted by five separate systems: canyon air exhaust, center section exhaust, B-Line

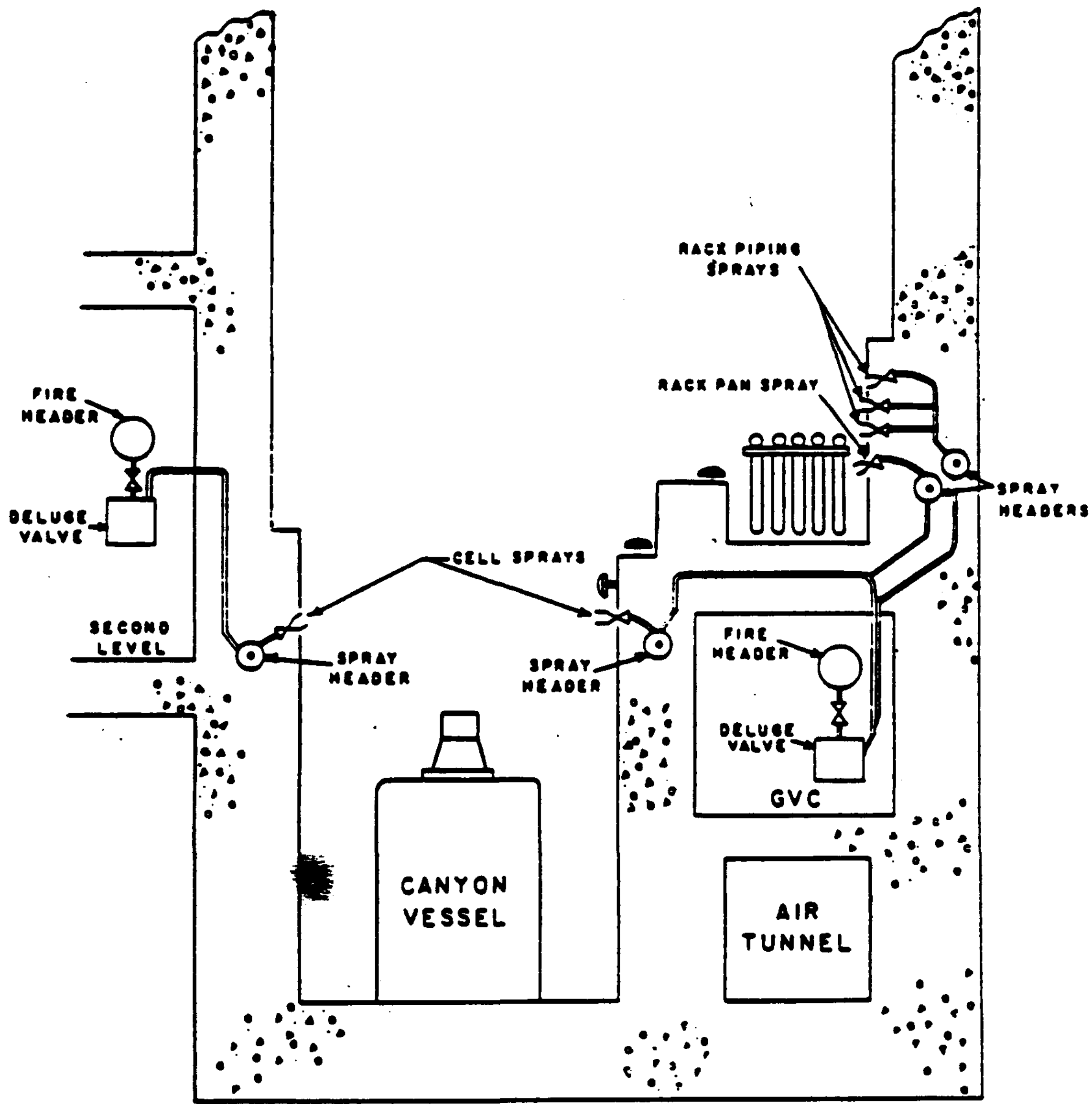

FIGURE 3-16. Canyon Spray System - Typical Section

exhaust, process vessel and vent system, and recycle vent system. The data logger indicates pressure differentials in various areas.

The ventilation system in Figure 3-17 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-H. 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 which will start automatically. An alarm will sound in the control room in case of mechanical or electrical failure of an on-line 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-H, all supply blowers stop, and the dampers for all blowers automatically open. This allows the Building 292-H exhaust fans to maintain the normal air flow pattern through the canyons. In case of power failure to Building 292-H, all canyon supply blowers automatically shut down.

In case of loss of instrument air in Building 221-H, the canyon supply blower that is on line continues to operate, and all the dampers on all the supply blowers automatically open. Each blower is equipped with a manually operated damper that 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-H drops to less than a 0.75-in water column.

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.

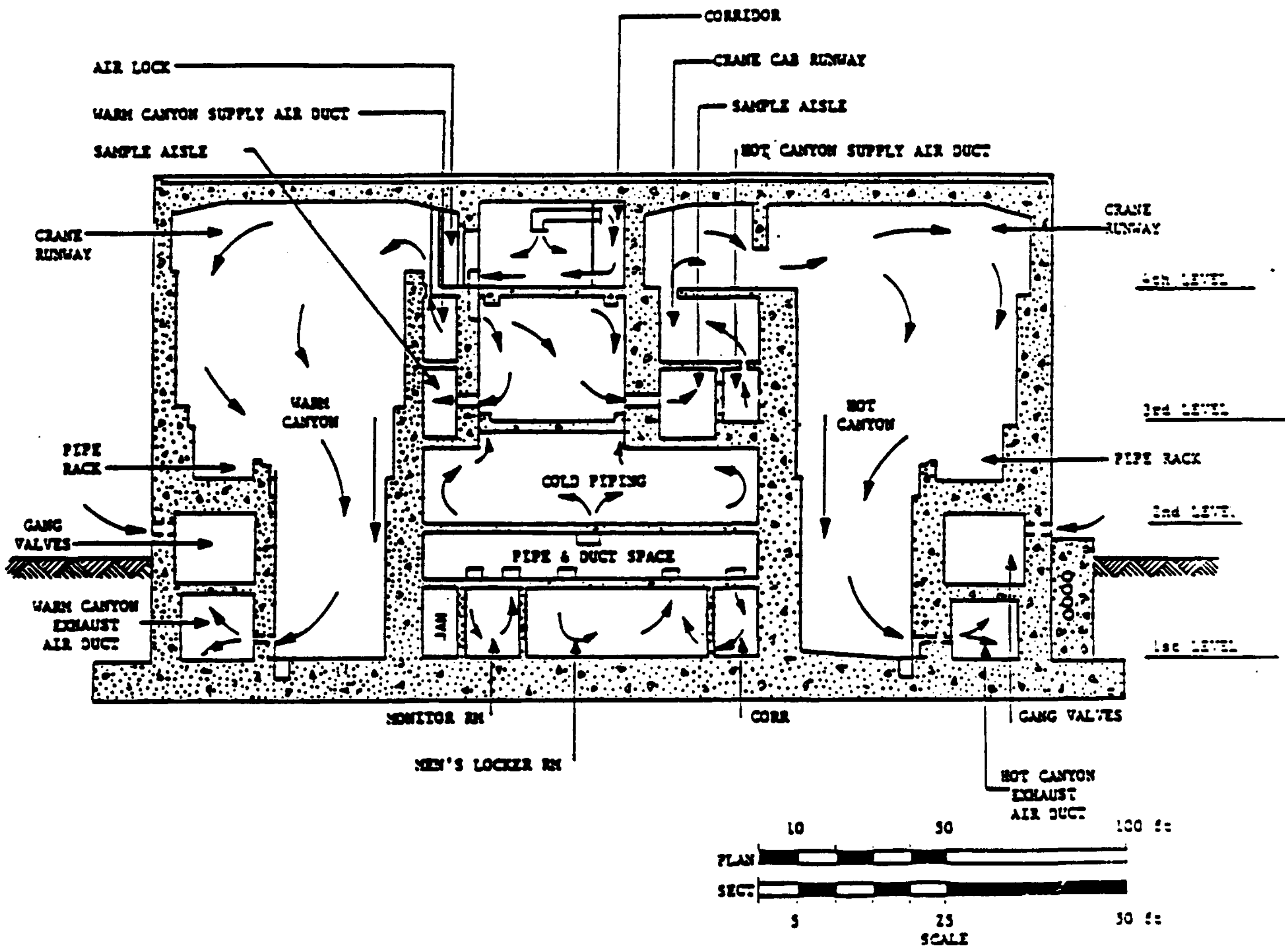

FIGURE 3-17. 221-H Ventilation Air Flow

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 0 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 that 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-H 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. These fans, if set on automatic control, will operate on emergency power in the event of normal power failure but, if set on hand control, will not start on emergency power. Failure of any booster fan will activate an alarm in the dispatcher's office. 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, EP 47, 48, 49 and 50. 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-H, 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-H 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 in order 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 that close if the blower fails, but 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 that 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.

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 that 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-H 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.
Sample aisle	- Should be negative with respect to third level.

TABLE 3-2. Differential Air Pressure

Between		Inches Water Gauge	
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

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.

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

Exhaust routes are shown in Figure 3-18.

Sand Filters. These filters, east of Buildings 221-H and 292-H are rectangular concrete structures 240 x 100 x 16 ft. [redacted] supplies 12 distribution laterals under original filter and 17 distribution laterals that run under the new filter that was constructed in 1976. The new filter has a stainless steel grid placed over the laterals to provide support for the filter media. The first layer of filter bed is coarse stone; succeeding layers are of decreasing-size gravel and sand, adding 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.

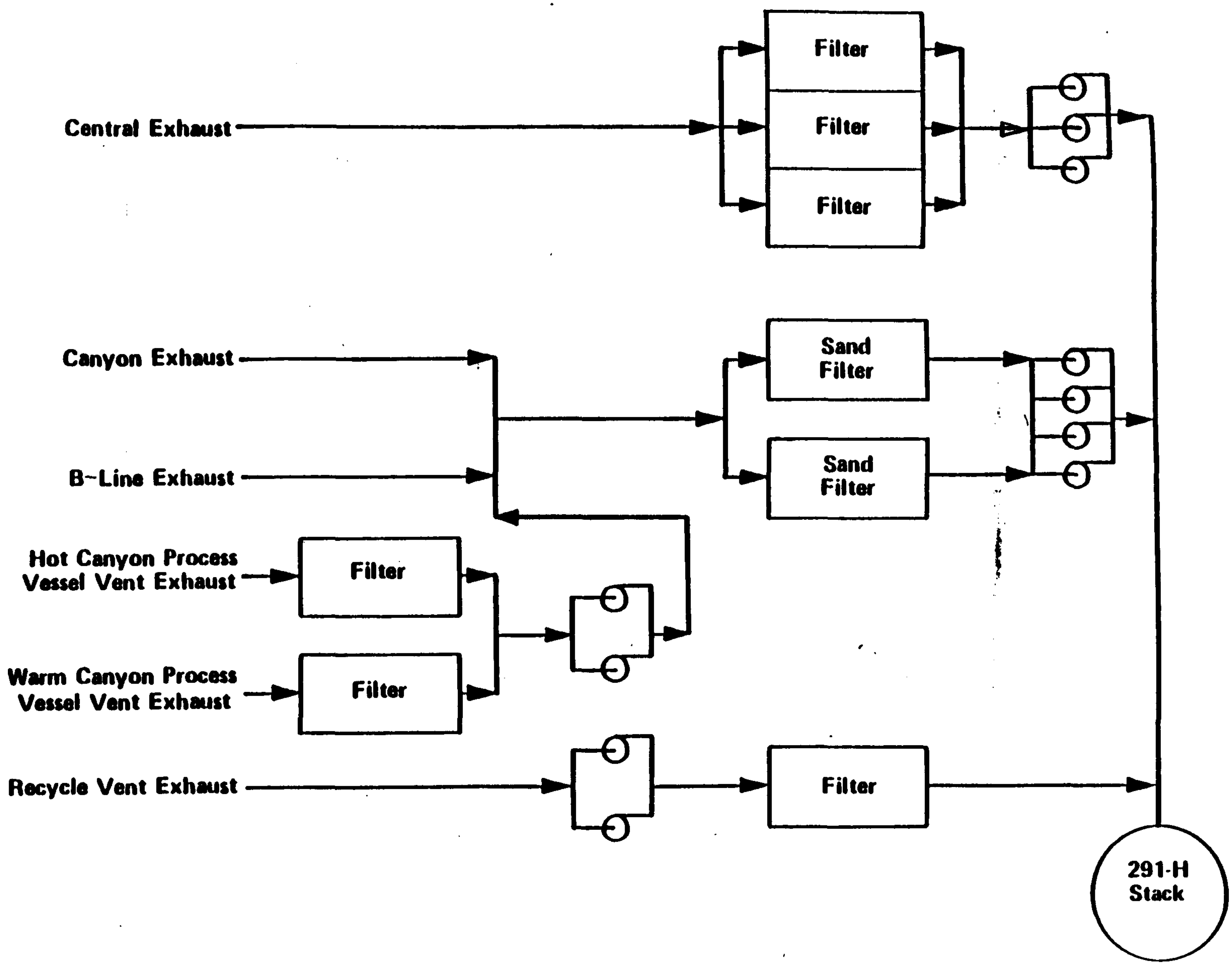

FIGURE 3-18. Exhaust Routes for Ventilation Air

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 [redacted] respectively. When the sumps become filled, as indicated by local gauges, they can be jetted or pumped to Transfer Tank 805. [redacted] drains to a sump just outside Building 221-H that has a high-level alarm in the hot canyon control room.

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

In 1969, some of the internal supports in the 294-H 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.

Exhausters. Air is exhausted from the hot and warm canyons and from the B-Line by four fans located in Building 292-H. Each 300-hp fan is rated at 70,000 cfm. Two of these units are normally in operation and the other two are on standby; they come on-line if the vacuum drops to less than 1.0-in water column. This also activates an alarm in the dispatchers office and an alarm in the Building 211-H control room. If the vacuum drops to less than 0.75-in water column, the canyon supply fans in Building 221-H are automatically shut down.

Emergency power from the 600 kW generator in Buildling 292-H is available to three of the canyon exhaust fans. In addition, two of the fans are served by dedicated diesel-driven generators.

Exhaust Stack (Building 291-H). The 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. The stack is 200 ft high. A high level alarm on this tank is located in the dispatcher's office. 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 dispatcher's office. This alarm can also mean that Tank 602 has overflowed to its sump.

A small blower located just south of Building 291-H 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-H 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-H 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 0.35-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-H emergency generator.

Central exhaust filters in Building 292-H are the "Chemical Warfare Service" type (CWS) and are installed as three separate units operating in parallel. This system is designed to operate on two units, which makes it possible to remove a unit from service to make repairs or change filters without shutting down the system.

Each bank of filters is equipped with motorized dampers on the inlet and outlet sides for isolation when changing filters. A differential pressure manometer is located across filters which indicates a pressure drop across the unit. Filters are replaced when operating two units if pressure drop across filters increases to 2.5-in H_2O.

3.3.1.2.3 Process Vessel Vent System

Canyon vessels are maintained at a slight vacuum with respect to the canyons by two exhaust fans with 40-hp motors installed in Section 5W of Building 221-H. One of these fans operates while the other is an installed spare which comes on-line automatically if the operating fan fails. Each fan has a capacity of 5,000 cfm of air at a static pressure of 30-in of water. Normal air flow is 4000 cfm. Only Section 6H dissolvers and their associated off-gas reactors and filters are not connected to the process vessel vent system.

Canyon air flows into process vessels through the overflow lines and then into one of two vessel vent headers. These 18-in headers are 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. The warm canyon air is dehumidified by heating to 50 to 60°C to prevent nitric acid condensation in the filter, then filtered in Module 5.7. The hot canyon air is scrubbed with process water in Module 7.3 to partially remove particulates, primarily ammonium nitrate. This air is then filtered in Module 7.2. These filters (Figure 3-19) typically remove 95 to 99% of the radioactivity present.

FIGURE 3-19. Vessel Vent Scrubber

After filtration, the vessel vent air is routed to the vessel vent fans in Section 5W. These fans obtain suction through an 18-in diameter inlet header supplied by 12-in diameter ducts from the warm and hot canyon vessel vent filters. Each 12-in suction duct has an electrically-operated butterfly damper valve that can be adjusted to balance the air flow from each canyon as desired. Each fan discharges into a separate 12-in diameter duct that protrudes through a core-drilled hole in the west wall of the warm canyon into the warm canyon exhaust tunnel, where the air is picked up by suction of the sand filter exhaust fans, passed through the sand filter, then discharged to atmosphere through the stack. These fans will be run to destruction, then replaced as necessary; all maintenance work on this system is done remotely from the warm crane.

Three air sampling stations are located in the warm canyon sample aisle to obtain filter paper samples of the vessel vent air upstream and downstream of the warm canyon filter and downstream of the hot canyon filter. These sampling stations are normally operated continuously with the filter paper changed and read once per shift. Sampled air is exhausted to Section 5W.

3.3.1.2.4 Recycle Vent System

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

Third level and cold feed prep tanks connect to the recycle vent header. This vent header goes out of Building 221-H through the center section exhaust air tunnel and proceeds to exhausters in Building 292-H. The vent header from Building 211-H area joins the Building 221-H header in the air tunnel.

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-H.

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 281-8H (14). 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 (281-6H) to delaying basin 281-5H. This basin has a capacity of 380,000 gal and is divided into two sections, 281-5H east and 281-5H west. When one of the sections fills, it is emptied into Four Mile Creek, if the activity is within limits. About a 5-hr 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 281-8H 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-H cooling tower reservoir. In case of low level at 285-H, 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 281-6H. 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 281-4H. 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 depleting the water supply.

When an alarm for the circulated water system is received in the control room, the supervisor and Health Protection representative at Building 211-H are contacted immediately. The alarm is promptly investigated at Building 281-4H by the Building 211-H 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-H control room, 281-4H monitor building, or at the 281-1H 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 281-8H is a lined excavation which receives aqueous waste streams for temporary storage and which has a capacity of four 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. Most sections are separated from each other by a concrete or stainless steel fire wall, and 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.

The concrete roof 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 H-Canyon Systems Analysis (2).

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.

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 transferred 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.

Sump solutions are transferred through rack piping to a canyon tank for sampling and further processing.

Additional provisions for confinement of spills is provided by pans under some of the vessels. The bottom of each frame has been completely enclosed with stainless steel sheet metal to form a 450-gal sump for the collection of overflow or spillage from within the frame, thereby protecting other canyon processes against contamination with ^{238}Pu. Stainless steel pans are also installed under the high level waste evaporators to improve the bearing surface for equipment and protect the floor against corrosive spills and leaks.

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.

3.4.1 Remote Operation

The capability to remove canyon process equipment remotely and place it into a decontamination area for decommissioning is in place. This is particularly desirable for equipment that is contaminated with materials which emit highly penetrating radiation. All types of canyon equipment 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.

3.5 REFERENCES

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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 H-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 (see Section 4.1) after review throughout the organization. Decisions which 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).

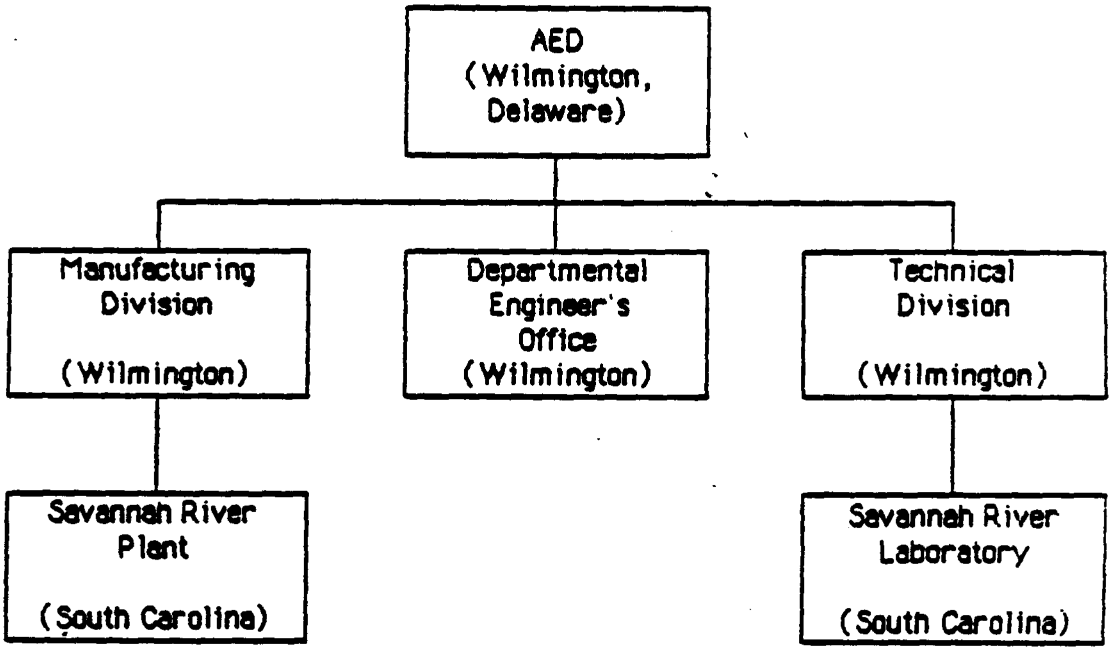

FIGURE 4-1. Atomic Energy Division Management

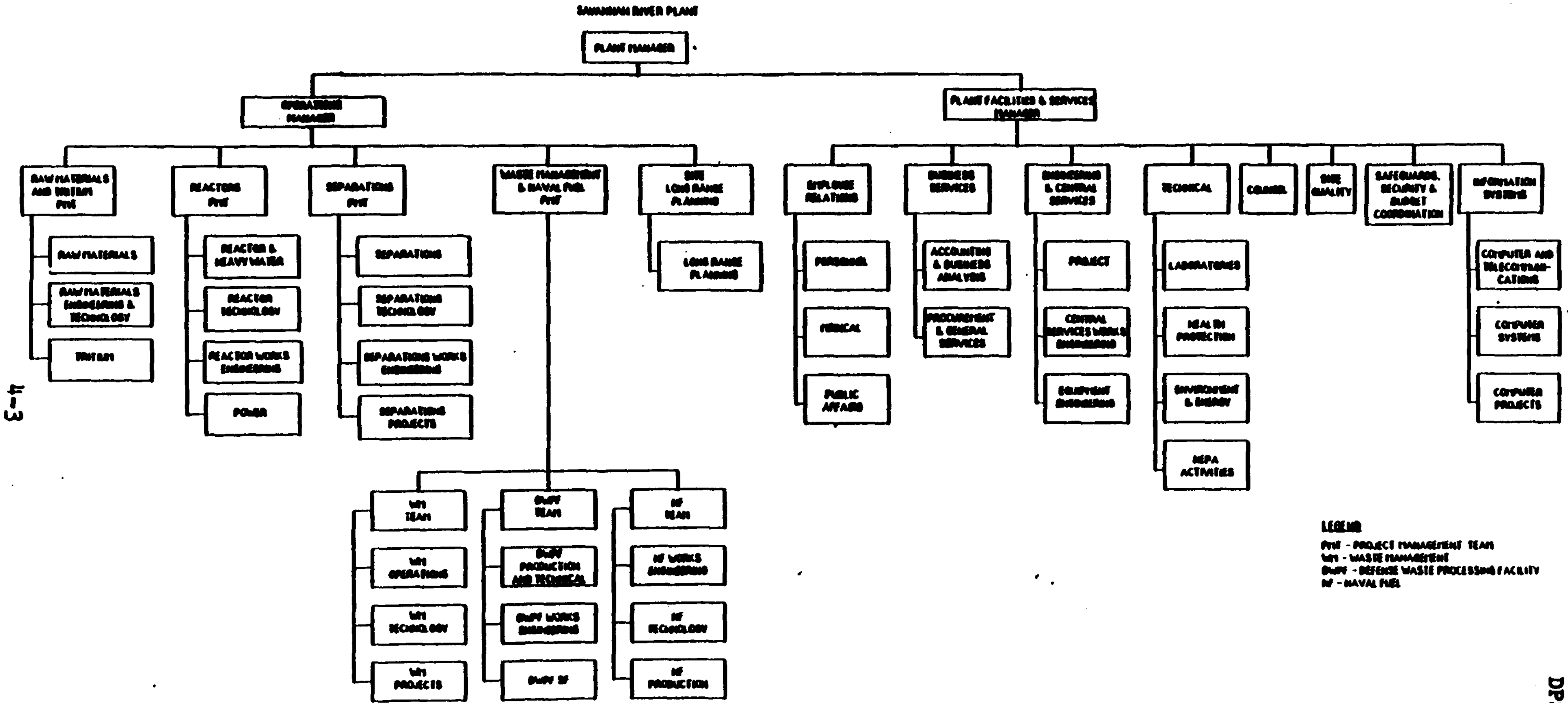

FIGURE 4-2. Organization of Savannah River Plant

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:

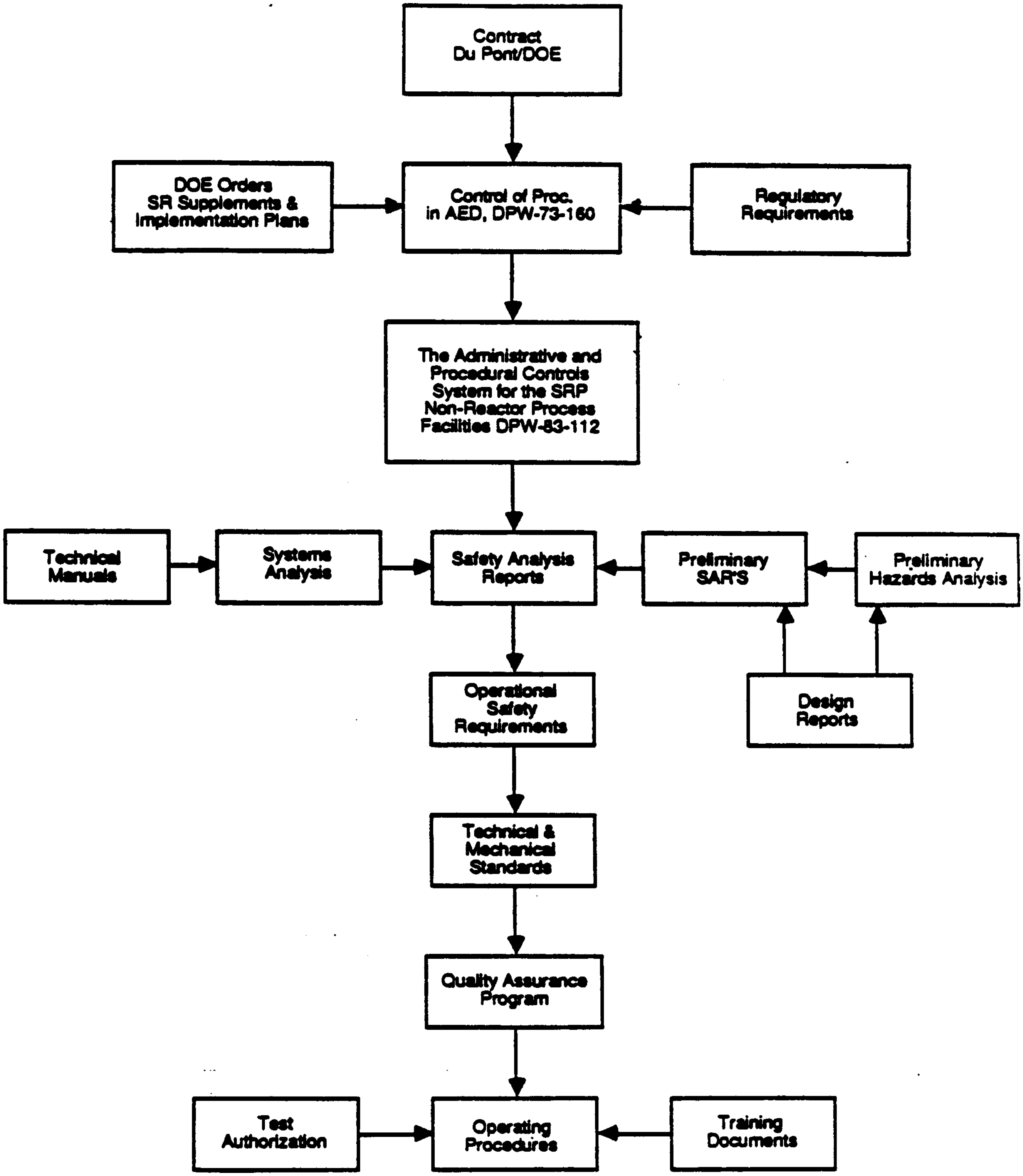

FIGURE 4-3. Document Requirements and Procedures Flow Chart

- 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 DPSOPs, 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-min 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 H-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.

<u>Nuclear Incident</u>. 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 reentry.

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.

<u>Offsite Warning Plan</u>. 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 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 Department 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 continually 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 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.

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

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

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 (see for example discussion of "red oil" explosion in Section 3.2.2.8.5), 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-H 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-H 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 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 4-2. Significant Radionuclides Present in Gases and Vapors from 200-Area Operations

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

*May be significant from a criticality accident.

[a] 1.7E7 = 1.7×10^{7}.

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 H-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, 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 200-Area releases are listed in Table 4-5.

4.8.2 Effluent Treatment and Control

Liquid Effluents. There are five normal sources of liquid effluents from 200 H-Area, as shown in Figure 4-4. Generally, 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 ^{3}H, ^{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 is included in the Seepage and Retention Basins SAR (11).

4.8.3 Effluent Monitoring Program

All effluents from 200 H-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 H-Area liquid effluents are shown in Figure 4-4. Each system is separately discussed below:

The 221-H Canyon building has two cooling water systems. The Circulating Cooling Water is monitored continuously in the 281-4H 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-H dispatcher's office.

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	28.10 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 yrs
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

[a] 1.62E5 = 1.62×10^5.

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

Isotope	Annual Release Activity, Ci 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

[a] 4.3E5 = 4.3×10^{5}; 2.7E-3 = 2.7×10^{-3}.

Source: Environmental Monitoring at the Savannah River Plant, Annual Report 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

	Annual Release Activity, Ci		
Isotope	1983	1984	Average
^{3}H[a]	1.5E2[b,c]	2.0E2[c]	1.8E2
$^{89,90}Sr$[a]	--	--	--
$^{134,137}Cs$	1.3E-1	2.2E-1	1.8E-1
Alpha	5.8E-3	3.0E-2	1.8E-2
Other Beta-Gamma	3.1E-2	8.9E-2	6.0E-2

[a] Primarily groundwater migration from F- and H-Areas Seepage Basins.

[b] $1.5E2 = 1.5 \times 10^{2}$.

[c] Surface transport component.

Source: Environmental Monitoring at the Savannah River Plant, Annual Report for 1984, DPSPU-85-302, June 1985.

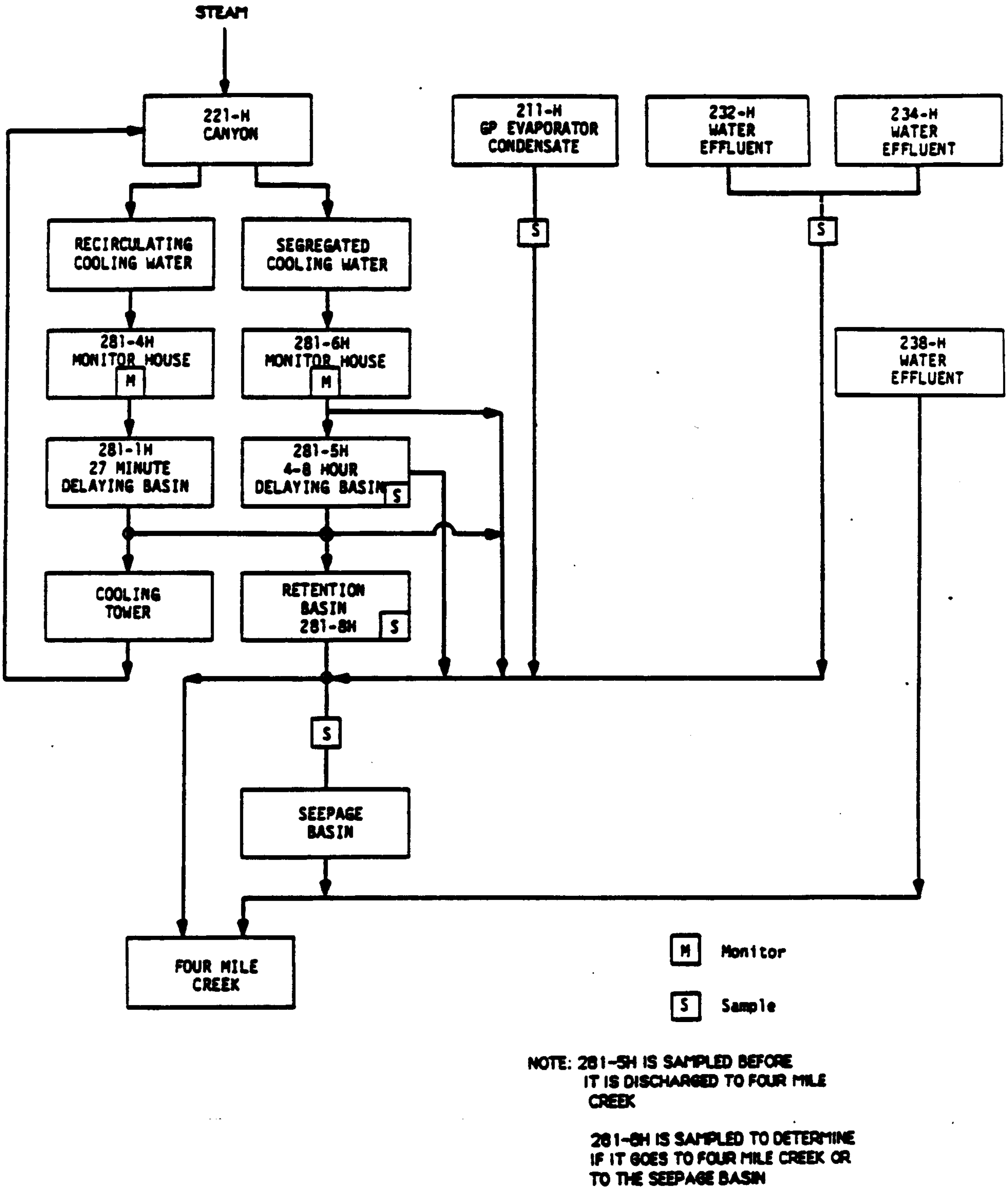

FIGURE 4-4. 200 H-Area Liquid Effluents

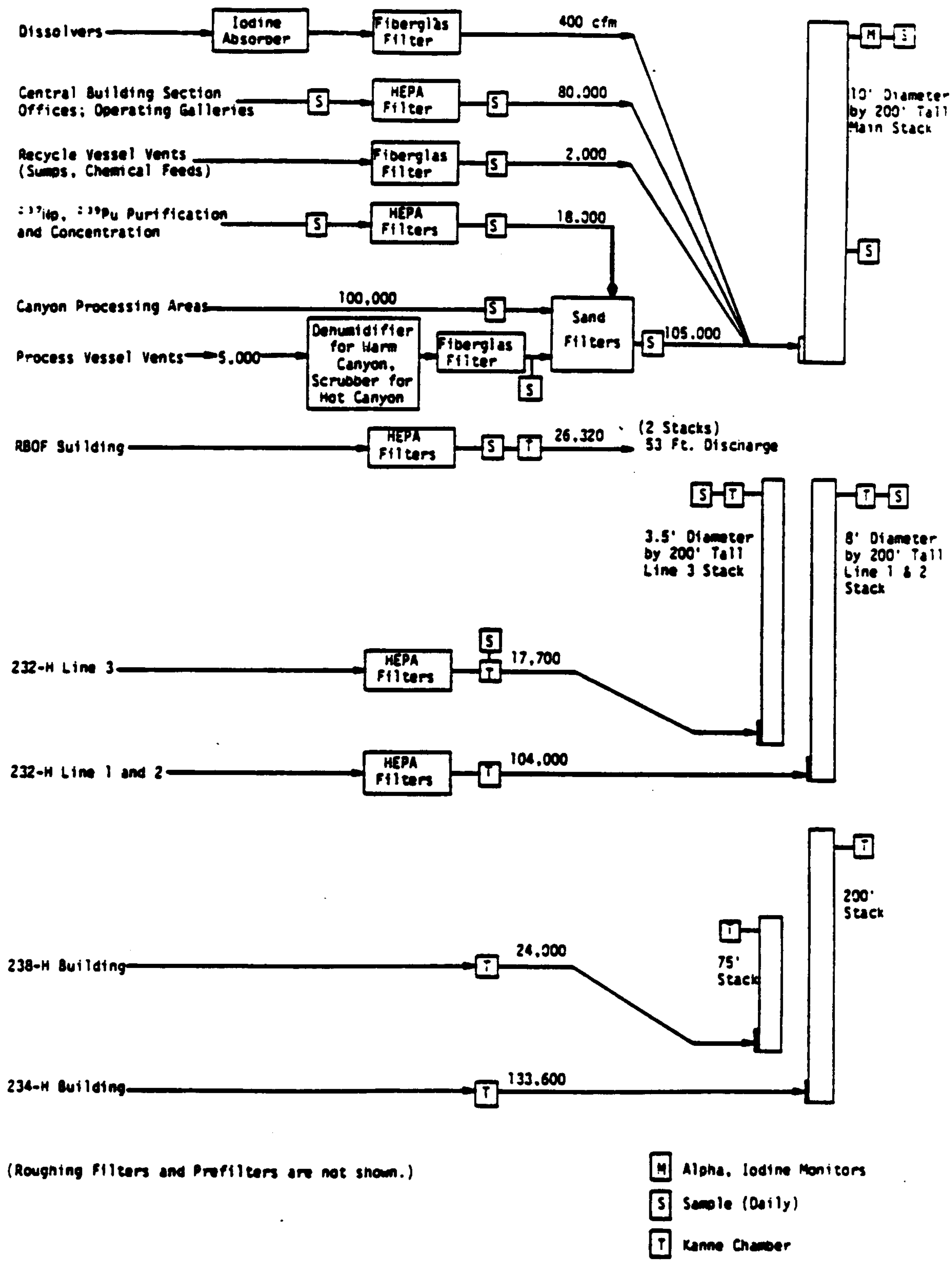

FIGURE 4-5. 200 H-Area Airborne Effluents

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-H 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 281-6H monitor house. A recorder for these monitors is located in 221-H 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 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 281-8H 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 from the 200 H-Area are shown in Figure 4-5. 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-H 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.

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 Separations 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 populations.

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 maximum individual dose from atmospheric releases, POPGASP 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

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
^{3}H	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
$^{89,90}Sr$	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
$^{235,238}U$	2.9E-7	2.3E-2
^{238}Pu	2.9E-7	3.2E-2
^{239}Pu	1.2E-7	1.3E-2
$^{241,243}Am$	4.0E-8	3.8E-3
$^{242,244}Cm$	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
^{3}H	7.1E-7	3.1E-2
$^{134,137}Cs$	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

[a]Doses assume conversion factor equivalent to ^{137}Cs.

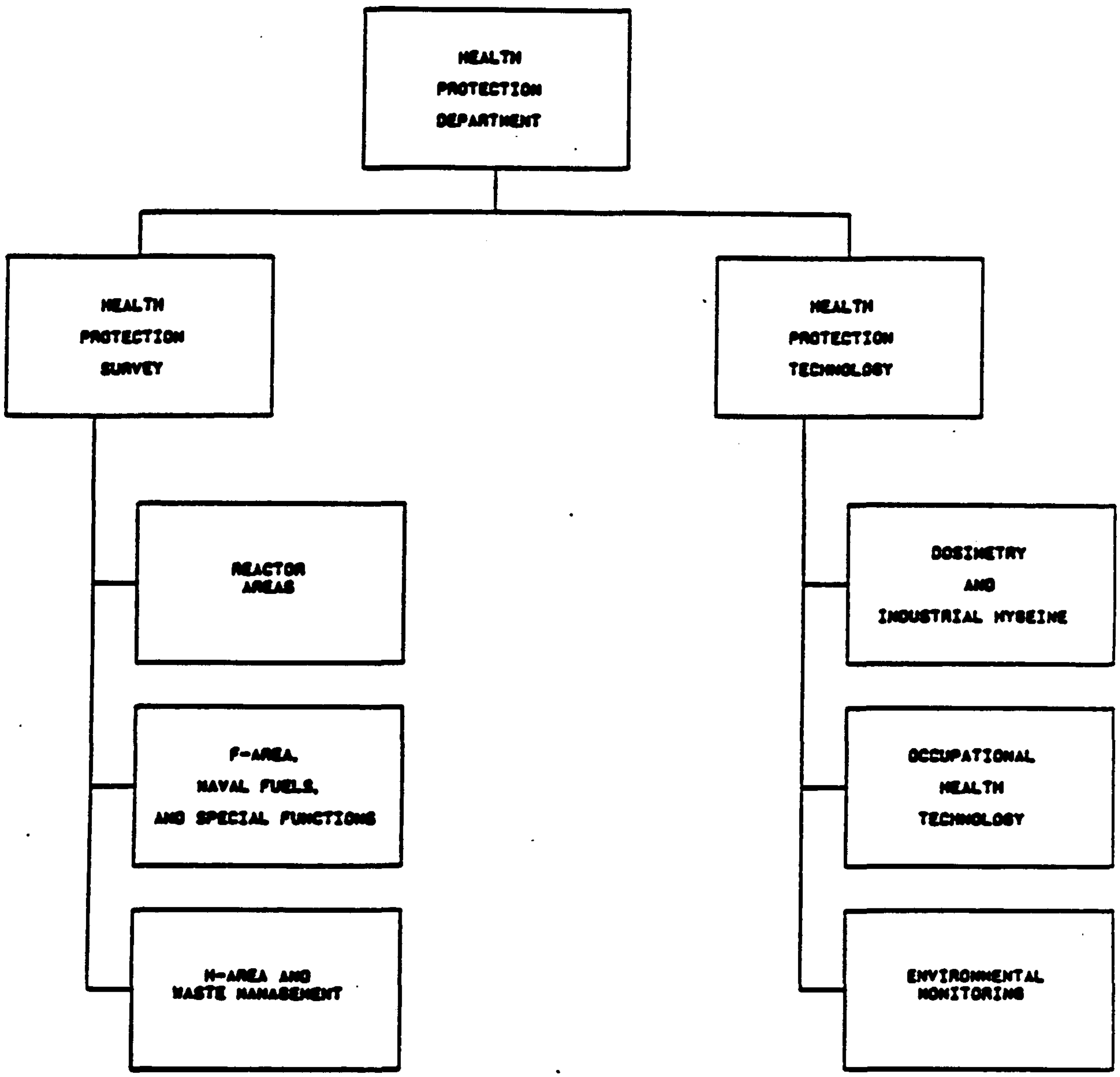

FIGURE 4-6. Health Protection Department Organization

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

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 H-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-H 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, transferable) is low. A Radiation Zone (RZ) is an area where the radiation or contamination levels exceed the limits for a Regulated Area.

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. Values obtained from Health Protection records for annual exposure lists for various occupational groups for 1984 are listed in Table 4-10.

TABLE 4-9. Plant Radiation Dose Guide Values[a]

Type Exposure, Organ	Dose in Rems Per Qtr	Dose in Rems 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
	Plant Emergency[b]	Life Saving
Emergency Exposure:		
Whole Body	25	100
Hands	100[c]	200[d]

[a]Values at or less than those given in Department of Energy, Environment, Safety and Health Manual, Chapters XI, XII.

[b]Involving protection of property or personnel.

[c]Includes whole body exposure.

[d]In addition to whole body exposure.

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

Employee Group	Number in Work Group	Annual Average Exposure (rems)	Annual Total Exposure person-rem/yr
Separations	134	2.04×10^{-1}	27.4
Maintenance	19	1.84×10^{-1}	3.5
Electrical and Instrumentation	18	1.83×10^{-1}	3.3
Health Protection	20	4.65×10^{-1}	9.3
Separations Technology	27	3.3×10^{-2}	0.9
Total			44.4

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

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

[a]Based on 40-hr week.

[b]The 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 ^{244}Cm to 6.0E-13 for ^{248}Cm. Because of the low yield of ^{248}Cm (0.05 weight percent, mass abundance), the RCG for unidentified alpha isotopes will be followed for all isotopes of curium.

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 H-Area Central Safety Committee is composed of a representative from each of the departments regularly working in the 200-H 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 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

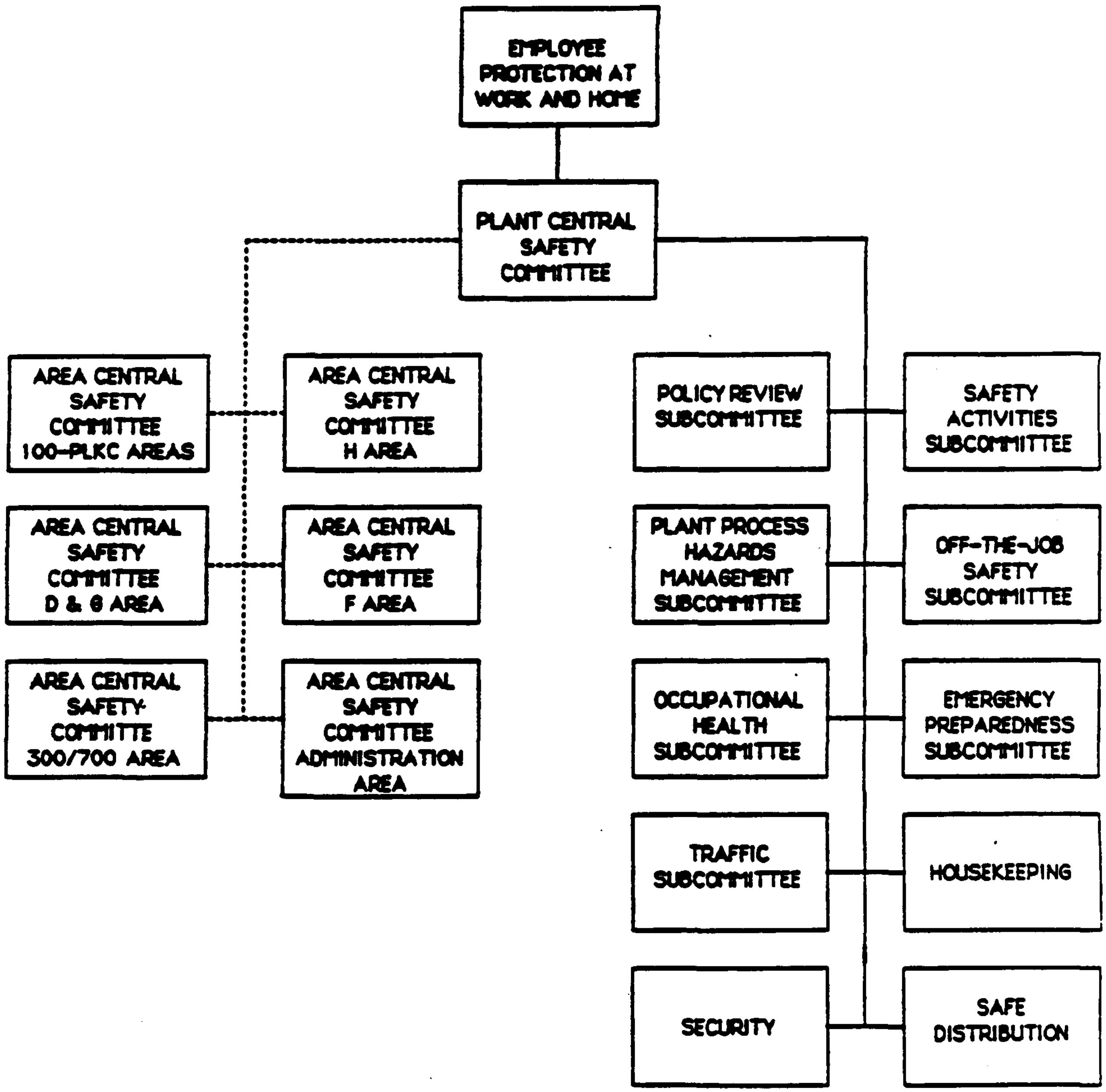

FIGURE 4-7. SRP Safety Committees

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 Preparedness	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").

Department. The Fire Protection Group maintains adequate mobile fire fighting equipment, trains personnel, and inspects protection equipment. Additional details on Fire Protection are given in Section 3.2.5.4.

4.9.6 Criticality Control

There are many places in the 221-H 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-H 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 H-Area Facilities 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,

TABLE 4-13. Technical Standards Containing Limits for Isotopes Handled in 200 H-Area Facility

Facility and/or Operation	Document No.	Title
General	DPSTS-NIM	Nuclear Incident Alarm Systems
221-H, ^{235}U	DPSTS-221-H-HM-0.06	Nuclear Safety
221-H, Pu	DPSTS-221-H-Pu A1-0.06	Canyon Nuclear Safety
221-H, Pu	DPSTS-221-H-Pu-A1-0.09	B-Line Nuclear Safety
221-H, ^{233}U	DPSTS-221-H-INT-23-0.06	Canyon Nuclear Safety
221-H, ^{233}U	DPSTS-221-H-INT-23-0.09	B-Line Nuclear Safety
221-H, Th	DPSTS-221-H-TH-0.06	Canyon Nuclear Safety
221-H, Th	DPSTS-221-H-TH-0.09	B-Line Nuclear Safety
221-H, U, Pu	DPSTS-221-H-Pu-U-0.06	Canyon Nuclear Safety
221-H, U, Pu	DPSTS-221-H-Pu-U-0.09	B-Line Nuclear Safety

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 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 min 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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14. Technical Standard Nuclear Incident Alarm System. Internal Report DPSTM-NIM-78, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, SC, January 1979.

5.0 ACCIDENT ANALYSIS

The H-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 H-Canyon facilities and operations described in Section 3.0 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 H-Canyon operations were identified and quantified from data in the 200-Area Fault Tree Data Bank (2). Other initiating events were identified from analysis 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 Base, 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 dose 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 H-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 H-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 due to normal releases due to H-Canyon operations.

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 H-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, 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 H-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 H-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-H.

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 H-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 prime 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 ~0.9 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 a 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 H-Canyon building will cause damage. It can be expected to completely disrupt the ventilation system and all external components such as the stack, filters, 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 may 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 measureable 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 H-Canyon that might impact H-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 H-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 H-Canyon process area can lead to a medium energetic event. Causes of fires with H-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 auto-ignition 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).

There have been no fires in ion exchange in H-Canyon since startup. Therefore, scenarios for this type of fire are based on a similar fire that occurred in F-Canyon in 1964 after degraded resin gradually accumulated atop a tank. This accumulation occurred because of a series of small leaks during transfers of either fresh, or spent resin slurries. (Note that the radioactive content of such resin is relatively small.) The resin may have ignited from friction heat around an agitator shaft or from a spark. This fire lasted for less than five minutes.

In the postulated H-Canyon fire, resin that is loaded with ^{238}Pu accumulates on a tank or on the canyon floor. Two mechanisms are assumed to contribute to this accumulation: a leak of resin from the column and an uncontrolled reaction in the column. The leak pathway is through the resin-retaining screen at the bottom of the column and then through a failed (leaking) connection in the elution outlet piping or in the waste outlet piping. The pathway for release of resin from the column in an uncontrolled reaction is through the column vent piping and seal pot. Accumulations of resins in canyon sumps from these release mechanisms are not considered in this analysis; the fire hazard there is mitigated by the practice of maintaining a minimum liquid level in each sump. Thus, the resin would not dry out in a sump, as it could on the canyon floor or on the top of the equipment as a result of the high heat generation of ^{238}Pu.

The risk of fires inside the columns is considered to be much smaller than the risks of fires outside the columns (discussed above) and, therefore, are not included. Loaded resin is maintained under liquid at all times to prevent resin dry-out. A fire in the column would be contained in the column; fumes would go to the vessel vent system.

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 unforseen 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. In the HM process, during the metal dissolution step, the noncondensible portion of the dissolver off-gas can contain up to 7% hydrogen by volume (air-free basis). To maintain the hydrogen content of the off-gas well below the lower explosive limit of 4 vol %, the off-gas is diluted by a continuous air purge in addition to any air inleakage.

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 eructation 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 the safe-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 (>2M) that destroys the HAN as it enters. Hydrazine nitrate and hydrazoic acid are not normally used in the HM process, but are sometimes present in the other process wastes that are blended with HM process wastes. Hydrazoic acid is hazardous in both the aqueous solution and the vapor phase, but hazardous concentrations cannot be reached by evaporating solutions containing less than 0.05M hydrazine.

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 H-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

$$HNO_2 + N_2H_4 \longrightarrow HN_3 + 2\ H_2O$$

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_2 at 30°C. Calculated estimates of the lower limit of hazardous concentrations of HN_3 show that sufficient NH_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, 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 neutralization of resin prior to discharge to the waste tanks, or the digestion of resin in alkaline permanganate, eliminates the hazard of resin-induced explosion in the waste handling facilities.

The safety of the anion exchange process for the recovery of neptunium and plutonium, 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 eructed 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. In the HM process, dissolution of enriched uranium requires a number of controls for the safe handling of ^{235}U. These controls involve the mass of uranium charged to the dissolver, the volume of dissolvent, and the heel of undissolved fuel. Administrative controls also avoid conditions that lead to precipitation of fissile material from solution. The principal mechanical control is the dissolver insert. The insert is necessary to maintain fuel tubes in a row so that any fuel tube fragments created during the dissolution are confined in a slot configuration and can be detected.

In addition, the concentration of ^{235}U in solution (22) must not exceed 11.5 g/l. This prevents criticality in the event that the dissolver solution is accidentally concentrated in a dissolver by evaporation from the normal volume to the minimum volume attainable with installed heating coils. The normal volume of dissolver solution is well in excess of the minimum volume by virtue of the quantity and concentration of nitric acid solution needed for complete charge dissolution.

Nuclear safety in HM dissolution also depends on control of the quantity of fuel element fragments (23) because fuel fragments may assume a configuration more reactive than tubes. A maximum height of residual fragments is measured by mechanically probing each insert compartment before fresh fuel is charged.

Precipitation of uranium from the dissolver solution represents a potential nuclear safety hazard. Administrative controls require that a minimum of 0.3M HNO_3 be present to avoid hydrolysis and precipitation of fissile material. Accidental addition of a precipitating chemical, such as NaOH, is prevented by a combination of nuclear safety blanks in process lines from certain cold feed tanks and by administrative controls.

<u>Head-End System</u>. The only conceivable mechanisms by which fissile materials could become concentrated to unsafe levels in the head end system are:

- Sorption on the agent that is added to clarify the raw metal solution
- Precipitation from solution
- Evaporation

In the HM process, evaporation of uranium solutions provides a mechanism for reaching an unsafe ^{235}U concentration in the head end system. To guard against this, the concentration is limited by temperature controls and by the presence of aluminum nitrate, which raises the boiling point as the solution is concentrated. Further, the total quantity of uranium that may be contained in the head end evaporator and its feed tank is limited to prevent the total in the evaporator from exceeding the "safe" loading above any bottom surface on which it might settle if precipitated or overconcentrated. The evaporator temperature instrumentation is interlocked to cut off the steam supply so the concentration of fissile material cannot exceed the maximum subcritical concentration. The specific gravity instrumentation gives an alarm and cuts off the steam supply if the specific gravity indicates overconcentration. The possibility of overconcentration in the strike step is even less likely because of the lower operating temperature, but equivalent safeguards are still provided.

Precipitation of ^{235}U in the head end vessel by the accidental addition of a precipitating chemical is a possible hazard. For the precipitation of uranium to become a nuclear hazard, at least a critical mass would have to be present, and enough of the precipitating chemical would have to be added to cause the fissile material to come out of solution. These two conditions would require, for example, that caustic be accidentally added to the head end tank normally containing large quantities of ^{235}U, or that a stream containing uranium be fed to a tank containing caustic, such as a waste neutralizer. A third condition requires that the precipitate settle so that it occupies less than one-half the original volume of solution, since concentration by a factor greater than two is required to exceed the safe concentration normally specified. Since the tanks contain agitators, this means that the precipitate would be kept suspended unless the agitator failed or was turned off. However, caustic is not normally piped to any tank that may contain unsafe quantities of uranium.

Solvent Extraction System. 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. 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 neutron monitors. However, there exists the potential to accumulate fissile material in unsafe concentrations through operational error or unforeseen process conditions.

Mixer-settlers, decanters, and surge tanks are areas of potential hazard. The 1A, 1B, and 1D banks in the HM process in H-Area (see Figure 3-9), 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 1A and 1D banks, refluxing can occur when the salting chemicals (such as nitric acid or aluminum nitrate) are left out of the scrub or when the flow rates of the various input streams are off-standard (such as low solvent flow and high feed and scrub flow). However, in each reflux situation, some product would be lost to the waste raffinate before unsafe concentrations are exceeded. In the 1C and 1E banks, the concentration of fissile material in the banks is directly proportional to the flow ratio of organic to aqueous streams. Because of the sensitivity of refluxing to flows entering the banks, duplicate flow recording devices are used on all streams except the 1CX and 1EX. The flows of these two streams are measured by extra meters on the aqueous discharge (1CU and 1EU) of the banks in the canyon as well as by the flow meters originally installed in the cold feed gallery. In addition, a specific gravity instrument is used to monitor the salting stream, 1AS, and a conductivity meter is used to monitor the 1DS stream. In addition to these process control items, duplicate instruments (e.g., colorimeters and neutron monitors) measure the concentration of uranium in specific stages. Many of these instruments, including the colorimeters and neutron monitors, are supplied with emergency power in the event of failure of normal power.

Although material balance checks and close adherence to a safe flowsheet are the primary controls, there are also the neutron monitors. 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 (24). Other instruments such as conductivity probes and/or colorimeters are also used to determine unusual conditions within the banks. In addition, end streams are routinely sampled and analyzed to determine if abnormalities exist. The feed rate of ^{235}U is also controlled.

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. In the 1A and 1D banks, the fissile-bearing solvent is diluted and removed from the banks by passing only the extractant stream through the bank.

<u>Evaporation</u>. There are three evaporators in H-Canyon that concentrate solutions containing fissile material: the 11.3E head end evaporator, the 17.2E rerun evaporator, and the 17.6E 1CU evaporator. Nuclear safety in these evaporators is maintained by control of the endpoint concentration and areal density. Evaporator bottoms are monitored for specific gravity, temperature, and liquid level; an interlock automatically shuts off the flow of steam to the evaporator when the preset endpoint is reached.

5.1.5 <u>Low Energetic Events</u>

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 Systems Analysis for the particular facility with the exception of a transfer error to the 211-H 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 year 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. One case in H-Canyon occurred during the past ten years that is considered significant. This occurred during scrap dissolving.

Mechanisms identified as potential contributors 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 if significant releases appear later, this 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 use. 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 adsorbed; 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 condensible vapors typically 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-H building.

Methodology used in this analysis and the hazards for various chemicals are discussed in detail in Reference 25.

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 odor.

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 facilities, the impacts 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

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

Name	Storage Solution Concentration,* %	Potential Hazard
Aluminum nitrate	34	Toxic
Boric acid	2.5	Toxic, carcinogen**
Ferrous sulfamate	40	Toxic NO_2 fumes
Gelatin	1	--
Hydrazine mononitrate	30	Toxic, carcinogen
Ion exchange resin	--	Fire
Kerosene	100	Fire
Manganous nitrate	50	Toxic
Mercuric nitrate	5	Toxic
Nitric acid	64	Toxic, corrosive
Potassium permanganate	6	Toxic, corrosive
Sodium carbonate	5	--
Sodium hydroxide	50	Toxic, corrosive
Sodium nitrite	30	Toxic, carcinogen**
Tributyl phosphate	100	--

*Lower concentrations may also be handled.

**Suspect carcinogen.

TABLE 5-3. Hazardous Chemicals

Name	AEL,* mg/m^3	Odor Threshold, mg/m^3
Boric acid	5	None
Hydrazine mononitrate	0.13	4
Mercuric nitrate	0.1	None
Nitric acid	5	9
Nitrogen dioxide	9	9
Potassium permanganate	5	None
Sodium hydroxide	2	None
Sodium nitrite	0.1	None

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

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.

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 recorded cases 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 of 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-H), the fan house (Building 292-H), and the 200-ft tall stack (Building 291-H). 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 evaporating. 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 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. In fact, the material in the filter is confined, which would also increase 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-H, 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 pumpout error can occur allowing activity to reach the system.

5.1.9.3 Seepage Basins

The H-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 26, and a complete analysis of the seepage basins is published in Reference 27.

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 (28,29). 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 rather than from the vessel into the coil or tube. 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 H-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 expected 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 determinations 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 on the basis for calculating distributions for materials released to the environment.

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 defined 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 which 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

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

	Isotopic Curie Fraction						
Nuclide	Raw Metal Solution	Head End	First Cycle	Second U Cycle	Second Np Cycle	Waste	Ion Exchange
^{89}Sr	0.71E-01	0.71E-01	0.83E-01	0.64E-01	0.77E-01	0.83E-01	0.11E-01
^{90}Sr	0.49E-02	0.49E-02	0.58E-02	0.45E-02	0.26E-02	0.58E-02	0.23E-02
^{90}Y	0.49E-02	0.49E-02	0.58E-02	0.45E-02	0.26E-02	0.58E-02	0.23E-02
^{91}Y	0.11E+00	0.11E+00	0.13E+00	0.10E+00	0.60E-01	0.13E+00	0.21E-01
^{95}Zr	0.10E+00	0.10E+00	0.12E+00	0.94E-01	0.54E+00	0.12E+00	0.41E-01
^{95}Nb	0.29E-01	0.29E-01	0.34E-02	0.26E-01	0.76E-02	0.34E-02	0.37E-01
^{103}Ru	0.24E-01	0.24E-01	0.18E-01	0.14E-01	0.14E-01	0.18E-01	0.20E-01
^{106}Ru	0.18E-01	0.18E-01	0.14E-01	0.11E-01	0.10E-01	0.14E-01	0.77E-01
^{106}Rh	0.18E-01	0.18E-01	0.21E-01	0.16E-01	0.95E-02	0.21E-01	0.77E-01
^{110}Ag	0.43E-03	0.43E-03	0.51E-03	0.39E-03	0.23E-03	0.51E-03	0.26E-02
^{123}Sn	0.44E-03	0.44E-03	0.52E-03	0.40E-03	0.23E-03	0.52E-03	0.44E-04
^{125}Sb	0.56E-03	0.56E-03	0.65E-03	0.50E-03	0.29E-03	0.65E-03	0.12E-02
^{127}Te	0.78E-03	0.78E-03	0.81E-03	0.70E-03	0.41E-03	0.81E-03	0.15E-02
^{129}Te	0.44E-03	0.44E-03	0.52E-03	0.40E-03	0.23E-03	0.52E-03	0.44E-04
^{134}Cs	0.11E-01	0.11E-01	0.13E-01	0.99E-02	0.57E-03	0.13E-01	0.16E-02
^{137}Cs	0.20E-01	0.20E-01	0.18E-01	0.13E-01	0.78E-02	0.18E-01	0.66E-02
^{141}Ce	0.22E-01	0.22E-01	0.26E-01	0.20E-01	0.12E-01	0.26E-01	0.72E-02
^{144}Ce	0.26E+00	0.26E+00	0.30E+00	0.23E+00	0.13E+00	0.30E+00	0.78E-01
^{144}Pr	0.26E+00	0.26E+00	0.30E+00	0.23E+00	0.13E+00	0.30E+00	0.78E-01
^{147}Pm	0.28E-01	0.28E-01	0.33E-01	0.26E-01	0.15E-01	0.33E-01	0.17E-01
^{148}Pm	0.29E-01	0.29E-01	0.58E-01	0.44E-03	0.26E-03	0.58E-01	0.15E-02
^{155}Eu	0.46E-03	0.46E-03	0.54E-03	0.10E-05	0.10E-05	0.54E-03	0.46E-04
^{234}U	0.12E-05	0.12E-05	0.13E-05	0.10E+00	0.10E-05	0.12E-09	0.97E-11
^{235}U	0.16E-07	0.16E-07	0.18E-07	0.14E-02	0.10E-05	0.20E-11	0.13E-11
^{236}U	0.14E-06	0.14E-06	0.17E-06	0.14E-01	0.10E-05	0.16E-10	0.12E-11
^{238}U	0.40E-09	0.40E-09	0.47E-09	0.37E-04	0.10E-15	0.40E-13	0.34E-11
^{237}Np	0.73E-07	0.73E-07	0.86E-07	0.27E-03	0.36E-03	0.80E-09	0.44E-04
^{238}Pu	0.27E-03	0.27E-03	0.31E-03	0.48E-04	0.28E-02	0.31E-03	0.50E+00
^{239}Pu	0.23E-05	0.23E-05	0.27E-05	0.42E-06	0.24E-04	0.27E-05	0.14E-03
^{240}Pu	0.17E-05	0.17E-05	0.20E-05	0.40E-06	0.18E-04	0.20E-05	0.86E-04
^{241}Pu	0.77E-03	0.77E-03	0.91E-03	0.14E-03	0.80E-02	0.91E-03	0.95E-02
^{242}Pu	0.34E-08	0.34E-08	0.40E-08	0.60E-09	0.10E-05	0.40E-08	0.10E-08
^{241}Am	--	--	--	--	--	--	0.95E-05

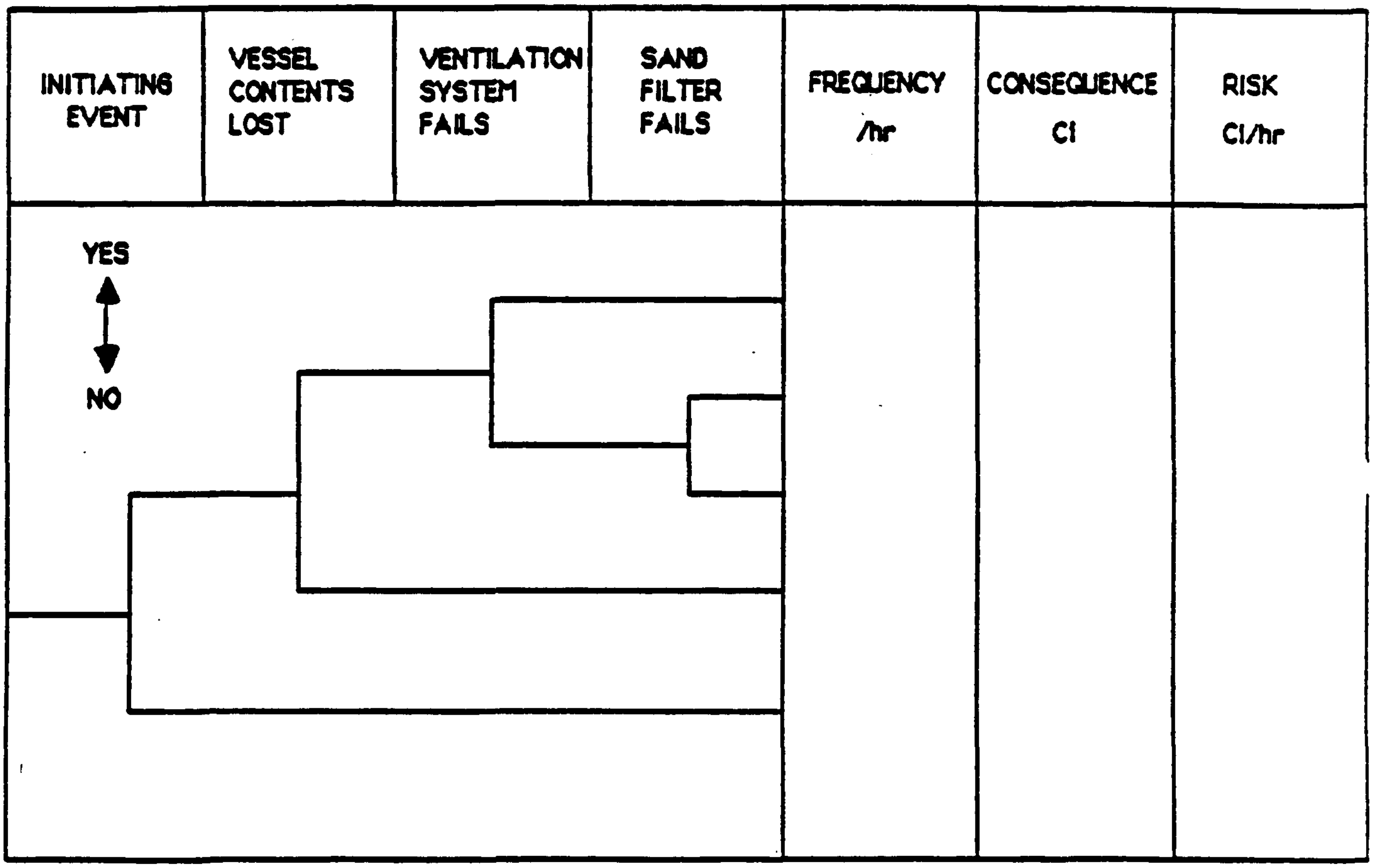

FIGURE 5-1. Example Event Tree for Airborne Releases

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.

5.2.4 Fault Trees

The fault tree method was used to estimate expected frequencies for events for which no data were 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 H-Canyon Systems Analysis (1) and utilized in this study 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 H-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 H-Canyon during an accident were evaluated for three populations 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.

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 transport models used are based on the U.S. Nuclear Regulatory Commission Regulatory Guide 1.145 (30). The exposure pathways considered in the AXAIR code include inhalation of radionuclides and gamma irradiation from the radioactive plume (31).

Radiation doses from inhalation of radionuclides in air depends on the amount of radionuclides released, the gaseous dispersion factor, the physical,

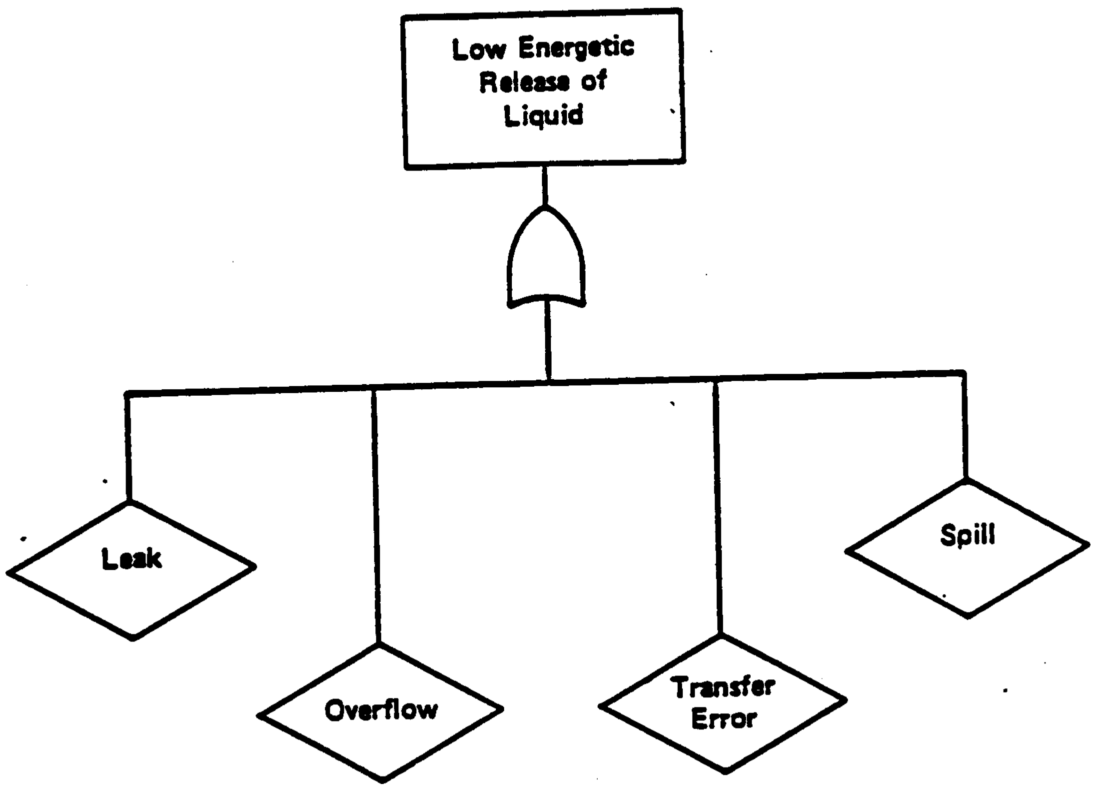

FIGURE 5-2. Example Fault Tree for Low Energetic Event Release of Liquid

chemical, and radiological nature of the radionuclides, and various biological parameters such as breathing rate. Standard sets of breathing rates for different age groups are used in the AXAIR code (31).

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 doses were calculated using a non-uniform Gaussian model because of its more realistic modeling compared to the conventional uniform semi-infinite plume model (32).

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, and therefore, the irrigation of food crops was not considered as a pathways 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 (33). 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 were used as specified in Regulatory Guide 1.109, with minor correction by the original author (34).

5.3 ACCIDENT FREQUENCIES

The expected frequencies for accidents leading to release of radioactivity were developed in the H-Canyon Systems Analysis (1) and are summarized in this section. The event trees for the 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 H-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 on plant at a frequency of less than 1 x 10^{-6}/yr (1.1 x 10^{-10}/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 with its orientation relative to prevailing tornado direction as well as in the tornado damage area (13). For H-Canyon (35), the probability of a damaging tornado strike is <10^{-8} year (<1.2 x 10^{-12}/hour).

5.3.1.2 Earthquake

Earthquake frequencies for SRP are shown in Figure 5-3 (36). The expected frequency of a MM VIII is 2 x 10^{-4}/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 37. 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 filter would not be expected to occur for an MM VIII earthquake (28).

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 (38). Based on the impact of a one ton meteorite at SRP, the estimated frequency of impact from a meteorite for H-Canyon is 1.5 x 10^{-15}/hr.

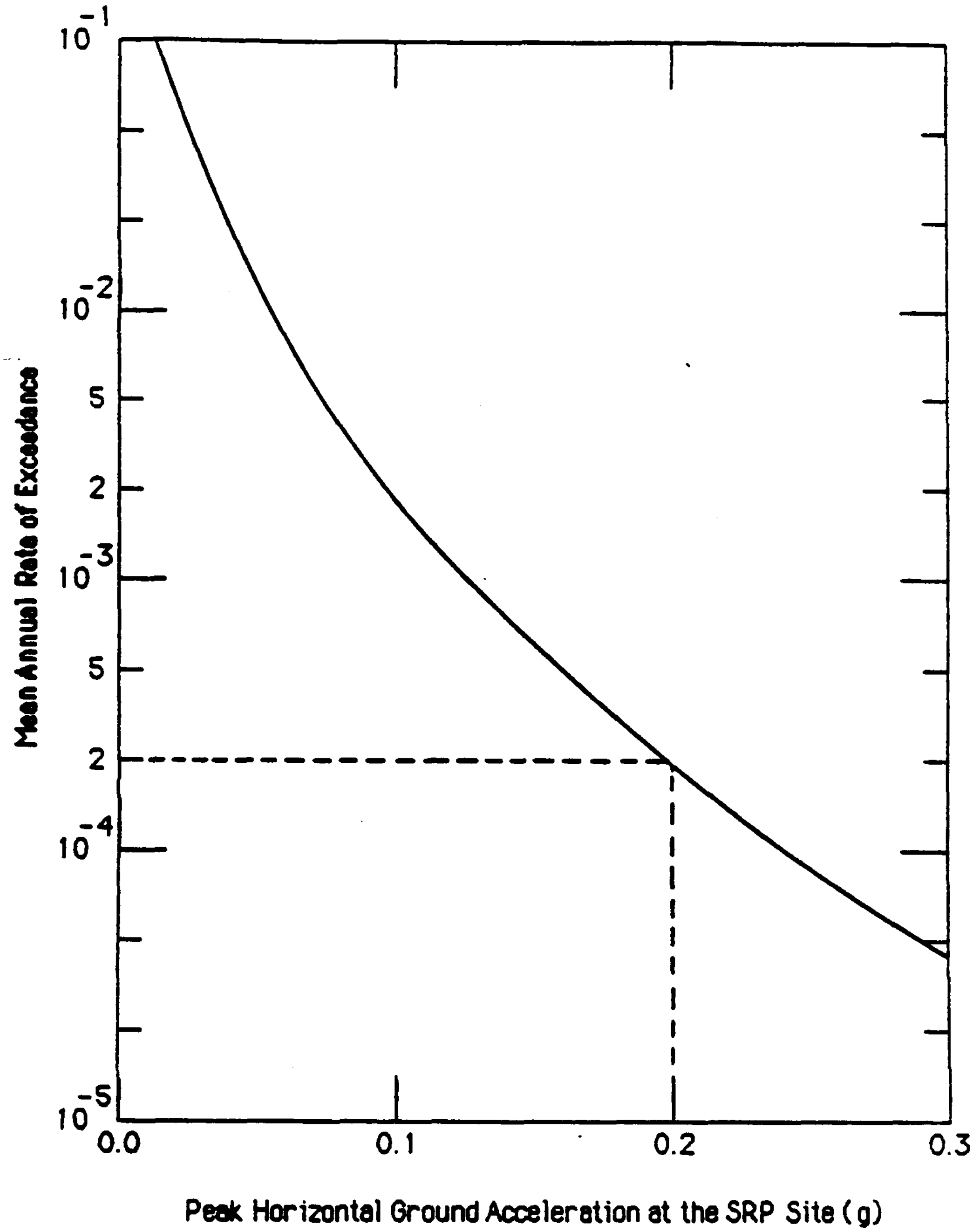

FIGURE 5-3. Earthquake Frequency at the Savannah River Plant

5.3.1.4 Floods

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

5.3.1.5 Other Natural Phenomena Related Events

Extremes in temperature, snow, rain, and lightning may adversely affect operations, but will not generally result in release of radioactive materials. The cumulative frequency from all other natural phenomena related events is 5 x 10^{-4}/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 x 10^{-4}/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 x 10^{-6}/hr.

The expected frequency of sufficient rain falling in a 24-hr period to aggravate a deteriorated situation is estimated to be 8 x 10^{-3}/hr, and to cause internal building damage, 1 x 10^{-4}/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 x 10^{-4}/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 x 10^{-5}/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 3×10^{-9}/(flight)(mile^2) for commercial aviation and about 7×10^{-9}/(flight)(mi^2) for general aviation (15).

Based on a total estimated frequency of 4000 flights over SRP per year and the 4×10^{-3} mi^2 "target" area of the H-Canyon building, the overall aircraft crash frequency is calculated to be 1×10^{-11}/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 3×10^{-12}/hr.

5.3.2.2 Adjacent Explosion

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

5.3.2.3 Adjacent Fire

Of the 143 fires reported in H-Area, 104 have occurred outside the canyon and its ventilation system (Table 5-5). The expected frequency of occurrence of fires in H-Area outside of the canyon building and its ventilation system is 4.4×10^{-4}/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 H-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 39 fires within the canyon and its ventilation system (excluding B-Line) (Table 5-5). Of the 39, none involved process materials. Assuming the frequency is similar to the historical average, the expected frequency for a fire in H-Canyon from all causes is 1.6×10^{-4}/hr. The

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

Cause of Ignition	Total Number for H-Area	Frequency, Occurrences/hr	Number Only for H-Canyon	Frequency, Occurrences/hr
Welding	43	1.8×10^{-4}	15	6×10^{-5}
Electrical short	26	1.1×10^{-4}	8	3×10^{-5}
Miscellaneous or unspecified	26	1.1×10^{-4}	2	8×10^{-6}
Spontaneous combustion	16	7×10^{-5}	8	3×10^{-5}
Process heat	15	6×10^{-5}	1	4×10^{-6}
Smoking materials	5	2×10^{-5}	1	4×10^{-6}
Sparks (nonwelding)	5	2×10^{-5}	2	8×10^{-6}
Explosion	3	1×10^{-5}	1	4×10^{-6}
Friction	2	8×10^{-6}	0	--
Lightning	1	4×10^{-6}	1	4×10^{-6}
Lighting	1	4×10^{-6}	0	--
Total	143	6.0×10^{-4}	39	1.6×10^{-4}

*Fires in facilities adjacent to H-Canyon:

Number = 143 total - 39 (only in H-Canyon)

= 104 fires in adjacent facilities

Frequency = 6.0×10^{-4}/hr - 1.6×10^{-4}/hr (only in H-Canyon)

= 4.4×10^{-4}/hr for fires in adjacent facilities expected frequencies for process related fire initiators are developed in the event trees in Figures B-3 and B-4 (7×10^{-8}/hr for each cycle of solvent extraction and 4×10^{-6}/hr for ion exchange).

5.3.4.2 Uncontrolled Reaction

Uncontrolled reactions occur with an overall frequency of about 2.3×10^{-4}/hr. A typical eructation discharges 1000 lb of materials, but up to 13,000 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 H-Canyon was estimated.

The systems in which uncontrolled reactions have occurred are shown in Table 5-6. Evaporation and dissolving operations account for 69% of the occurrences. Ion exchange, solvent extraction, and waste disposal account for about 10% each.

5.3.4.3 Criticality

There have been no criticality accidents experienced either at H-Canyon or at any other facilities at SRP. Frequencies for criticality in various H-Canyon systems are therefore calculated by fault tree analysis (39,40). 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 <u>Low Energetic Events</u>

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 H-Canyon, the frequency of occurrence of transfer errors is 1.2×10^{-3}/hr. About 8% 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 8% of the transfer errors result in liquid movement outside of the confines of the shielded canyon; such as to Building 211-H (4%), 241-H (3.5%) or HB-line (0.5%). Unexpected transfer to these locations could result in radiation exposure to personnel,

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

System	Number of Occurrences	Frequency, Occurrences/hr
Dissolving	17	9.2×10^{-5}
Evaporation	12	6.5×10^{-5}
Ion exchange	4	2.2×10^{-5}
Solvent extraction	4	2.2×10^{-5}
Waste disposal	4	2.2×10^{-5}
Rerun	1	5×10^{-6}
Head end	0	2×10^{-6} (est)
Tank gallery	0	2×10^{-6} (est)
	42	2.3×10^{-4}

TABLE 5-7. H-Canyon Criticality Potential

Unit Operation	Frequency, Occurrences/hr
Dissolving	1.5×10^{-9}
Head end	2.2×10^{-10}
First Cycle	1.8×10^{-7}
Second U Cycle	1.8×10^{-7}
Head end Evaporator	1.2×10^{-8}
Total	3.7×10^{-7}

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 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 ion exchange (20%) and solvent extraction (10%). Ion exchange is most likely to be affected by the errors (20%), followed by dissolving (15%), and solvent extraction (15%).

5.3.5.2 Overflow

The overall frequency of occurrence of overflows (7.5×10^{-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 3.2×10^{-4}/hr. Ion exchange, evaporation, solvent extraction, and waste disposal are in the range of 6×10^{-5}/hr to 1×10^{-4}/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 1×10^{-3}/hr. As shown in Table 5-13, about 90% of all chemical addition errors are initiated in the cold chemical or water addition facilities. Six percent are from solvent extraction and solvent recovery 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 catalyst and acid concentration errors or by unwanted water additions.

TABLE 5-8. Canyon System in Which Transfer Error was Initiated

System	Number of Occurrences	Frequency, Occurrences/hr
Cold chemicals and water additions	86	4.7×10^{-4}
Ion exchange	46	2.5×10^{-4}
Solvent extraction	25	1.4×10^{-4}
Evaporation	14	8×10^{-5}
Solvent recovery	13	7×10^{-5}
Waste disposal	9	5×10^{-5}
Dissolving	7	4×10^{-5}
Cell flush	7	4×10^{-5}
Head end	5	3×10^{-5}
B-Line	3	2×10^{-5}
Decontamination cells	2	1×10^{-5}
Vessel vent scrubber	1	5×10^{-6}
Not determined	3	2×10^{-5}
	221	1.2×10^{-3}

TABLE 5-9. Canyon System into Which Erroneous Transfer was Made

System	Number of Occurrences	Frequency, Occurrences/hr
Ion exchange	43	2.3×10^{-4}
Dissolving	32	1.7×10^{-4}
Solvent extraction	32	1.7×10^{-4}
Evaporation	19	1.0×10^{-4}
Sumps	18	1.0×10^{-4}
Cold chemicals	13	7×10^{-5}
Solvent recovery	11	6×10^{-5}
Waste disposal	11	6×10^{-5}
Outside facilities	9	5×10^{-5}
Head end	8	4×10^{-5}
241-H	7	4×10^{-5}
Drain headers	5	3×10^{-5}
Rerun	5	3×10^{-5}
Sump collection tanks	4	2×10^{-5}
B-Line	1	5×10^{-6}
Process vessel vent scrubber	1	5×10^{-6}
Cooling water system	1	5×10^{-6}
Not determined	1	5×10^{-6}
	221	1.2×10^{-3}

TABLE 5-10. Canyon Systems Affected by Overflows

System	Number of Occurrences	Frequency, Occurrences/hr
Tank gallery	58	3.2×10^{-4}
Ion exchange	18	1.0×10^{-4}
Evaporation	17	9×10^{-5}
Solvent extraction	15	8×10^{-5}
Waste disposal	11	6×10^{-5}
Dissolving	8	4×10^{-5}
Sumps	6	3×10^{-5}
Head end	2	1×10^{-5}
Sample aisle	2	1×10^{-5}
Rerun	2	1×10^{-5}
	139	7.5×10^{-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	--	8×10^{-4}
	Contaminated solution	--	5×10^{-4}
	Uncontaminated solution	--	3×10^{-4}
Dissolver*	Cooling water	1	5×10^{-6}
	Cold chemicals	15	7×10^{-5}
	Product solution	13	6×10^{-5}
Head end	Cooling water/steam	12	3×10^{-4}
	Process solution	21	5×10^{-4}
	Cold chemicals	2	5×10^{-5}
Solvent extraction	All leaks	--	3×10^{-3}
	Contaminated waste solution transfer	--	2×10^{-3}
	Uncontaminated waste solution transfer	--	1×10^{-3}
Evaporation	All leaks	--	7×10^{-3}
	Contaminated solution	--	4×10^{-3}
	Uncontaminated solution	--	4×10^{-3}
Ion exchange	Cooling water/steam	--	6×10^{-4}
	Process solutions	--	1×10^{-3}
	Cold chemicals	--	2×10^{-4}
Waste disposal	Cooling water/steam	11	3×10^{-4}
	Process solutions	13	7×10^{-5}
	Cold chemicals	3	3×10^{-4}

*Average frequency for each of four dissolvers.

TABLE 5-12. Frequencies of Contaminated Process Solution Leaks

Unit Operation	Leak Frequency, Occurrences/hr
Ancillary Operations and Equipment	
Shops and decontamination facilities	5×10^{-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	6×10^{-5}
Head end	5×10^{-4}
Solvent extraction	3×10^{-3}
Evaporation	7×10^{-3}
Ion exchange	6×10^{-4}
Waste disposal	3×10^{-4}

TABLE 5-13. Canyon System from Which Chemical Addition Error was Initiated

System	Number of Occurrences	Frequency, Occurrences/hr
Cold chemicals and water addition	156	9×10^{-4}
Solvent recovery	5	3×10^{-5}
Solvent extraction	5	3×10^{-5}
Dissolving	2	1×10^{-5}
Evaporation	2	1×10^{-5}
Ion exchange	2	1×10^{-5}
Crane	1	5×10^{-6}
Head end	1	5×10^{-6}
Steam	1	5×10^{-6}
	175	1×10^{-3}

TABLE 5-14. Canyon System into Which Chemical Addition Error was Made

System	Number of Occurrences	Frequency, Occurrences/hr
Solvent extraction	42	2×10^{-4}
Ion exchange	33	2×10^{-4}
Dissolving	29	2×10^{-4}
Cold chemicals	27	1×10^{-4}
Evaporation	12	7×10^{-5}
Head end	11	6×10^{-5}
Waste disposal	7	4×10^{-5}
Solvent recovery	6	3×10^{-5}
Rerun	2	1×10^{-5}
Sump solution hold tank	1	5×10^{-6}
Not determined	5	3×10^{-5}
	175	1×10^{-3}

5.3.5.5 Pluggage

The most common form of pluggage experienced in H-Canyon is instrument line pluggage that occurs at a rate of one to two per day (5×10^{-2}/hr). Sampler pluggage occurs about twice per week (1×10^{-2}/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 1×10^{-5}/hr. The conditions that lead to the reported cases have been corrected; however, it is estimated that it could occur at an expected frequency of 1×10^{-6}/hr for new equipment.

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 solvent extraction operations.

One case in the H-Canyon dissolver system occurred that is considered significant. This occurred during scrap dissolving. Release to the environment of 400 mCi of ^{106}Ru occurred. This single event leads to a frequency of 1×10^{-5}/hr based on operating history.

One case in the H-Canyon head end system has also occurred. This occurred when a strike tank was decontaminated. Release to the environment of 45 mCi of ^{106}Ru occurred. This single event leads to a frequency of 1×10^{-5}/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

TABLE 5-15. Canyon Systems Affected by Coil and Tube Failures

System	Number of Occurrences	Frequency, Occurrences/hr
Evaporators		
LAW	15	
HAW	4	
Product/intercycle	5	
Rerun	7	
Head end	3	
Subtotal	34	1.8×10^{-4}
Dissolvers	12	6.5×10^{-5}
Tanks		
Head end	4	
Product/intercycle	2	
Waste	1	
Solvent recovery	1	
Subtotal	8	4.3×10^{-5}
	54	2.9×10^{-4}

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 Canyon for an overall frequency of 2.9×10^{-4}/hr. The frequency with which activity is detected in the segregated water is 1×10^{-4}/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 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 an waste header failure. On one occurrence, however, waste was released from the canyon because of a transfer error, and sent to the waste farm unknown to 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-H 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 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 <u>Residual Activity Release Events</u>

A residual activity release event is one that allows the residual activity which accumulates in the secondary confinement to reach personnel areas or the outside. These events include suckback and air reversals.

5.3.6.1 Suckback

Eighteen cases of suckback from 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.

5.3.6.2 Air Reversal

Air reversals result usually from multiple causes that vary somewhat according to the affected location. Open doors contributed to half of the recorded occurrences. In three of the ten cases, maintenance operations were being performed on the ventilation or power systems at the time and procedural difficulties were involved. Other contributing causes are local power failures, damper malfunctions, plugged filters, loss of air balance between supply and exhaust systems, and ventilation air leaks past barriers.

The overall frequency of air reversals somewhere in the canyon is 5.4×10^{-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 concern, rather than safety, 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

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	3.2×10^{-4}	0.007	2.2×10^{-6}	Seepage basin
Leak	5.6×10^{-3}	0.003	1.7×10^{-5}	Undesignated
Transfer error	5.4×10^{-4}	0.03	1.6×10^{-5}	211-H
Uncontrolled reaction	3.8×10^{-5}	0.026	1.0×10^{-6}	Undesignated
		Total	3.6×10^{-5}	

*Data from 200-Area Data bank.

**Data from 200-Area Data bank. Some data includes F-Area data also.

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.

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-H (Table 5-16). The overall frequency is 3.6×10^{-5}/hr. Release sites outside the building 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 is 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 $1 \times 10^3 \times$ RCG.

TABLE 5-17. Frequencies of Chemical Releases to Occupied Areas

Type of Incident		Frequency*, Occurrences/hr
Hazardous liquids		
Overflow		3.3×10^{-6}
Leak		2.4×10^{-4}
Transfer error		7.5×10^{-7}
Uncontrolled reaction		1.7×10^{-6}
	Total	2.4×10^{-4}
Hazardous vapors		
Overflow		3.1×10^{-7}
Leak		9.3×10^{-5}
Transfer error		7.8×10^{-8}
Uncontrolled reaction		2.9×10^{-6}
	Total	9.6×10^{-5}

*Frequencies were calculated from data in Table 5-16 and Figure B-8.

5.3.8.2 Cell Covers

The historical frequency of dropping a cell cover has been 2×10^{-5}/hr; however, none have caused significant damage to equipment.

5.3.8.3 Gang Valve Corridors

Frequency data for gang valve failures indicate a mean time between occurrences of 2.5 days (2×10^{-2}/hr). Leaks are reported about once per week (6×10^{-3}/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. Two personnel injuries have occurred in the past five years.

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 two weeks. Two fires have been recorded for the aisle. Fumes have migrated into the sample aisle from the cold chemical storage area. Two injuries have been recorded.

5.3.8.6 Steam Distribution

Complete area-wide steam outages occur at a rate of 2.7×10^{-4}/hr and last an average of about nine hours. About 25% of these outages are due to failures

TABLE 5-18. Cause and Frequencies of Crane Malfunctions

Cause	Hot Crane Frequency	Warm Crane Frequency
Wheel and bearing failure	Infrequent	Infrequent
Major brake repair or adjustment	Every 2 weeks	Once per quarter
Cable replacement	Once per month	Every 2 months
Breakdown of communications	Once per year	Every 3 years
Impact wrenches	Once per week	Once per week
Electrical failure	1-2 per month	1-2 per month
Relamping	Once per quarter	Once per quarter

TABLE 5-19. Causes and Frequencies of Equipment Malfunctions in Sample Aisles

Cause	Frequency
Failure to sample (all causes)	2×10^{-2}/hr
Failure to sample due to plugged sample jet	1×10^{-2}/hr
Hoist failure	1/3 weeks
Elevator failure	~2/year
Radiation and contamination detection instruments	1/3 weeks

in the powerhouse, and 75% 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 1×10^{-3}/hr and is usually due to problems at the powerhouse (9×10^{-4}/hr). The remainder is caused by failures in the 325 psi distribution system (2×10^{-4}/hr).

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 H-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 H-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% (41). 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 lets 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 system.

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 (42).

Complete failure of either of these exhaust fan systems could cause an air reversal. Such complete failure has not occurred to the 221-H exhaust system, although only one of the fans has been operable on several occasions. Also, the central air supply fans have all been out at the same time: in 1975 when a breaker was shorted in Motor Control Center 3, in 1978 and 1979 when the fans were erroneously shut down for maintenance operations, and in 1979 and 1980 during diesel load tests as shown in Appendix B of the H-Canyon Systems Analysis (1).

The dominant contributors to the ventilation system failure are the one component cut sets that 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 (43). 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 1×10^{-7}/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 1×10^{-5}/hr.

Other ventilation system single-component failure rates are shown in Table 5-21 (44).

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 27.

TABLE 5-20. Frequencies of Local Air Reversals

Location	Frequency, Occurrences/hr
Center section	2.2×10^{-5}
Regulated maintenance area	1.1×10^{-5}

TABLE 5-21. Ventilation System Single Blower Failure Rates

Ventilation System	Failure Rate, Occurrences/hr/fan: Air Supply	Failure Rate, Occurrences/hr/fan: Air Exhaust
Canyon air	7.5×10^{-5}	1.2×10^{-4}
Central air	1.9×10^{-4}	5.0×10^{-5}
Process vessel vent	--	2.7×10^{-4}
Recycle vent	--	6.0×10^{-5}

TABLE 5-22. Frequencies for Release of Activity into Plant Streams

Mechanism	Causes	Frequency
Activity in segregated water		1×10^{-4}/hr
	Diversion valve failure	1.5×10^{-4}/demand
	Monitor failure	1×10^{-2}/demand
	Personnel error	3×10^{-2}/demand
Activity in either Upper Three Runs or Four Mile Creek		8×10^{-7}/hr
	Activity in segregated or circulated water systems	1.4×10^{-4}/hr
	Small break in effluent piping	3×10^{-2}/demand
	Sufficient rain to result in runoff	0.2/demand
Major break in effluent piping		3×10^{-4}/demand
Activity reaching cooling tower at same time cooling tower excess is overflowing to Four Mile Creek		1.4×10^{-6}/hr
	Activity in circulated cooling water and failure to divert to retention basin	1×10^{-6}/hr
	Activity from waste tank due to heat exchanger leak	4×10^{-7}/hr
	Leak in isolation gate valve and activity in circulated cooling water	1×10^{-10}/hr
Improper removal of contaminated sludge from delaying basin		1×10^{-6}/hr
Activity sorbs in soil of effluent ditch		2×10^{-6}/hr
Malfunctions within canyon		4×10^{-6}/hr

5.3.9.4 Secondary Confinement for Liquids

The frequency of reported leakage from expansion joints is slightly over once per month.

Liquids in canyon sumps are monitored by sump liquid-level detectors. The mean failure frequency is 3.6×10^{-4}/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 (45). The mean failure rate for each Cash regulator is about one failure every two years (6×10^{-5}/hr).

5.3.9.6 Neutron Monitors

The overall frequency of neutron monitor failures is 4.0×10^{-4}/hr for each monitor unit.

5.4 ACCIDENT CONSEQUENCES

The source term calculations were performed using the methodology documented in Appendix A of the H-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 release 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 H-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 of up to 175 mph.

Tornado. The H-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 canyon building has been analyzed with regard to tornado resistance and the outer walls would remain intact (28) during a DBT. 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 an intensity MM VIII (28,29)

The results of the analyses 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 axial compression; 2) local failure under combined bending and axial tension; 3) formation of 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-H Section 2 (Typical)

- Floor slabs at levels 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 a level 1 - local crushing under combined compression and bending moment with ductility ratio of 1.46 and ratio of axial load to capacity of 0.48.

- Frame between column E and wall F above level 3 - lateral instability due to formation of local failure mechanisms.

Building 221-H Section 6 (Typical)

- Floor slab between column lines D and E, at 2nd floor level - instability in vertical direction due to formation of three hinge failure mechanism.

- Base of column D at 1st floor level - local crushing failure under combined axial compression and bending with ductility ratio of 1.39 and ratio of axial load to capacity of 0.47.

- Base of column D at 2nd floor level - local crushing failure with ductility ratio of 1.03 and ratio of axial load to capacity of 0.40.

- Base of column E at 1st floor level - local crushing failure with ductility ratio of 1.13 and ratio of axial load to capacity of 0.69.

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 H-Canyon Systems Analysis (1), one half of the contents of a system would be released 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 5×10^{-6}% to the atmosphere (8). 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 H-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 4.8 dis/min-ml and to the ground of 1×10^5 dis/min have been measured. Releases to the storm sewer are included in the overall risk due to releases to the seepage and retention basins. Trebler 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 x 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 H-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 is 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 x 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 H-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 H-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 H-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 Process Fire

No fires have occurred involving process materials. Thus, the fires that were analyzed are postulated fires in solvent extraction and in 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 loaded with ^{238}Pu accumulates 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 vessels. The radionuclide releases 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 on 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 (46). 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 (47).

Consequences of uncontrolled reactions are based on a mean value of 2580 lb ejected from the vessel. A postulated explosion in ion exchange is also considered. Sequence frequencies are based on the frequency that 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 (48-50). 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 (50), the cumulative distribution functions for the spectrum of possible conditions is infeasible to determine. A generic estimate 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 (48). 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 Table 5-23. Data for consequence calculations used in this analysis are given in Table 5-24. The event tree for airborne releases 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.

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*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. 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}	0.06	1.00	3×10^{15}	0.06	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
^{83m}Kr	5.1×10^0
^{85m}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
^{135m}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.

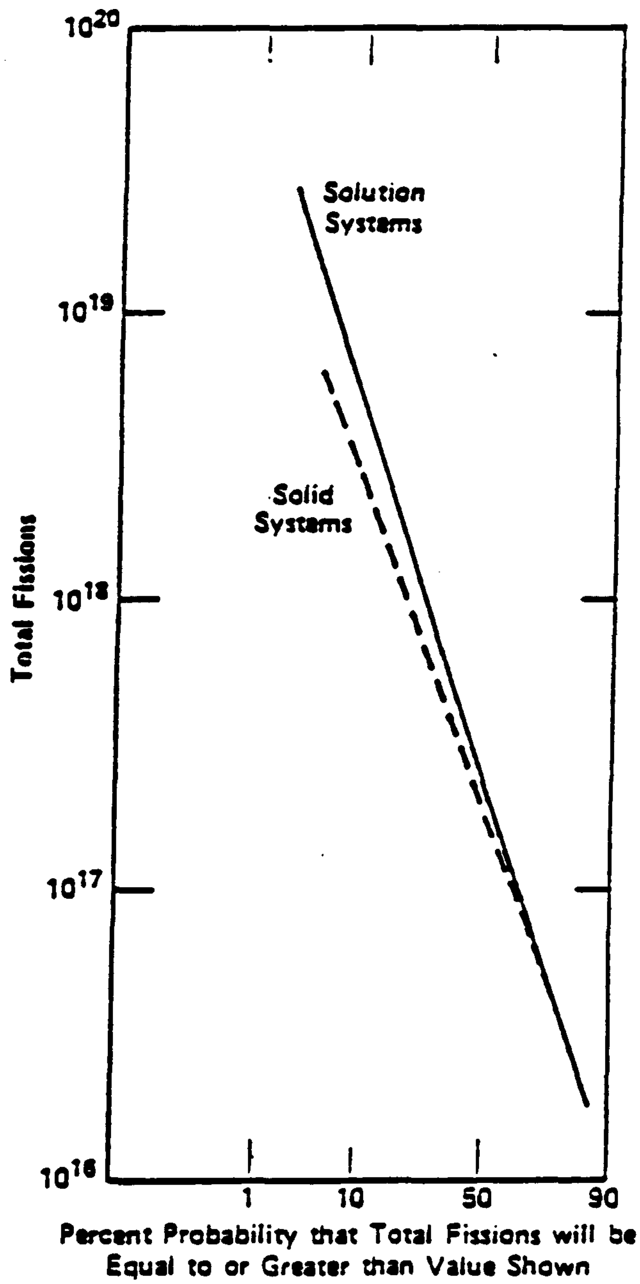

FIGURE 5-4. Cumulative Distribution Functions for Criticality in the Canyons

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-H (Figure B-6 and Table C-6) and for airborne releases from the stack due to transfer errors, overflows, and leaks to the sump (Figure B-7 and Tables C-7 - C-9). The fault tree for releases to Four Mile Creek from coil failures is shown in Figure B-9; the source terms developed in Tables C-10 and C-11. Assumptions and data used in developing the event trees and fault tree are documented in the H-Canyon Systems Analysis (1). In addition, liquid releases to 211-H 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 221 cases examined, 184 unique errors occurred. Seventeen occurred twice, four occurred three times, one occurred four times, and two occurred five times. For the two cases that occurred five times, one involved adding water to the 6.1D dissolver that was intended for the 6.4D dissolver, and the other involved adding water to 6.4D that was intended for 6.1D.

Amounts of material range from a few pounds to more than 400 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 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 an outside facility (211-H) 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 H-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 are analyzed in Reference 51, 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 0.048 Ci/lb of actinides and 45 Ci/lb of fission products. The radionuclides transferred to 211-H are obtained from the isotopic fraction as described in Appendix A of Reference 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 H-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

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

TABLE 5-26. Liquid Releases to 211-H 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	0.048	45	1.4×10^{5}	1.4×10^{-1}
Head end	1×10^{-6}	3170	0.048	45	1.4×10^{5}	1.4×10^{-1}
Solvent Extraction						
First Cycle	1×10^{-6}	3170	0.047	39	1.2×10^{5}	1.2×10^{-1}
Second U Cycle	4×10^{-7}	3170	--	8.3×10^{-4}	2.6×10^{0}	1.1×10^{-6}
Second Np Cycle	1×10^{-6}	3170	--	3.6×10^{-2}	1.1×10^{2}	1.1×10^{-4}
Total	2.4×10^{-6}	--	--	--	--	1.2×10^{-1}
Evaporation						
HAW	1.5×10^{-6}	3170	0.047	39	1.2×10^{5}	1.8×10^{-1}
LAW	1.5×10^{-6}	3170	0.011	1.1×10^{-4}	3.5×10^{1}	5.3×10^{-5}
Ion exchange	5×10^{-6}	280	7.5	7.3	4.1×10^{3}	2.1×10^{-2}
Waste disposal	2×10^{-6}	3170	0.047	39	1.2×10^{5}	2.5×10^{-1}

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

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 ft^3) 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 H-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 5.4.5.9.

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	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 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 400 mCi of ^{106}Ru were released from the dissolver and 45 mCi were released from head end. 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 4000 mrad/400 mrem/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.

About once per year, greater than 300 d/m/ml of beta-gamma or greater than 10 d/m/ml of alpha activity is detected by a monitor, and the water is diverted to a rentention basin with no detectable release to Four Mile Creek. Circulated water is diverted about once every four years (0.25/yr). Segregated water is diverted about 0.8/yr. The median activity (Table 5-28) is about 0.2 Ci per event and the mean value is 17 Ci (26,45,51,52).

In two cases, tank cooling coil leaks resulted in detectable beta-gamma activity in Four Mile Creek. In 1966, 30 mCi were released to the creek from a leak in the 10.2 tank coil. In 1978, 11.5 mCi were released in the creek from a cooling coil leak in the 8.4 waste neutralization tank. In the latter case, about five million gallons of water containing about 40 Ci of fission product activity were diverted to retention basin 281-8H. Ninety percent of this activity was released in the first 180,000 gal (45).

Releases in excess of 40 Ci to a water system are expected to occur at a frequency no more 3.4×10^{-6}/hr (26,52).

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

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

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 3.2×10^{-5} 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 be presented in the revision of Reference 53.

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 Ci/gal, or 21,000 Ci. For a slow release, the mean leak release is assumed to be 7 Ci (52). The release 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 5.4.5.2.

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

Eighteen cases from the hot canyon of gamma intensity at varying distances were reported. The data from both F- and H-Areas were normalized to a

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	Release Rate, Ci/hr Four Mile Creek**
Decontamination	--	1.5×10^{-6}	0.2	3.0×10^{-7}	1.2×10^{-8}
Dissolving	22	2.6×10^{-5}	17.2	4.5×10^{-4}	1.8×10^{-5}
Head End	13	1.6×10^{-5}	17.2	2.7×10^{-4}	1.1×10^{-5}
Solvent Extraction	6	7.2×10^{-6}	19.6	1.4×10^{-4}	5.6×10^{-6}
Evaporation	57	6.8×10^{-5}	19.6	1.3×10^{-3}	5.4×10^{-5}
Waste Disposal	2	2.4×10^{-6}	19.6	4.7×10^{-5}	1.9×10^{-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.

distance of 3 in 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 H-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. No uptakes have occurred as a consequence of an air reversal in H-Canyon. Skin contaminations or contamination of a work area are less severe consequences, usually resulting only in lost work time during cleanup. In the ten recorded cases, nine air reversals resulted in no detectable airborne activity; in one case, the airborne activity was less than the Radiation Control Guide (RCG). Air reversal data are given in Table 5-30.

5.4.7 Chemical Hazards

Leaks are the largest contributer to outside releases of hazardous chemicals (3.6×10^{-5}/hr) and of hazardous liquid (2.4×10^{-4}/hr) or vapor (9.6×10^{-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 2.4×10^{-4}/hr. These incidents may include effects extending from loss of outer clothing only, to burns to 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 will occur is calculated to be 0.067 for leaks and 0.62 for the other three initiating events (overflow, transfer error, and uncontrolled reaction). The frequency for hazardous liquid injury is therefore:

  $$(2.4 \times 10^{-4}/\text{hr})\ (0.067) = 1.6 \times 10^{-5}/\text{hr}$$

  $$(5.8 \times 10^{-6}/\text{hr})\ (0.62) = 33.6 \times 10^{-6}/\text{hr}$$

  $$\text{TOTAL} = 2.0 \times 10^{-5}/\text{hr}$$

  The most likely injury is an acid burn to the skin.

TABLE 5-30. Consequences of Air Reversals

Airborne activity data reported, cases	10
No measureable airborne activity, cases	21
Inadequate data reported, cases	4
Number of air reversals, total cases	35
Mean airborne activity, RCG	31
Median airborne activity, RCG	<0.1
90% bounds, RCG	<0.1-132

RCG = Radiation Control Guide:

1×10^{-9} μCi β-γ/cc air

2×10^{-12} μCi α/cc air

- The frequency for potential exposure to hazardous vapors in an occupied area is 9.6×10^{-5}/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 contains 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 assued to be less than 0.15, and the frequency for hazardous vapor injury in CFP is $<1.4 \times 10^{-5}$/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 one is recorded and this occurred in 1956. This involved sparks from welding operations.

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 up to 1×10^6 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 during 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 100 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 1 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.

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 "fireyes", limit switches, jumpers, yokes, air conditioners, and radiation monitors. Other operating errors that 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. Shielding doors and removable rails, especially the rails, require frequent maintenance.

Other failures include blown fuses, sticking doors, 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 rodded.

About 5% 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/h 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 occurred during the early years of canyon operation.

One incident of fumes from leaks in the cold chemical storage area migrating into the sample aisle is recorded.

Two injuries as a result of nitric acid burns 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-H 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 <u>Engineered Safety Features</u>

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 eructation, 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 that 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 shown in Figure B-9.

Overall, the rate of having activity escape to plant streams through the cooling water system is 3.2×10^{-5} 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 27.

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 unconfirmed except in specific locations known to be especially vulnerable, such as in the [redacted] personnel tunnel. Radiation levels from negligible to 80 R/h at 3 in have been recorded. Alpha contamination to 4×10^6 dis/min has been reported.

No information is currently available regarding activity seepage through the expansion joints into the soil beneath the H-Canyon. Although test drilling around the periphery of the building are planned (54).

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 (26). 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 5.4.4.2).

5.5 RADIOLOGICAL RISKS

The radiological consequences and risk due to accidents for H-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 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 3.8 person-rem/yr for the offsite population and is 0.71 person-rem/yr for the onsite population. The maximum calculated risk to an offsite individual at the site boundary is 4.0×10^{-4} rem/yr. Radiological risks from accidents are summarized in Table 5-36.

5.5.5 Risks to Operating Personnel

Operating personnel in H-Canyon are normally exposed to certain radiological and nonradiological risks both from normal operations and as a result of

TABLE 5-31. Risks Due to Natural Phenomena for H-Canyon Unit Operations*

Unit Operation	Frequency, /yr	Release, Ci	Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
Dissolver	2.0E-04	3.4E-02	6.0E-06	2.2E-06	1.3E-09
Head End	2.0E-04	3.1E-02	5.4E-06	2.0E-06	1.2E-09
First Cycle	2.0E-04	2.6E-02	4.5E-06	1.6E-06	1.0E-09
2nd U Cycle	2.0E-04	5.8E-12	4.4E-15	1.4E-15	1.3E-18
2nd Np Cycle	2.0E-04	4.6E-09	6.1E-12	2.3E-12	1.1E-15
HAW Evaporator	2.0E-04	2.9E-02	5.0E-06	1.8E-06	1.1E-09
LAW Evaporator	2.0E-04	1.4E-09	3.1E-13	1.1E-13	6.5E-17
Ion Exchange	2.0E-04	4.5E-05	9.9E-06	3.8E-06	1.8E-09
High Heat Waste	2.0E-04	2.9E-02	5.0E-06	1.8E-06	1.1E-09
Total for all Unit Operations			3.6E-05	1.3E-05	7.5E-09

*Event leading to significant release is an MM VIII earthquake.

TABLE 5-32. Risks Due to Externally Induced Failures for H-Canyon Unit Operations*

Unit Operation	Frequency, /yr	Release, Ci	Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
Dissolver	8.8E-03	1.7E+00	2.1E-03	3.7E-04	2.8E-07
Head End	8.8E-03	1.6E+00	1.9E-03	3.3E-04	2.5E-07
First Cycle	8.8E-03	9.6E-01	1.6E-03	2.7E-04	2.0E-07
2nd U Cycle	8.8E-03	2.0E-10	1.4E-12	2.2E-13	2.5E-16
2nd Np Cycle	8.8E-03	1.5E-07	2.0E-09	3.7E-10	2.1E-13
HAW Evaporator	8.8E-03	1.1E+00	1.7E-03	3.0E-04	2.3E-07
LAW Evaporator	8.8E-03	4.4E-08	1.0E-10	1.8E-11	1.3E-14
Ion Exchange	8.8E-03	4.9E-04	3.2E-03	6.0E-04	3.3E-07
High Heat Waste	8.8E-03	1.1E+00	1.7E-03	3.0E-04	2.3E-07
Total for all Unit Operations			1.2E-02	2.2E-03	1.5E-06

*Event leading to significant release is impact.

TABLE 5-33. Risks Due to Process Related Occurrences for H-Canyon Unit Operations

Unit Operation	Accident Category	Accident	Frequency, /yr	Release, Ci	Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
Decontamination	Low Energetic (Airborne)	Transfer Error to 211-H	1.8E-05	1.5E-04	2.2E-09	8.1E-10	4.9E-13
		Transfer Error to Sump	3.5E-03	7.0E-07	1.7E-09	2.9E-10	2.1E-13
		Overflow to Sump	4.4E-03	4.2E-07	1.2E-09	2.1E-10	1.5E-13
		Leak to Sump	4.2E+00	1.4E-07	3.9E-07	6.8E-08	4.9E-11
		Total (Airborne)			3.9E-07	6.9E-08	5.0E-11
	Coil Failure (Liquid)		5.3E-04	2.0E-01	1.9E-05		1.6E-09
	Total (Airborne)				3.9E-07	6.9E-08	5.0E-11
Dissolver	Medium Energetic (Airborne)	Uncontrolled Reaction	7.9E-01	1.6E-01	1.3E-02	2.1E-03	2.0E-06
		Criticality	1.3E-05	4.8E+04	1.9E-05	4.3E-05	1.7E-08
		Total (Airborne)			1.3E-02	2.1E-03	2.0E-06
	Low Energetic (Airborne)	Transfer Error to 211-H	7.0E-05	3.1E+00	3.1E-04	1.2E-04	5.6E-08
		Transfer Error to Sump	1.4E-02	1.7E-01	3.3E-04	5.8E-05	4.4E-08
		Overflow to Sump	3.5E-01	9.9E-02	4.9E-03	8.5E-04	6.4E-07
		Leak to Sump	5.4E-01	1.4E-02	1.1E-03	1.8E-04	1.4E-07
		Processing Short-Cooled Fuels	3.5E-02	1.7E+00	3.6E-03	6.1E-04	5.5E-07
		Ruthenium Volatilization	8.8E-02	4.0E-01	1.1E-03	1.5E-04	2.0E-07
		Total (Airborne)			1.1E-02	2.0E-03	1.6E-06
	Coil Failure (Liquid)		9.1E-03	2.5E-01	2.2E-02		1.9E-06
	Total (Airborne)				2.4E-02	4.1E-03	3.7E-06
Head End	Medium Energetic (Airborne)	Uncontrolled Reaction	1.8E-02	1.5E-01	5.6E-04	9.8E-05	7.2E-08
		Criticality	1.1E-04	4.8E+04	1.6E-04	3.5E-04	1.4E-07
		Total (Airborne)			7.2E-04	4.5E-04	2.1E-07
	Low Energetic (Airborne)	Transfer Error to 211-H	1.1E-04	2.2E+01	7.0E-04	2.7E-04	1.4E-07
		Transfer Error to Sump	2.1E-02	1.5E-01	4.5E-04	7.8E-05	5.9E-08
		Overflow to Sump	8.8E-02	8.9E-02	1.1E-03	1.9E-04	1.4E-07
		Leak to Sump	4.4E+00	1.3E-02	7.6E-03	1.3E-03	1.0E-06
		Ruthenium Volatilization	8.8E-02	4.5E-02	1.2E-04	1.7E-05	2.3E-08
		Total (Airborne)			1.0E-02	1.9E-03	1.4E-06
	Coil Failure (Liquid)		5.6E-03	1.7E+01	1.4E-02		1.1E-06
	Total (Airborne)				1.1E-02	2.3E-03	1.6E-06

TABLE 5-33. Risks Due to Process Related Occurrences for H-Canyon Unit Operations (Continued)

Unit Operation	Accident Category	Accident	Frequency, /yr	Release, Ci	Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
First Cycle	Medium Energetic (Airborne)	Fire	6.1E-04	5.8E-01	2.3E-04	4.1E-05	2.9E-08
		Uncontrolled Reaction	8.8E-02	9.8E-02	1.2E-03	1.9E-04	1.8E-07
		Criticality	1.6E-03	4.8E+04	2.3E-03	5.1E-03	2.1E-06
		Total (Airborne)			3.7E-03	5.4E-03	2.3E-06
	Low Energetic (Airborne)	Transfer Error to 211-H	1.1E-04	1.9E+01	5.4E-04	1.9E-04	1.2E-07
		Transfer Error to Sump	2.2E-02	9.3E-02	3.8E-04	6.7E-05	5.0E-08
		Overflow to Sump	3.3E-01	5.5E-02	3.4E-03	5.9E-04	4.4E-07
		Leak to Sump	6.3E+00	7.6E-03	9.1E-03	1.6E-03	1.2E-06
		Total (Airborne)			1.3E-02	2.4E-03	1.8E-06
	Coil Failure (Liquid)		8.4E-04	2.0E+01	2.1E-03		1.8E-07
	Total (Airborne)				1.7E-02	7.8E-03	4.1E-06
2nd U Cycle	Medium Energetic (Airborne)	Fire	6.1E-04	9.4E-06	1.6E-08	2.5E-09	2.9E-12
		Uncontrolled Reaction	8.8E-02	3.9E-07	9.2E-08	1.4E-08	1.6E-11
		Criticality	1.6E-03	4.8E+04	2.3E-03	5.1E-03	2.1E-06
		Total (Airborne)			2.3E-03	5.1E-03	2.1E-06
	Low Energetic (Airborne)	Transfer Error to 211-H	3.7E-05	6.7E-09	3.6E-13	1.2E-13	9.7E-17
		Transfer Error to Sump	7.4E-03	3.3E-11	2.0E-13	3.0E-14	3.4E-17
		Overflow to Sump	3.0E-02	1.9E-11	4.7E-13	7.0E-14	8.1E-17
		Leak to Sump	3.2E+00	2.7E-12	7.0E-12	1.0E-12	1.2E-15
		Total (Airborne)			8.1E-12	1.3E-12	1.4E-15
	Coil Failure (Liquid)		8.4E-04	6.9E-09	7.4E-13		6.3E-17
	Total (Airborne)				2.3E-03	5.1E-03	2.1E-06
2nd Np Cycle	Medium Energetic (Airborne)	Fire	6.1E-04	5.1E-04	1.3E-06	2.4E-07	1.4E-10
		Uncontrolled Reaction	8.8E-02	1.8E-05	7.8E-06	1.4E-06	8.3E-10
		Total (Airborne)			9.1E-06	1.7E-06	9.7E-10
	Low Energetic (Airborne)	Transfer Error to 211-H	9.6E-05	1.2E-05	2.3E-09	9.0E-10	4.4E-13
		Transfer Error to Sump	1.9E-02	5.9E-08	1.7E-09	3.2E-10	1.9E-10
		Overflow to Sump	5.4E-02	3.5E-08	2.9E-09	5.3E-10	3.1E-10
		Leak to Sump	1.1E+01	4.9E-09	7.8E-08	1.4E-08	8.3E-09
		Total (Airborne)			8.5E-08	1.6E-08	8.8E-09
	Coil Failure (Liquid)		8.4E-04	9.3E-02	1.4E-09		1.2E-13
	Total (Airborne)				9.2E-06	1.7E-06	9.8E-09

TABLE 5-33. Risks Due to Process Related Occurrences for H-Canyon Unit Operations (Continued)

Unit Operation	Accident Category	Accident	Frequency, /yr	Release, Ci	Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
HAW Evaporator	Medium Energetic (Airborne)	Uncontrolled Reaction	3.4E-01	1.1E-01	1.0E-02	1.8E-03	1.3E-06
	Low Energetic (Airborne)	Transfer Error to 211-H	1.4E-04	2.1E+01	7.4E-04	2.6E-04	1.7E-07
		Transfer Error to Sump	2.8E-02	1.0E-01	5.5E-04	9.5E-05	7.1E-08
		Overflow to Sump	3.9E-01	6.1E-02	4.5E-03	7.8E-04	5.8E-07
		Leak to Sump	1.6E+01	8.6E-03	2.0E-02	3.7E-03	2.3E-06
		Total (Airborne)			2.6E-02	4.8E-03	3.1E-06
	Coil Failure (Liquid)		1.2E-02	2.0E+01	3.0E-02		2.5E-06
	Total (Airborne)				3.6E-02	6.6E-03	4.4E-06
LAW Evaporator	Medium Energetic (Airborne)	Uncontrolled Reaction	2.3E-01	4.5E-06	8.7E-07	1.5E-07	1.1E-10
	Low Energetic (Airborne)	Transfer Error to 211-H	1.4E-04	8.8E-07	4.4E-11	1.6E-11	9.6E-15
		Transfer Error to Sump	2.8E-02	4.3E-09	3.3E-11	5.8E-12	4.1E-15
		Overflow to Sump	3.9E-01	2.5E-09	2.7E-10	4.8E-11	3.3E-14
		Leak to Sump	1.6E+01	3.6E-10	2.1E-04	2.9E-05	4.0E-08
		Total (Airborne)			2.1E-04	2.9E-05	4.0E-08
	Coil Failure (Liquid)		1.2E-02	8.2E-07	1.3E-09		1.1E-13
	Total (Airborne)				2.1E-04	2.9E-05	4.0E-08
Ion Exchange	Medium Energetic (Airborne)	Fire	3.6E-02	2.9E-02	7.7E-01	1.4E-01	8.0E-05
		Uncontrolled Reaction	7.9E-02	7.5E-03	4.4E-01	8.2E-02	4.6E-05
		Explosion	4.4E-04	2.5E-02	8.2E-03	1.5E-03	8.5E-07
		Total (Airborne)			1.2E+00	2.3E-01	1.3E-04
	Low Energetic (Airborne)	Transfer Error to 211-H	4.0E-04	5.6E-02	2.2E-02	8.6E-03	4.0E-06
		Transfer Error to Sump	8.1E-02	2.8E-04	1.6E-02	3.1E-03	1.7E-06
		Overflow to Sump	8.8E-01	5.4E-04	3.5E-01	6.6E-02	3.7E-05
		Leak to Sump	1.1E+01	2.6E-04	2.0E+00	3.8E-01	2.1E-04
		Total (Airborne)			2.4E+00	4.6E-01	2.5E-04
	Total (Airborne)				3.6E+00	6.8E-01	3.8E-04

TABLE 5-33. Risks Due to Process Related Occurrences for H-Canyon Unit Operations (Continued)

Unit Operation	Accident Category	Accident	Frequency, /yr	Release, Ci	Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
High Heat Waste	Medium Energetic (Airborne)	Uncontrolled Reaction	1.8E-01	1.1E-01	5.3E-03	9.3E-04	6.8E-07
	Low Energetic (Airborne)	Transfer Error to 211-H	1.8E-04	2.1E+01	9.2E-04	3.3E-04	2.1E-07
		Transfer Error to Sump	3.5E-02	1.0E-01	6.8E-04	1.2E-04	8.9E-08
		Overflow to Sump	5.3E-01	6.1E-02	6.0E-03	1.0E-03	7.8E-07
		Leak to Sump	6.1E-01	8.6E-03	7.8E-04	1.4E-04	8.8E-08
		Total (Airborne)			8.4E-03	1.6E-03	1.2E-06
	Coil Failure (Liquid)		8.4E-04	2.8E-01	2.1E-03		1.8E-07
	Total (Airborne)				1.4E-02	2.6E-03	1.8E-06
Total for all Unit Operations	Medium Energetic (Airborne)	Fire			7.7E-01	1.4E-01	8.0E-05
		Uncontrolled Reaction			4.7E-01	8.7E-02	5.0E-05
		Explosion			8.2E-03	1.5E-03	8.5E-07
		Criticality			4.8E-03	1.1E-02	4.3E-06
		Total (Airborne)			1.3E+00	2.4E-01	1.3E-04
	Low Energetic (Airborne)	Transfer Error to 211-H			2.5E-02	9.8E-03	4.7E-06
		Transfer Error to Sump			1.9E-02	3.5E-03	2.0E-06
		Overflow to Sump			3.7E-01	6.9E-02	3.9E-05
		Leak to Sump			2.1E+00	3.9E-01	2.1E-04
		Processing Short-Cooled Fuels			3.6E-03	6.1E-04	5.5E-07
		Ruthenium Volatilization			1.2E-03	1.7E-04	2.3E-07
		Total (Airborne)			2.5E+00	4.7E-01	2.6E-04
	Coil Faillure (Liquid)				7.0E-02		5.9E-06
	Total (Airborne)				3.7E+00	7.1E-01	4.0E-04

FIGURE 5-34. Summary of Risks for H-Canyon Accidents for All Unit Operations

Accident		Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
Natural Phenomena		3.6E-05	1.3E-05	7.5E-09
Externally Induced Failures		1.2E-02	2.2E-03	1.5E-06
Process Related Occurrences				
Medium Energetic (Airborne)	Fire	7.7E-01	1.4E-01	8.0E-05
	Uncontrolled Reaction	4.7E-01	8.7E-02	5.0E-05
	Explosion	8.2E-03	1.5E-03	8.5E-07
	Criticality	4.8E-03	1.1E-02	4.3E-06
	Total (Airborne)	1.3E+00	2.4E-01	1.3E-04
Low Energetic (Airborne)	Transfer Error to 211-H	2.6E-02	9.8E-03	4.7E-06
	Transfer Error to Sump	1.9E-02	3.5E-03	2.0E-06
	Overflow to Sump	3.7E-01	6.9E-02	3.9E-05
	Leak to Sump	2.1E+00	3.8E-01	2.1E-04
	Processing Short-Cooled Fuels	3.6E-03	6.1E-04	5.5E-07
	Ruthenium Volatilization	1.2E-03	1.7E-04	2.3E-07
	Total (Airborne)	2.5E+00	4.6E-01	2.6E-04
Coil Failure (Liquid)		7.0E-02		5.9E-06
Total (Airborne)		3.7E+00	7.1E-01	4.0E-04
Total (Process Occurrences)		3.8E+00	7.1E-01	4.0E-04
Total for all accidents	Airborne	3.7E+00	7.1E-01	4.0E-04
	Liquid	7.0E-02		5.9E-06
	Total	3.8E+00	7.1E-01	4.0E-04

FIGURE 5-35. Summary of Risks for H-Canyon Unit Operations for All Accidents

Unit Operation	Accident Category	Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
Decontamination	Process Occurrences (Airborne)	3.9E-07	6.9E-08	5.0E-11
	Process Occurrences (Coil Failure)	1.9E-05		1.6E-09
	Total (Airborne)	3.9E-07	6.9E-08	5.0E-11
	Total (Liquid)	1.9E-05		1.6E-09
	Total	1.9E-05	6.9E-08	1.6E-09
Dissolver	Natural Phenomena	6.0E-06	2.2E-06	1.3E-09
	Externally Induced Failures	2.1E-03	3.7E-04	2.8E-07
	Process Occurrences (Airborne)	2.5E-02	4.1E-03	3.7E-06
	Process Occurrences (Coil Failure)	2.2E-02		1.9E-06
	Total (Airborne)	2.7E-02	4.5E-03	4.0E-06
	Total (Liquid)	2.2E-02		1.9E-06
	Total	4.9E-02	4.5E-03	5.8E-06
Head End	Natural Phenomena	5.4E-06	2.0E-06	1.2E-09
	Externally Induced Failures	1.9E-03	3.4E-04	2.5E-07
	Process Occurrences (Airborne)	1.1E-02	2.3E-03	1.6E-06
	Process Occurrences (Coil Failure)	1.4E-02		1.2E-06
	Total (Airborne)	1.3E-02	2.6E-03	1.8E-06
	Total (Liquid)	1.4E-02		1.2E-06
	Total	2.6E-02	2.6E-03	3.0E-06
First Cycle	Natural Phenomena	4.5E-06	1.7E-06	1.0E-09
	Externally Induced Failures	1.6E-03	2.7E-04	2.0E-07
	Process Occurrences (Airborne)	1.7E-02	7.8E-03	4.1E-06
	Process Occurrences (Coil Failure)	2.1E-03		1.8E-07
	Total (Airborne)	1.9E-02	8.1E-03	4.3E-06
	Total (Liquid)	2.1E-03		1.8E-07
	Total	2.1E-02	8.1E-03	4.5E-06
2nd U Cycle	Natural Phenomena	4.4E-15	1.4E-15	1.3E-18
	Externally Induced Failures	1.5E-12	2.2E-13	2.5E-16
	Process Occurrences (Airborne)	2.3E-03	5.2E-03	2.1E-06
	Process Occurrences (Coil Failure)	7.4E-13		6.3E-17
	Total (Airborne)	2.3E-03	5.2E-03	2.1E-06
	Total (Liquid)	7.4E-13		6.3E-17
	Total	2.3E-03	5.2E-03	2.1E-06

FIGURE 5-35. Summary of Risks for H-Canyon Unit Operations for All Accidents (Continued)

Unit Operation	Accident Category	Risk: Offsite Population, person-rem/yr	Risk: Onsite Population, person-rem/yr	Risk: Offsite Maximum Individual, rem/yr
2nd Np Cycle	Natural Phenomena	6.1E-12	2.3E-12	1.1E-15
	Externally Induced Failures	2.0E-09	3.7E-10	2.1E-13
	Process Occurrences (Airborne)	9.1E-06	1.6E-06	9.7E-10
	Process Occurrences (Coil Failure)	1.4E-09		1.2E-13
	Total (Airborne)	9.1E-06	1.6E-06	9.7E-10
	Total (Liquid)	1.4E-09		1.2E-13
	Total	9.1E-06	1.6E-06	9.7E-10
HAW Evaporator	Natural Phenomena	5.0E-06	1.8E-06	1.1E-09
	Externally Induced Failures	1.7E-03	3.0E-04	2.3E-07
	Process Occurrences (Airborne)	3.6E-02	6.6E-03	4.4E-06
	Process Occurrences (Coil Failure)	3.0E-02		2.6E-06
	Total (Airborne)	3.8E-02	6.9E-03	4.6E-06
	Total (Liquid)	3.0E-02		2.6E-06
	Total	6.8E-02	6.9E-03	7.2E-06
LAW Evaporator	Natural Phenomena	3.1E-13	1.1E-13	6.5E-17
	Externally Induced Failures	1.0E-10	1.8E-11	1.3E-14
	Process Occurrences (Airborne)	2.1E-04	3.0E-05	4.0E-08
	Process Occurrences (Coil Failure)	1.3E-09		1.1E-13
	Total (Airborne)	2.1E-04	3.0E-05	4.0E-08
	Total (Liquid)	1.3E-09		1.1E-13
	Total	2.1E-04	3.0E-05	4.0E-08
Ion Exchange	Natural Phenomena	9.9E-06	3.8E-06	1.8E-09
	Externally Induced Failures	3.2E-03	6.0E-04	3.3E-07
	Process Occurrences (Airborne)	3.6E+00	6.8E-01	3.8E-04
	Total (Airborne)	3.6E+00	6.8E-01	3.8E-04
	Total	3.6E+00	6.8E-01	3.8E-04
High Heat Waste	Natural Phenomena	5.0E-06	1.8E-06	1.1E-09
	Externally Induced Failures	1.7E-03	3.0E-04	2.3E-07
	Process Occurrences (Airborne)	1.4E-02	2.6E-03	1.9E-06
	Process Occurrences (Coil Failure)	2.2E-03		1.8E-07
	Total (Airborne)	1.5E-02	2.9E-03	2.1E-06
	Total (Liquid)	2.2E-03		1.8E-07
	Total	1.8E-02	2.9E-03	2.3E-06
Total for all Unit Operations	Airborne	3.7E+00	7.1E-01	4.0E-04
	Liquid	7.0E-02		5.9E-06
	Total	3.8E+00	7.1E-01	4.0E-04

TABLE 5-36. Summary of Risk for Accidents

Pathways	Risks Due to Accidents: Natural Phenomena	Externally Induced Failures	Process Operations	Total
Airborne Release to Offsite Population, Person-rem/yr	3.6×10^{-5}	1.2×10^{-2}	3.7×10^{0}	3.7×10^{0}
Airborne Release to Onsite Population Person-rem/yr	1.3×10^{-5}	2.2×10^{-3}	7.1×10^{-1}	7.1×10^{-1}
Airborne Release to Offsite Maximum Individual, rem/yr	7.5×10^{-9}	1.5×10^{-6}	4.0×10^{-4}	4.0×10^{-4}
Liquid Releases to Offsite Population, Person-rem/yr*	--	--	7.0×10^{-2}	7.0×10^{-2}
Liquid Releases to Offsite Maximum Individual, rem/yr*	--	--	5.9×10^{-6}	5.9×10^{-6}

*Direct release to Four Mile Creek.

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 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 mrem/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 contamination 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, or 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 [redacted] personnel tunnel. Instensities 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-H 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 taken.

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 elevation of the H-Canyon facility 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 the rain, 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 H-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 H-Canyon is limited by rigid procedural control of ignition sources present in the systems and control of the inventory of combustibles. Similiarly, 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 control the rate of reagent additions to a reactive system where an accumulation of excess reagent may occur.

The confinement system of the H-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 H-Canyon processing system (excluding the HB-Line) is reduced through a system of rigid administrative and procedural controls that act to limit and monitor the quantities and the concentrations of fissile materials in process solutions to safe levels. Similiarly, procedures developed for nuclear safety control ensure that process conditions are monitored and maintained that prevent precipitation 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 Release Events

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

Because 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 ventilation system air flow rates also prevents the accumulation of hazardous chemical fume 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 review 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, that 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 that functions 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 H-Canyon, 221-H Building 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 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 for these systems are provided 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. Quality Assurance Assessment Reports which limit the potential for exceeding a QA criteria 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 Fuel Receipt and Storage

Uranium and uranium oxide fuels containing ^{235}U at enrichments from 1.1% to 93.5% are routinely received at Building 221-H in transfer casks mounted on special railroad cars. The cars are pushed into a tunnel at the south end of the Building 221-H hot canyon by a battery-powered, electric locomotive. The cover is removed from the top of the cask by the hot canyon crane. The irradiated material, normally bundled tubes, may be charged directly to a dissolver from the cask car, or may be stored under water in the fuel storage cell. Offsite fuels, except High Flux Isotope Reactor (HFIR) and a Reacteur a Haut Flux (RHF) fuels, are not stored, but are charged directly to the dissolvers when removed from cask cars.

If storage is required, the bundles are stored in racks under water: SRP fuels in bundle storage, neptunium targets in neptunium storage, and HFIR and RHF fuels in HFIR storage.

Intermediate storage of materials is restricted to storage racks and the shipping casks because only these pieces of equipment are designed to maintain nuclear safety by providing adequate spacing, and to provide adequate shielding and decay heat removal.

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	Fuel Receipt and Storage Dissolving Head End Solvent Extraction Frames Evaporation Rerun Waste Disposal 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
$AgNO_3$ Reactor Temperature Instruments	18 months
Colorimeters	6 months
Conductivity Meters	6 months

The following requirements and limitations from Technical Standards administratively control the unloading, charging, and storage of fuel elements to prevent fuel from being assembled in a supercritical array:

1. Materials that are readily identifiable may be handled according to rules specific to their own characteristics. Materials that look alike are handled according to the rules for the most reactive material having similar external appearance.

2. Materials received in Building 221-H are packaged in bundles or other suitable containers that can be handled remotely with the canyon crane. Quantities in a container shall be less than the minimum number that can be made critical if arranged in the most reactive configuration and submerged in water.

3. No more than one container of material shall be outside of the shipping cask, dissolvers, or the approved intermediate storage places at any given time.

4. Material awaiting processing is stored either in the shipping cask or in an approved storage location.

5. Individual tubes, if separated from their fuel assemblies, are considered as complete assemblies for the purpose of determining the permissible number of assemblies in a bundle.

To prevent a criticality accident resulting from overheating and melting fuel tubes, SRP assemblies are cooled the minimum time necessary to ensure that the inner tube does not exceed 385°C in air prior to receipt in 221-H Canyon. This is verified by reference to the Reactor Department batching memorandum for the specific fuel being received.

The pre-irradiation value for ^{235}U content shall be used in all nuclear safety calculations because unirradiated fuel cannot be distinguished from irradiated fuel.

6.1.2 Dissolving

Two chemical dissolvers, 6.1D and 6.4D, and one electrolytic dissolver, 6.3D, are used in Building 221-H to dissolve uranium and uranium oxide fuels containing ^{235}U at enrichments from 1.1% to 93.5%. Reactor fuels containing components that cannot be dissolved chemically in nitric acid are dissolved electrolytically. Materials dissolved include irradiated or nonirradiated enriched uranium fuel elements (including uranium oxide elements) or uranium scrap, all of which are called fuels.

A number of controls are necessary for the safe handling of ^{235}U. These controls involve the mass of uranium charged to the dissolver, the volume of dissolvent, and the heel of undissolved fuel. The principal mechanical control is the dissolver insert. The insert is necessary to maintain fuel tubes in a row so that any fuel tube fragments created during dissolution are confined in a slot configuration and can be detected.

The frame dissolver, 5.4D, used for dissolving neptunium target assemblies, is discussed under the Frame and Frame Waste Recovery Section 6.1.7.

A maximum of 7% hydrogen on an air-free basis is generated during the first hour of chemical dissolution, decreasing to an average of about 3%. The lower flammability limit for hydrogen in air is 4%. An air purge reduces the concentration of hydrogen in the off-gas to well below the flammability limit.

Hydrogen-air mixtures are flammable in the range of 4.1 to 75 volume percent in dry air and will explode at room temperature if any external source of ignition occurs such as sparks from electrical equipment, overheated bearings, or any open flame.

Hydrogen evolution at concentrations in the flammable range could be ignited in the dissolver or in the silver reactor. The latter incident could release a large fraction of the ^{131}I and ^{129}I absorbed as AgI and $AgIO_3$. Maintaining sufficient acid in the dissolver and a minimum air purge rate prevents hydrogen from reaching concentrations sufficient to explode.

Nitric acid concentration is checked by the control room operator by noting the solution specific gravity before the bundles are charged to ensure that sufficient acid is present before dissolution is initiated.

The concentration of ^{235}U is maintained within maximum safe limits specified in the Nuclear Safety Technical Standard (1):

- Maximum concentration in aqueous solution: 11.5 grams/liter.

This is accomplished through administrative controls and process instrumentation so that only through multiple failures or errors can an unplanned criticality occur.

1. The necessary nuclear safety data for each dissolver batch is conveyed to the operating group by means of a technical data sheet prepared by Separations Technology and approved by area supervisors of the Separations and Separations Technology Departments.

2. The following specific provisions apply to the operation of the dissolver.

   (a) Nuclear safety control steps are highlighted where they appear in dissolver operating procedures.

   (b) Each fuel component or bundle charged to a dissolver is identified by means of a unique number readable from the hot canyon crane or, if radiation levels permit, by personnel handling the fuel before its entry into the canyon. If positive visual identification cannot be made, the fuel remains outside the dissolver until positive identification can be made by reference to fuel loading or shipping data.

(c) The total quantity of ^{235}U permitted in a dissolver at any one time is limited to prevent exceeding the maximum safe solution concentration (11.5g $^{235}U/l$), in the event that the dissolver solution is concentrated by evaporation to the minimum volume, which is at the level of the bottom of the top (steam) coil.

(d) Fuels charged to a dissolver are packaged so as to be dimensionally compatible with the column insert compartments, or wells, and to ensure against inadvertent double-charging of a compartment.

(e) Insert compartments left vacant are blocked to prevent unintentional charging of fuel when such action could cause the ^{235}U concentration limit to be exceeded.

(f) The fragment height in each compartment is measured with a probe upon completion of each dissolution to preclude the possibility of charging fuel into an insert compartment that already contains an excessive quantity of fuel fragments.

3. The following provisions prevent the inadvertent addition of a precipitant to a vessel which contains fissile material.

(a) All installations of in-canyon piping are reviewed and authorized by approved procedures.

(b) Any addition of a chemical to the dissolving system by other than permanently installed piping (e.g., the pump cart) is reviewed and authorized by an approved procedure.

(c) Separations Technology reviews indirect work requests and procedures, and verifies before approving them that the work will not allow a precipitant to be added to a vessel that may contain fissile material.

4. Inadvertent concentration of fissile material by evaporation is prevented or detected and corrected.

(a) Prior to the start of each campaign, it is confirmed by visual inspection that nuclear safety blanks are installed in the steam lines to the lower coils in dissolvers 6.1D and 6.4D, and in the steam line to dissolver 6.3D.

(b) All installations and removal of nuclear safety blanks will be according to approved procedures.

(c) If any dissolver (6.1D, 6.3D, or 6.4D) or head-end tank contain fissile material during a period of extended process outage, when slow evaporation may not be apparent, a special logsheet is used for surveillance. The logsheet specifies the action to be taken if a specified minimum level is reached. The weight factor and specific gravity instrumentation is scheduled for calibration and testing at the beginning of the outage and at 6-mo intervals until the end of the outage (1).

5. Dissolver 6.4D normally contains a Mark XII column insert. Mark XII-A spacers are installed before charging any fuel if the dissolver nuclear safety study shows the spacers are needed to provide maximum separation between fuel bundles in adjacent quadrants. Spacers, when required, will be specified.

6. The following provisions apply to the operation of electrolytic dissolver 6.3D:

   (a) Before each electrolytic dissolver campaign, a Test Authorization (TA) specifying the necessary conditions and limitations for dissolving the proposed fuels shall be written and approved.

   (b) Dissolver 6.3D has the required concentration of boric acid in each dissolution as a secondary nuclear safety control.

   (c) The boron detector instrumentation and interlock are calibrated and tested before each electrolytic dissolver campaign and at 6-month intervals during operation (1).

   (d) Before the start of each electrolytic dissolver campaign, it is confirmed by visual inspection that the proper nuclear safety blanks are installed to prevent dilution or diversion of boric acid poison solution during fuel dissolution in 6.3D.

6.1.3 Head End

The head end process decontaminates and clarifies the raw metal solution for subsequent solvent extraction processing. The head end cycle may include a concentration step, a "strike" or precipitation step, centrifugation to remove the precipitate, cake washing, and cake disposal (See Section 3.2.2.5).

A nuclear criticality caused by exceeding the critical concentration of ^{235}U in the head end evaporator or strike tank or by precipitation of excess uranium in a centrifuge cake is excluded by the following administrative controls:

1. The concentration of fissile material is maintained at less than the critical concentration through administrative controls and process instrumentation, so that only through multiple failures or errors can an unplanned criticality occur. This is

accomplished through appropriate nuclear safety control steps in head end procedures.

2. The necessary nuclear safety data for each fuel is conveyed to the operating group by means of Technical Data Sheet prepared by Separations Technology and approved by area supervisors of both Separations and Separations Technology Departments.

3. Inadvertent addition of a precipitant to a vessel that contains fissile material is prevented by the following controls:

   (a) Nuclear safety blanks are installed at specific locations to prevent addition of a precipitant to any head end vessel that might contain fissile material. These blanks are installed and inspected to verify that all nuclear safety blanks are in place prior to starting a campaign and routine intervals thereafter.

   (b) All installations of in-canyon piping and any addition of a chemical to the head end system by other than permanently installed piping is reviewed and authorized by approved procedures.

6.1.4 First Solvent Extraction Cycle

First solvent extraction cycle in Building 221-H has two functions: 1) to provide decontamination of the Np and U from fission products and chemical impurities, and 2) to separate Np and U into individual streams for further processing. First cycle consists of the 1A, 1B, and 1C mixer settler banks, decanters and process tanks.

A nuclear criticality can occur if the critical concentration of ^{235}U in a vessel or mixer settler is exceeded in this cycle. Concentration and inventories of fissile material 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 the procedural control system and all the limits specified in 221-H Building Technical Standards. The limit for maximum ^{235}U concentrations is 10.8 g/l (1). Nuclear safety controls shall be highlighted where they appear.

2. All installation of new or revised cold chemical piping is reviewed and authorized by an approved indirect repair order.

3. All installations of in-canyon piping or any addition of a chemical to a first cycle system other than permanently installed piping is reviewed and authorized. Separations Technology reviews indirect repair orders and procedures, and verifies before approving them that the work will not allow a precipitant to be added to a vessel that may contain fissile material.

4. Prior to the start of each campaign, it is confirmed that a nuclear safety blank (NSB) is installed according to requirements in the steam line to 1AF adjustment tank coil.

5. Removal of the NSB from the steam line to a tank coil requires an indirect repair order approved by the superintendents of the Separations and Separations Technology Departments.

6. If the NSB has been removed from the steam line to 1AF adjustment tank coil, and 1AF adjustment tank will contain fissile material, the tank wt ftr and sp gr instrumentation is placed on the nuclear safety - priority work - preventive maintenance schedule for calibration and testing at routine intervals.

7. Surveillance of 1AF adjustment tank wt ftr and temperature is conducted during periods of normal process outages. Any decrease in liquid level that may have been caused by evaporation is investigated and corrected.

8. Special logsheets are used for surveillance for 1AF adjustment and 1AF feed when they contain fissile material during period of extended process outage. Slow evaporation may occur that requires corrective action. The logsheet specifies the action to be taken if a specified minimum level is reached. The tank(s) wt ftr and sp gr instrumentation is placed on the nuclear safety priority work and preventive maintenance.

9. The flow measuring instrumentation is included on the nuclear safety priority work preventative maintenance schedule for calibration and testing at routine intervals.

   In addition, the alarms are also be included on the same nuclear safety priority work maintenance schedule.

10. The following provisions apply to the operation of mixer-settler 1A:

    (a) A high flow alarm is installed in the 1AF flow measuring instrument.

    (b) The 1AF high flow alarm is set to alarm when the 1AF flow increases above a preset percentage of the specified flow rate.

    (c) The cycle does not operate unless the 1AF flow instrument is functioning.

    (d) The cycle must be shut down if the undesired increase in 1AF flow is not corrected within a specified procedural time limit.

(e) 1AX low flow alarms are set to alarm when flow decreases below a preset percentage of the specified flow rate.

(f) At least one of the two 1AX flow measuring instruments must be functioning while first cycle is operating.

(g) The cycle must be shut down if the undesired reduction of 1AX flow is not corrected within a specified procedural time limit.

(h) The 1AS heat exchanger is pressure tested routinely and verified to be free of internal leakage that could dilute the 1AS-acid with steam. The completed pressure test procedure is filed in the facility QA file.

(i) Prior to the start of each campaign, it is confirmed that NSBs are installed at the proper locations to preclude inadvertant additions of process water and solutions to the 1A mixer-settler.

(j) Dilutent is not added to a first cycle solvent unless authorized by a PCN (process change notice), which is prepared by Separations Technology and approved by the Separations and Separations Technology area supervisors.

(k) The cycle is not operated when the TBP analysis of the solvent is less than the minimum operating limit.

11. If the neutron count exceeds the limit established for any mixer settler, corrective actions must return the neutron count to normal or the cycle is shut down.

12. The following provisions apply to the operation of mixer-settler 1B:

(a) The specific gravity instrumentation and low specific gravity alarms for 1BX head tanks are included on the nuclear safety priority work preventative maintenance schedule for calibration and testing at routine intervals.

(b) Prior to the start of each campaign, it is confirmed that NSBs are installed at the proper locations to preclude inadvertant additions of process water and solutions to the 1B mixer-settler.

(c) First cycle does not operate if the NSBs are removed.

(d) 1BX solutions are sampled and the acidity determined to be within specifications before any of the solution is transferred to head tanks.

(e) The cycle is shut down if the acidity instrumentation in 1B head tanks indicates the acidity of the 1BX is less than the minimum limit.

(f) The 1BX heat exchanger is pressure tested routinely and verified to be free of an internal leak that could dilute the 1BX with steam. The completed pressure test procedure shall be filed in the facility QA file.

(g) High flow alarms are installed on the two 1BX flow measuring instruments.

(h) 1BX high flow alarms are set to alarm when the 1BX flow increases to a preset level above of the specified flow rate.

(i) At least one of the two 1BX flow measuring instruments must be functioning while the first cycle is operating.

(j) The cycle must be shut down if an increase in 1BX flow of more than the preset alarm level is not corrected within the set procedural time limit.

(k) A high temperature alarm is installed for the 1BX stream.

(l) The temperature measuring instrumentation and high temperature alarms for specified streams and vessels are included on the nuclear safety priority work preventive maintenance schedule for routine calibration and testing at 6-mo intervals.

(m) The cycle must be shut down if the 1BX, 1BS, or 1AX temperature exceeds the operating limit by more than 5°C and is not corrected within the specified procedural time limit (1).

13. The following provisions apply to the operation of mixer-settler 1C:

(a) The 1CX and 1CU low flow alarms are set to alarm at the low flow rate specified in their respective operational procedures.

(b) The cycle does not operate unless at least one flow instrument (1CX or 1CU) is functioning.

(c) The cycle must be shut down if a decrease in 1CX or 1CU flow is more than specified and is not corrected within the specified procedural time limit.

(d) The 1CU colorimeter and 1CU conductivity instrumentation is included on the nuclear safety priority work preventive maintenance schedule for calibration and testing at 6-mo intervals.

(e) If the 1CU colorimeter or 1CU conductivity meter reading exceeds the limit established for normal cycle operation, action is taken so that the colorimeter or conductivity reading is either returned to normal, or the cycle is shut down.

6.1.5 Second Uranium Solvent Extraction Cycle

The second uranium cycle of solvent extraction provides additional decontamination for the enriched uranium that is recovered from spent reactor fuel by the first cycle. Second cycle consists of the 1D and 1E mixer settlers banks, decanters, and process tanks.

A nuclear criticality can occur in this cycle if the critical concentration of ^{235}U in a vessel or mixer settler is exceeded in this cycle. Concentration and inventories of fissile material below the critical limits are maintained through administrative control and process control instrumentation as described below.

1. Procedures for operating the second uranium solvent extraction cycle comply with procedural control system and all the limits specified in 221-H Building Technical Standards. The limits for maximum ^{235}U concentration are 11.5 g/l (1).

2. All installation of new or revised cold chemical piping is authorized by an approved indirect repair order.

3. All installations of in-canyon piping or any addition of a chemical to a second uranium cycle system other than permanently installed piping must be authorized. Separations Technology reviews indirect repair orders and procedures, and verifies before approving them that the work will not allow a precipitant to be added to a vessel that may contain fissile material.

4. Prior to the start of each campaign, it is confirmed that a NSB is installed according to requirements in the steam line to 1DF adjustment tank coil.

5. Removal of the NSB from the steam line to 1DF adjustment tank coil requires an indirect repair order approved by the superintendents of the Separations and Separations Technology Departments.

6. If the NSB has been removed from the steam line to 1DF adjustment tank coil, and 1DF adjustment tank will contain fissile material, the tank wt ftr and sp gr instrumentation are placed on the nuclear safety priority work preventive maintenance schedule for calibration and testing at routine intervals.

7. Surveillance of 1DF adjustment and 1DF tank(s) and wt ftr and temperature is conducted during periods of normal process outages. Any decrease in liquid level which may have been caused by evaporation shall be investigated and corrected.

8. If 1DF tank will contain fissile material during a period of extended process outage, when slow evaporation may not be apparent, a special logsheet is used for surveillance. The logsheet specifies the action to be taken if a specified minimum level is reached. The 1DF tank wt ftr and sp gr instrumentation are placed on the nuclear safety priority work preventive maintenance schedule for calibration and testing at routine intervals.

9. The following instrumentation, alarms and automatic cutoffs are included on the nuclear safety priority work preventive maintenance schedule.

   - 1DF tank temperature is tested on 18-mo intervals.
   - 1DF high temperature alarm and automatic high temperature cutoff are tested routinely.
   - 1DS conductivity meter and low acidity alarm is tested on 6-mo intervals.
   - 1DS flow measuring instrumentation and low flow alarm are tested routinely.
   - 1DS - FS flow measuring instrumentation and high flow alarm are tested routinely.
   - 1EU and 1EX flow measuring instrumentation and low flow alarm are tested routinely.

10. The following provisions apply to the operation of mixer-settler 1D:

    (a) The 1DF flow measuring instrumentation and high flow alarm are included on the nuclear safety priority work preventive maintenance schedule for calibration and testing at routine intervals.

    (b) The 1DF high flow alarm is set to alarm when the 1DF flow increases to a preset percentage above the specified flow rate.

(c) The cycle does not operate unless the 1DF flow instrument is functioning.

(d) The cycle must be shut down if an increase in 1DF flow is more than specified and is not corrected within the specified procedural time limit.

(e) Dilutent is not added to a second uranium cycle solvent unless authorized by a PCN (process change notice), which is prepared by Separations Technology and approved by the Separations and Separations Technology area supervisors.

(f) The cycle is not operated when the TBP analysis of the solvent is less than the minimum operating limit.

(g) Prior to the start of each campaign, it is confirmed that a NSB is installed in the process water line to 1DS head tanks.

(h) Second uranium cycle does not operate if the nuclear safety blank listed above is removed.

(i) The 1DS made up in cold feed preparation is sampled and the acidity determined to be within specifications before any of the solution is transferred to the head tanks.

(j) The cycle is shut down if the 1DS conductivity meter indicates the acidity of the 1DS is less than the minimum limit.

(k) The low flow alarms for the 1DS stream are set to alarm when flow decreases below a preset percentage of the specified flow rate.

(l) At least one of the two 1DS flow instruments must be functioning while the cycle is operating.

(m) The cycle must be shut down if a reduction of 1DS flow is more than specified and is not corrected within the time limit specified in the procedures.

(n) The high flow alarm for the 1DS-FS stream is set to alarm when flow increases a preset percentage above the specified flow rate.

(o) If the 1DS-FS flow measuring instrument is not functioning, the 1DS-FS second level rotometer must be read as specified in the procedures to verify that the flow rate is within specifications.

(p) The cycle must be shut down if an increase in 1DS-FS flow is more than specified and is not corrected within the time limit specified in the procedures.

(q) The 1DS heat exchanger is pressure tested routinely and verified to be free of an internal leak which could dilute the 1DS with steam. The completed pressure test procedure is filed in the facility QA file.

11. If the neutron count exceeds the limit established for any mixer settler corrective actions must return the neutron count to normal or the cycle is shut down.

12. The following provisions apply to the operation of mixer-settler 1E:

(a) The 1EX and 1EU low flow alarms are set to alarm at a preset percentage of their respective flow rates.

(b) The cycle does not operate unless at least one flow instrument (1EX or 1EU) is functioning.

(c) The cycle must be shut down if the 1EX or 1EU flow decreases more than specified and is not corrected within the time limit specified in the procedures.

(d) The 1EU colorimeter and 1EU conductivity instrumentation are included on the nuclear safety priority work preventive maintenance schedule for calibration and testing at 6-mo intervals.

(e) If the 1EU colorimeter or the 1EU conductivity reading exceeds the limit established for normal cycle operation, action is taken so that the colorimeter or conductivity reading is either returned to normal, or the cycle is shut down.

6.1.6 Second Neptunium Solvent Extraction Cycle

Procedural and administrative controls preclude a nuclear criticality resulting from a reflux of a critical mass of uranium in the 2B mixer settler by limiting the amount of material fed to second neptunium cycle. The inventory of fissile material is controlled to assure that the equivalent ^{235}U will always be less than the mass limit.

The second neptunium cycle inventory of ^{235}U is calculated routinely during normal operations. The cycle will not be operated if the calculated inventory is greater than the operating limit for ^{235}U, except to flush uranium from the 2B mixer-settler.

Criticality during concentration of 2BP is prevented by limiting the concentration of ^{235}U to less than 11.5 g/l. This is accomplished by limiting the amount of ^{235}U in the 2BP system inventory and by setting critically safe lower limits on evaporator endpoint levels.

6.1.7 Frames and Frames Waste Recovery

In the H-Area frames operation, irradiated neptunium targets are dissolved and ^{238}Pu is separated from unconverted ^{237}Np. The raw metal solution is processed through a series of anion exchange columns to separate and decontaminate neptunium and plutonium for transfer to HB-Line. Raffinate solutions from the frames containing small amounts of actinides are collected in the frame waste recovery system and processed through an additional anion exchange column to recover and concentrate the neptunium and plutonium for recycle to the frames.

The receipt or processing of scrap plutonium from HB-Line is restricted to more than 67 wt % ^{238}Pu.

A nuclear criticality could theoretically occur if enough ^{238}Pu were allowed to concentrate in a frame unit to form a critical mass. The probability of criticality in a frame vessel is very small, however, because of the large amount of ^{238}Pu required (8.15 kg) and the large amount of heat that would be generated before it reached criticality.

Because of their size and location in the process, the B-Line Solution Receipt tank, Frame Dissolver and Waste Hold tank, and FWR Raffinate tank/Feed tank are the only tanks where such an event is credible. In Frame I, Frame II, and the frame waste recovery vessels, physical size and process limitations of time and sequencing preclude the possibility of accumulating a critical mass of ^{238}Pu.

Procedures for the receipt of solution in the above mentioned tanks involve nuclear safety control steps.

6.1.8 Evaporation

Batch evaporators are used in Buildings 221-H and 211-H to reduce the volume of feed to solvent extraction cycles for operating efficiency, and to remove excess water and acid for economy in waste storage. Except for the general purpose evaporator, all are acidic evaporations so that nitric acid can be recovered from the overheads.

A nuclear criticality could be caused by exceeding the critical concentration of fissile material in an evaporator. The head end, 1CU, rerun, HAW, and LAW evaporations are where a criticality could occur. There is no credible mechanism for a criticality incident in the general purpose evaporator. Unplanned nuclear criticalities are prevented by administrative controls that limit areal density and concentration of ^{235}Pu.

(a) The concentration of ^{235}U is maintained within the maximum safe limits of 11.5 g/l (1).

Procedures for evaporation processes involving solution containing fissile material implement the applicable conditions and limitations of the Nuclear Safety Technical Standards.

(b) Inadvertant addition of a precipitant to a solution that contains fissile material that is to be concentrated by evaporation (except for caustic evaporation) is prevented.

- To prevent the addition of sodium hydroxide to the high activity waste system, a NSB is installed in the line from chemical make-up tank to HAW evaporator feed tank.

- Separations Technology reviews indirect repair orders and procedures and verifies before approving them that the work will not allow a precipitant to be added to an evaporator or evaporator feed tank containing fissile materials.

Canyon cell covers are designed to provide adequate shielding for personnel in the areas that would be affected by a nuclear criticality. Also, nuclear incident monitors located in the warm canyon will alert personnel to the danger and cause them to evacuate.

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 exceeding the safe temperature limit of 130°C (1).

- The amount of entrained solvent is limited 0.5 volume percent.

- The evaporator temperature is limited by restricting the steam coil pressure to 25 psig.

6.1.9 Rerun

The rerun system in H-Canyon converts off-standard process streams and sump materials into solutions suitable for introduction into the HM process. The system is also used to concentrate and dispose of aqueous waste streams, as well as to treat and dispose of spent solvent.

A nuclear criticality could occur in the rerun system through accumulation of fissile material if a precipitant (e.g., a caustic sump solution) was inadvertently added to either of the two sump collection tanks and, the sump solution hold tank, the rerun hold tank, or the batch extraction tank. During batch extraction, if an insufficient quantity of solvent or aqueous strip solution is used when enough fissile material is present, a nuclear criticality could result. Unplanned nuclear criticalities are prevented by administrative controls. The limits for mass, areal density, and concentration of equivalent ^{235}U are implemented for rerun processing by imposing operating limits chosen to ensure remaining well within the applicable Technical Standard limits.

Sump Collection Tanks. Before any new material can be transferred into a sump collection tank, the contents of the tank must be acidic and the tank must contain less than Technical Standard limit of equivalent ^{235}U. The solution in the tank must be acidic to assure that no precipitate has formed. A representative sample must be obtained. Laboratory analyses of duplicate samples must agree within the limits specified to assure that the samples are representative of the tank contents. The limit on fissile material content of the tank prior to receiving another sump solution assures that initially the tank will only contain a small amount of uranium should the sump solution be a precipitant.

Sump Collection Tank procedures for the receipt and transfer comply with these standards.

Sump Solution Hold Tank. The maximum amount of equivalent ^{235}U in the Sump Solution Hold tank must be less than the Technical Standard areal density limit.

Rerun Hold Tank. The maximum amount of equivalent ^{235}U in the rerun hold tank must be less than the Technical Standard areal density limit. Each receipt of sump solution must be analyzed by the laboratory from duplicate samples taken in sump collection tanks and the results must agree with the specified limits. Before receiving any solution to be processed by batch extraction, the equivalent ^{235}U inventory is calculated and verified to be below the operating limit. As an added precaution, the final concentration of solution in the Rerun Hold tank is calculated and verified to be below the Technical Standard nuclear safety concentration limit of 9.6 grams of equivalent ^{235}U per liter before any receipt (1).

Batch Extraction Tank. For batch extraction, areal density limit prevents a criticality. As an additional precaution, the solution volumes during batch extraction are specified to assure that the nuclear safety concentration limit will not be exceeded, either in the solvent or the strip solution. The batch extraction procedure requires verification that the batch extraction tank must contain at least the minimum specific amount of solvent, and that each aqueous strip must contain more than the minimum specified amount of process water. Batch extraction procedures are followed that contain nuclear safety controls and operating limits necessary to prevent a criticality.

Prevention of Precipitant Additions.

- A NSB is installed in the chemical addition line from chemical make-up tank to Rerun Hold tank to prevent addition of sodium hydroxide to the rerun system.

- Separations Technology reviews indirect work requests and procedures, and verifies before approving them that the work will not allow a precipitant to be added to a vessel that might contain fissile material.

6.1.10 Waste Disposal

In the waste disposal systems of Building 221-H, acidic waste concentrates from solvent extraction, solvent washes, decontamination solutions, frame waste recovery (FWR), resin digestion, and sumps are neutralized and transferred to Building 241-H waste storage tanks via the waste headers.

A nuclear criticality could only occur during neutralization by exceeding the critical mass limit of fissile material. The HAW, LAW, and FWR/Rerun neutralization tanks are the tanks where there is a credible mechanism for this to occur. There is no credible mechanism for this to occur in the resin digestion tank, the general purpose bottoms tank, or the waste headers.

The amount of fissile material in a tank during neutralization is maintained at less than the critical mass limit through administrative controls and process instrumentation, so that only through multiple failures or errors could criticality occur. Appropriate nuclear safety controls are included in all procedures for waste disposal of uranium-bearing solutions. In the event of a criticality in the LAW neutralization tank, personnel working on the warm canyon crane or the maintenance bridge directly over the incident could be subjected to a lethal dose of radiation if the cell covers were off.

A criticality in the unoccupied, shielded, hot canyon would present no hazard to personnel.

Cell covers provide adequate shielding for personnel working on the warm crane or in the canyon proper. Verification that the proper cell covers are in place, and all personnel are evacuated from Sections 7 through 9 of the warm gang valve corridor while the LAW concentrate neutralization tank is being neutralized. The cell covers over warm canyon modules 7.8, 8.5, 8.6, 8.7, and 8.8 are painted red to facilitate verification that they are in place.

If any red cell cover is off, the warm canyon, warm canyon access walkway, and truckwell must be evacuated during neutralization of LAW concentrate neutralization tank contents.

6.1.11 Chemical Handling and Storage

The primary operations in the chemical storage area of Building 211-H are the receipt, storage, and distribution of new chemical supplies for H-Area processes. Tributyl phosphate, n-paraffin, and sodium hydroxide in solution are received in tank cars. Aluminum nitrate solution is also received in tank cars, but only rarely, as usage is very small. Concentrated nitric acid is received in tank trailers. The chemical handling and storage facilities in H-Area include unloading, storage, solution makeup, and pumping of new chemicals in Building 211-H, the 221-H southwest loading dock, 221-H cold feed preparation area, 221-H north loading dock, and third level chemical storage area.

There are two types of incidents which could cause death or permanent disability to personnel. These are explosion or fire from a chemical spill, and exposure to toxic chemicals. These hazards were 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 procedural control and protective equipment as follows:

- Safety Manual Items 85 and 86 define the Minimum Protective Equipment for Handling Hazardous Materials and Toxic Materials.
- 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). Employee exposure is monitored and maintained below the TLV by the Health Protection Department (2).
- New employees receive occupational health training as part of the new employee orientation. Retraining is conducted at least annually as discussed in Section 4.

6.1.12 Ventilation System

Three ventilation systems supply air to Building 221-H: The canyon air system; the center section air system; and the gang valve corridor air system. Air is exhausted by five separate systems: the canyon air exhaust; the center section exhaust; the B-line exhaust; the process vessel vent system; and the recycle vessel vent system.

The ventilation system is designed to prevent spread of airborne contamination to clean areas. This containment is accomplished through the use of a multiple air zone concept, maintaining positive static pressure in clean areas, essentially atmospheric pressure in areas with low contamination potential, and slight vacuum in area with high contamination potential.

The quantities of ammonium nitrate that have accumulated periodically in the process vessel vent filters are large enough to produce pressurization and serious damage to the canyon if such an accumulation should explode. This event was discussed in Section 5.1.9.1 and is analyzed in detail in Reference 3.

6.1.13 Cooling Water System

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, this 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 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-H. 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 281-2H pump basin from where it is returned to the Building 285 reservoir without monitoring. Although the clean water return cannot be contaminated by the equipment from which it discharges, it can contain activity which is 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 281-4H monitor, then flows through the 281-1H delaying basin to the 281-2H pump basin, from where it is returned to the Building 285-H reservoir.

- The segregated cooling water accumulates from canyon vessel coils which are supplied with both cooling water and steam. It flows underground to the 281-6H monitor, into the 281-5H delaying basin, then is discarded to Four Mile Creek if within discard limits. In H-Area there are two basins which can be filled and drained alternately, allowing time for sample analysis before each basin is drained to the creek.

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 281-8H 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 min, a low water level may be reached in the Building 285-H reservoir, causing the segregating valve to close and leaving only the independent cooling water supply headers pressurized.

Action plans for minimizing the release of radioactive materials include:

1) Pressure in the coils of vessels containing radioactive materials must be maintained, 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 281-6H 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 routine intervals until monitor operation 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 281-8H retention basin before any major release of activity to the environment has occurred. Procedural control of this action is provided.

4) A functional check of the segregated cooling water monitor is performed daily by Health Protection personnel.

6.1.14 Emergency Power Distribution

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

The main emergency power source for Building 221-H is a 1080-hp diesel engine, [redacted]. This diesel drives a 1000-kw, 460-volt generator to furnish 1600 amps. The diesel is automatically started by batteries and reaches full load in approximately 15 seconds in the event of a normal power outage. A similar emergency diesel and generator are located in Building 292-H. 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 H-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 H-Canyon operations. The requirements include:

- 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).

3. Treatment of Process Vessel Vent Gases. Internal Report DPSOP-77-272-271, E. I. du Pont de Nemours and Co., Savannah River Plant, Aiken, SC, October 26, 1977.

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 suitably 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, that 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, _in vivo_, 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
Acceptable Exposure Level	AEL
Curie	Ci
Cold Feed Preparation	CFP
Cathode Ray Tube	CRT
Chemical Warfare Service	CWS
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
Frame Waste Recovery	FWR
Hydroxylamine Nitrate	HAN
High Activity Waste	HAW
High Efficiency Particulate Air	HEPA
High Flux Isotope Reactor	HFIR
Health Monitoring	HM
Health Protection	HP

Immediate Danger to Life and Health	IDLH
Low Activity Waste	LAW
Motor Control Center	MCC
Materials Test Facility	MTF
Maximum Resistance Construction	MRC
Nuclear Incident Monitor	NIM
Normal Paraffin Hydrocarbon	NPH
Nuclear Safety Blanks	NSB
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
Reactuer a Haut Flux	RHF
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
Water Gage Static Pressure	WGSP

APPENDIX A

H-CANYON EMERGENCY POWER TO CRITICAL EQUIPMENT

EMERGENCY POWER

There are four motor control centers on the first level of Building 221-H, in sections [redacted] In each motor control center, there are a number of panelboards from which power is supplied to equipment throughout the building, including H-Area B-line and the Building 211-H facility.

Emergency power is provided for Building 221-H and H-Area B-line by a 1,080-hp diesel located in [redacted] Emergency power is available to critical equipment such as gang valves, lighting, air compressors, and ventilation equipment. Selected motor control center panelboards are tied into the emergency power system. In case normal power is lost, the emergency diesel generator starts automatically, to provide power to these control center panels. An emergency power supply is provided so that a safe and orderly shutdown of the various canyon processes can be made; normal processing operations are suspended whenever normal power is lost.

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] Alternate Power Feeder [redacted]

Warm Canyon Shielding Door

Warm Canyon Crane
Hot Canyon Shielding Door

Instrument Air Dryer
Battery Charger for Emergency Diesel Batteries
Cooling System for Emergency Diesel

Panels, Breathing Air Relay Cabinet
Emergency Light Panels [redacted] -- Bell phones, truckwell lights
Emergency Light Panels [redacted] -- Shielding doors control, "Halon"

Locker Room Exhaust Fan
Plant Air Compressor No. 2

Cooling Water Pump No. 1 - For Air Compressors
Cooling Water Pump No. 2 - For Air Compressors
Lighting Panel [redacted] Sect. 2, Lev. 1
(The alternate power supply for panel AA is from [redacted]

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Emergency Feeder [redacted] for Building 284-H and 241 Cooling Pump
Hot Canyon Crane

Locker Room Exhauster, Sect. 4, Lev. 1 -- Interlocked with switch No. 5 on panel [redacted]

Plant Air Compressor No. 1

NOTE:
- Instrument Air Compressor No. 1 is supplied by emergency power feeder [redacted] or alternate power feeder [redacted]
- Instrument Air Compressor No. 2 is supplied only by emergency power feeder [redacted]
- Instrument Air Compressor No. 3 is supplied by emergency power feeder [redacted] or alternate power feeder [redacted]

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted] Aqueous to [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] to Noz [redacted] for jet flush (Also to Noz [redacted] spare)
[redacted]
SPARE - Gang Valve at [redacted] is removed

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Emergency Light Panel EMH, [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Instrument Power Panel AC - Power to 221-H PA System & PAX Phone System
Emergency Feeder [redacted] for 704-H
Gang Valve Motor; [redacted] Solvent to Center Sect.

Gang Valve Motor; [redacted] Center Sect. to [redacted]
Gang Valve Motor; [redacted] Spray

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] to Bline

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] to Waste Hdr. [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Spray

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

"DELETED VERSION"

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

"DELETED VERSION"

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Sump [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

SPARE (5-1)
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

SPARE (7-1)
SPARE - Gang Valve removed (7-2)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Pump "A" on vessel [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Pump "B" on vessel [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] Cent. Sect. to [redacted] Aqueous Sect.
Gang Valve Motor; [redacted] Aqueous Sect. [redacted]

"DELETED VERSION"

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]

SPARE; [redacted] No motor
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] Sludge Transfer
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Motor-operated valve in [redacted] normal cooling water header; Sect. 4, Lev. 2
Motor-operated valve in [redacted] normal cooling water header; Sect. 4, Lev. 2

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] / Alternate Power Feeder [redacted]

SPARE; Used to power [redacted] position
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] to Waste Hdr 2
Gang Valve Motor; [redacted] to Waste Hdr 1

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Spray

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

[redacted] Emergency Power Feeder [redacted] /Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Sump [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] (Receives Power from [redacted])
Breaker on north wall. Alternate power supply to [redacted] which supplies power to Instrument Power Panels [redacted]

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted] Spray Emergency Light Panel EMB, Sect. 14, Lev. 1, N. Freight Elevator Lights
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted] Emergency Light Panel [redacted] Sect. 12, Lev. 3 -- Red Phone and Bell Phones in 221-H control room
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted]

[redacted] (Power Supply from [redacted])
Instrument Power Panel [redacted] -- WC Control Rm. Instruments Instrument Power Panel [redacted] -- Dispatch Office and HC Control Rm Instruments Instrument Power Panel [redacted] -- Bldg. H & V controls [redacted]

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Center Section Supply Air Blower

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Spray

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]
Gang Valve Motor; [redacted] spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]
Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Spray

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]
Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Cent. Sect. to Aqueous Sect.
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Spray

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted] Spray

Gang Valve Motor; SPARE (5-1)
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Solvent to Cent. Sect. [redacted] Cent. Sect. to Aqueous Sect.

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
SPARE (1-2)

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Inst. Power Panel [redacted] Sect. 18, Lev. 1

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Power to 251-H Control House — Alternate power supply

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]
Power to 211-H Control House & Outside Facility NIM System

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Spray

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Cent. Sect. to [redacted] Aqueous Sect.

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; Multi-purpose [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

WC Fire Water System, Motorized Valve in WGVC; Panel [redacted]
WC Fire Water System, Motorized Valve in Sect. 18, Lev. 1; Panel [redacted]

"DELETED VERSION"

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted] Emergency Domestic Water Pump for Process Air Compressors (POWER)
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Spray Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Gang Valve Motor; [redacted]

"DELETED VERSION"

"DELETED VERSION"

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; Dual Purpose Gang Valve - [redacted] spray - No jet

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; Dual Purpose Gang Valve - [redacted]

Gang Valve Motor; [redacted] Sparge

Gang Valve Motor; [redacted]
Hot Canyon Fire Water System, Motorized Valve in HGVC, Sect. 18; Panel [redacted]

[redacted] Emergency Power Feeder [redacted] Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted] Spray

Gang Valve Motor; [redacted]

Gang Valve Motor; Dual Purpose Gang Valve - [redacted]

Gang Valve Motor; [redacted] Spray

Gang Valve Motor; [redacted]
Gang Valve Motor; Dual Purpose Gang Valve - [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

HC Fire Water System, Motorized Valve Sect. [redacted]
Gang Valve Motor; [redacted]

"DELETED VERSION"

TABLE A-1. Emergency Power to Critical Equipment (Continued)

[redacted] Emergency Power Feeder [redacted], Alternate Power Feeder [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; Dual Purpose Gang Valve - [redacted]

Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted] Spray
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

Gang Valve Motor; [redacted]
Gang Valve Motor; [redacted]

[redacted] Can be Supplied Emergency Power from [redacted] or Can be Supplied Normal Power from Feeder [redacted]

On West Side of Panel - Process Air Dryer
On East Side of Panel - Process Air Compressor No. 1, 50 HP RIX Unit

[redacted] Can be Supplied Emergency Power from [redacted] or Can be Supplied Normal Power from Feeder [redacted]

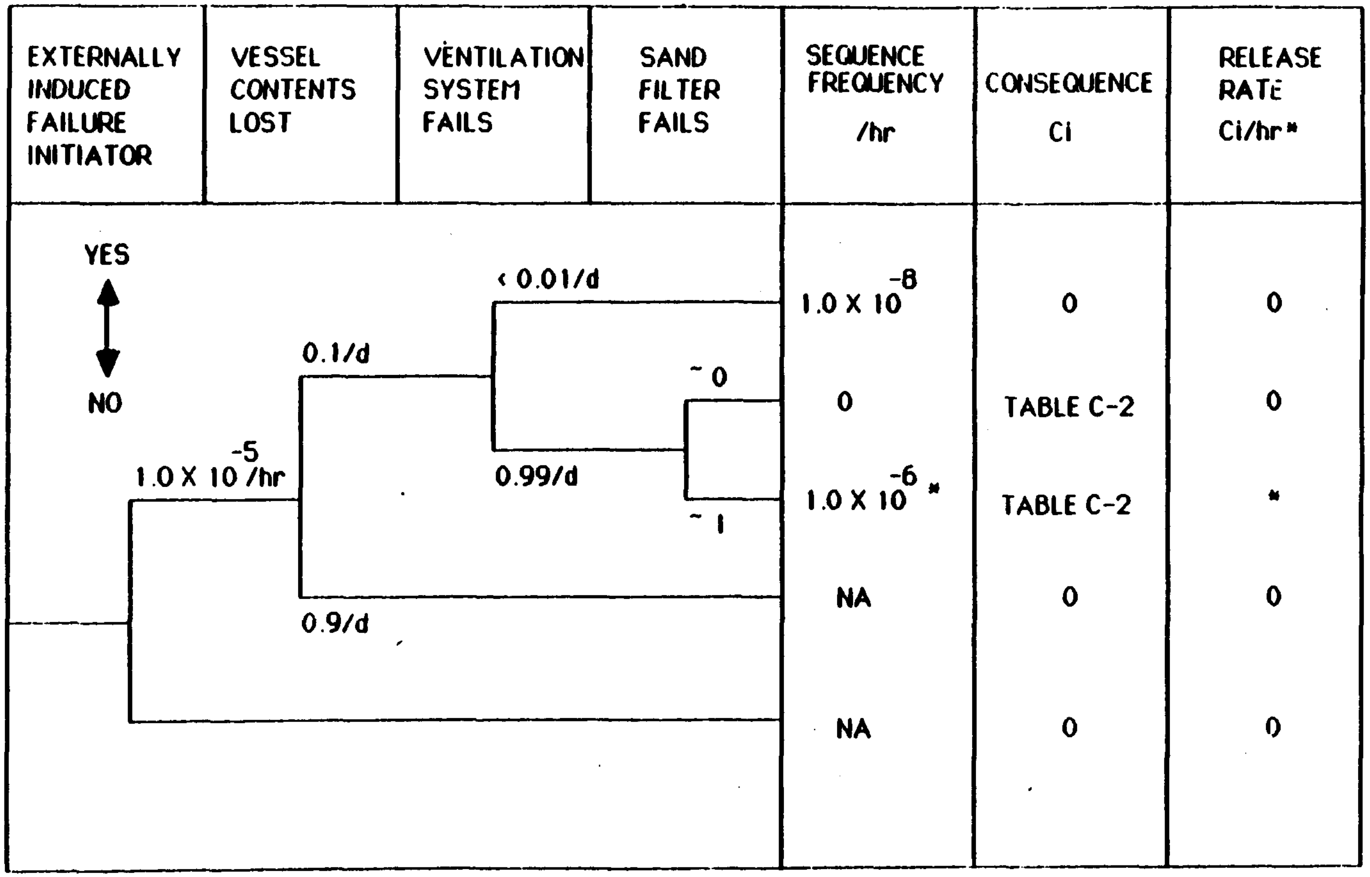
Process Air Compressor No. 2, 25 HP Gardener Denver Unit

[NOTE: Power to [redacted] and [redacted] is via transfer switches. When emergency power is provided to one of these MCC panels, the other panel is on normal power.]

"[redacted]LETED VERSION"

APPENDIX B

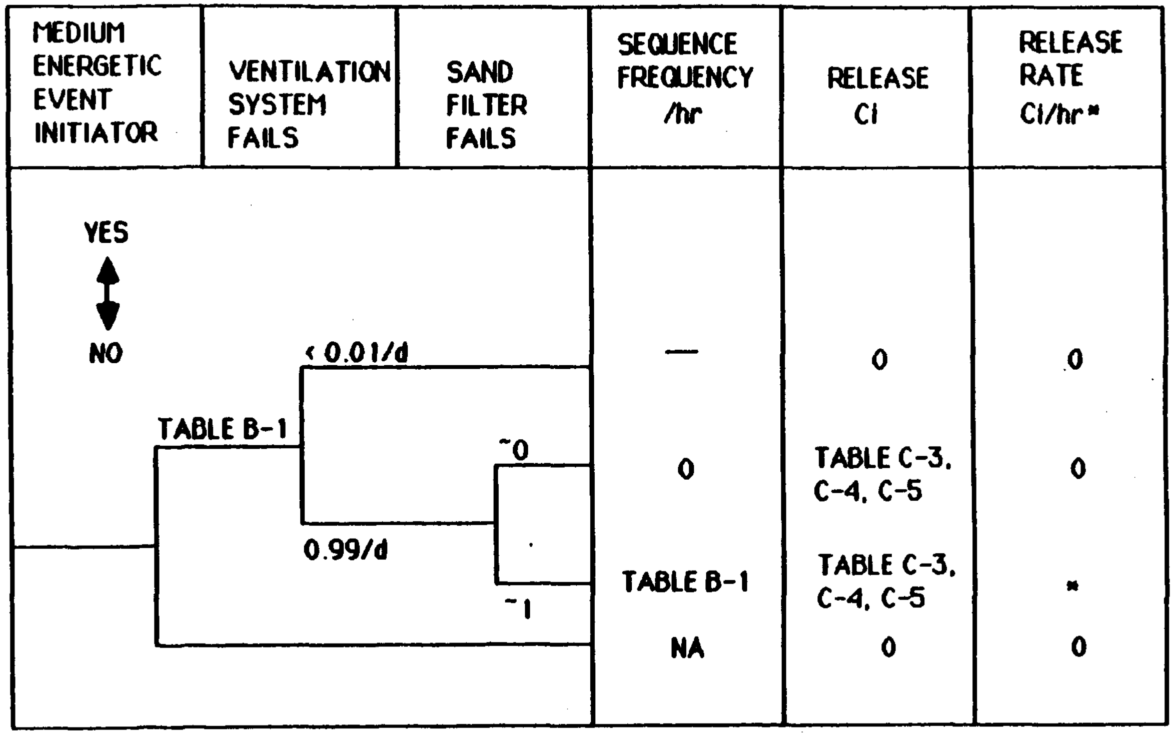
H-CANYON EVENT TREES AND FAULT TREE



* SEQUENCE FRQUENCY IS 1 X 10^{-6} FOR EACH UNIT OPERATION

** RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

FIGURE B-1 Event Tree For Externally Induced Failure Releases to the Stack.

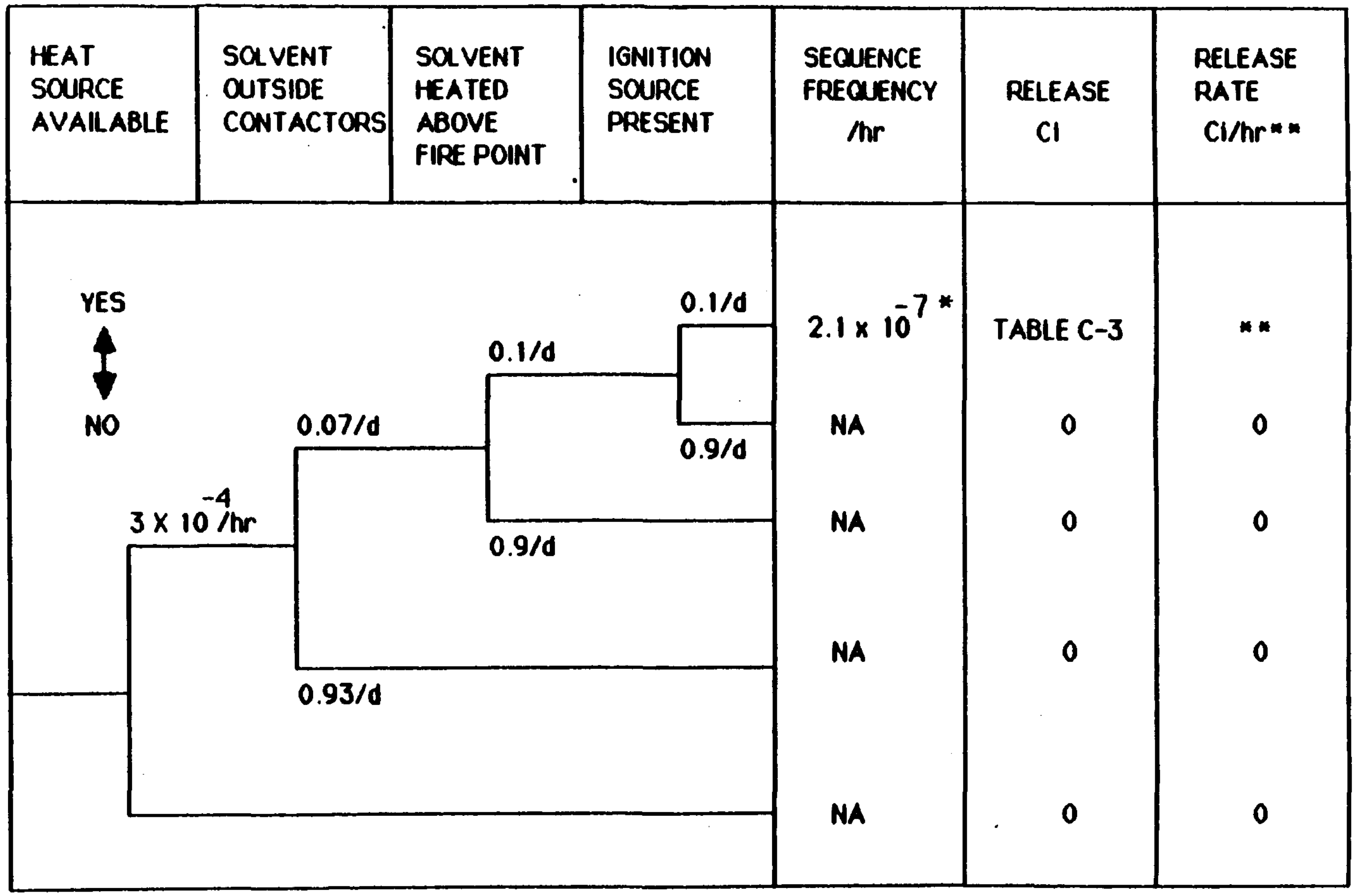


* RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

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

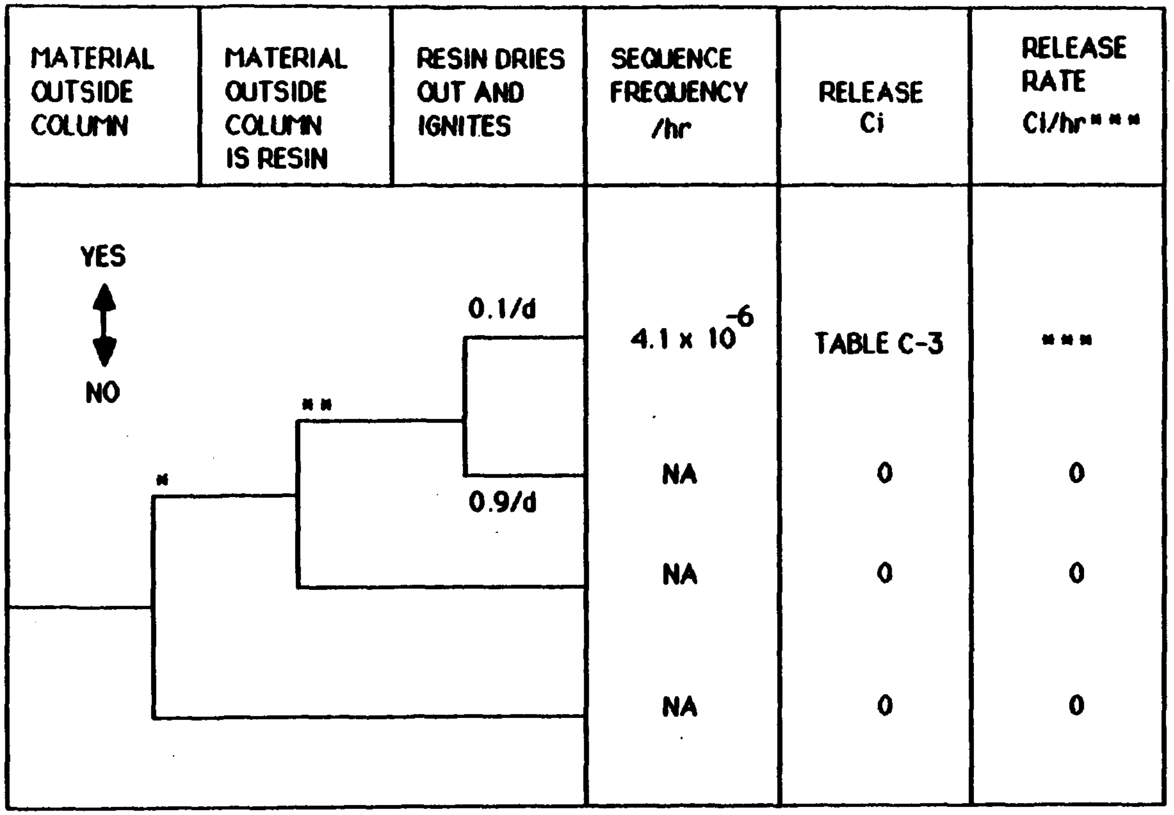
Initiator Location	Medium Energetic Event Initiator	Sequence Frequency, /hr	Release, Ci	Release Rate, Ci/hr
Dissolver	Uncontrolled Reaction	9.0E-05	1.6E-01	1.4E-05
Head End	Uncontrolled Reaction	2.0E-06	1.5E-01	3.0E-07
First Cycle	Fire	7.0E-08	5.8E-01	4.1E-08
	Uncontrolled Reaction	1.0E-05	9.8E-02	9.8E-07
2nd U Cycle	Fire	7.0E-08	9.4E-06	6.6E-13
	Uncontrolled Reaction	1.0E-05	3.9E-07	3.9E-12
2nd Np Cycle	Fire	7.0E-08	5.1E-04	3.6E-11
	Uncontrolled Reaction	1.0E-05	1.7E-05	1.7E-10
HAW Evaporator	Uncontrolled Reaction	3.9E-05	1.1E-01	4.3E-06
LAW Evaporator	Uncontrolled Reaction	2.6E-05	4.5E-06	1.2E-10
Ion Exchange	Fire	4.1E-06	2.9E-02	1.2E-07
	Uncontrolled Reaction	9.0E-06	7.5E-03	6.8E-08
	Explosion	5.0E-08	2.5E-02	1.3E-09
High Heat Waste	Uncontrolled Reaction	2.0E-05	1.1E-01	2.2E-06



* Each cycle assumed to have equal probability of fire (7×10^{-8} /hr)

** RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

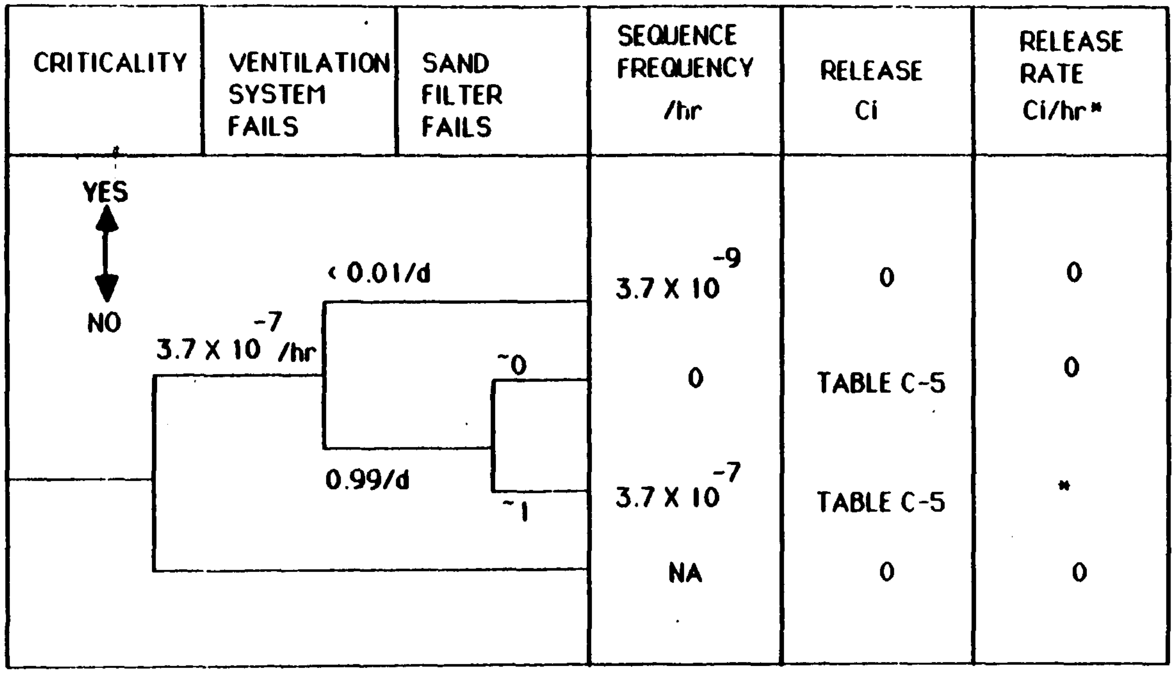
FIGURE B-3 Event Tree for Fire in Solvent Extraction



MECHANISM FOR MATERIAL OUTSIDE COLUMN	INITIATOR FREQUENCY, /hr*	MATERIAL OUTSIDE COLUMN IS RESIN, /d**	RESIN DRIES OUT AND IGNITES, /d	SEQUENCE FREQUENCY, /hr
LEAK	1.2×10^{-3}	0.03	0.1	3.6×10^{-6}
UNCONTROLLED REACTION	2.2×10^{-5}	0.21	0.1	4.6×10^{-7}
TOTAL				4.1×10^{-6}

*** RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

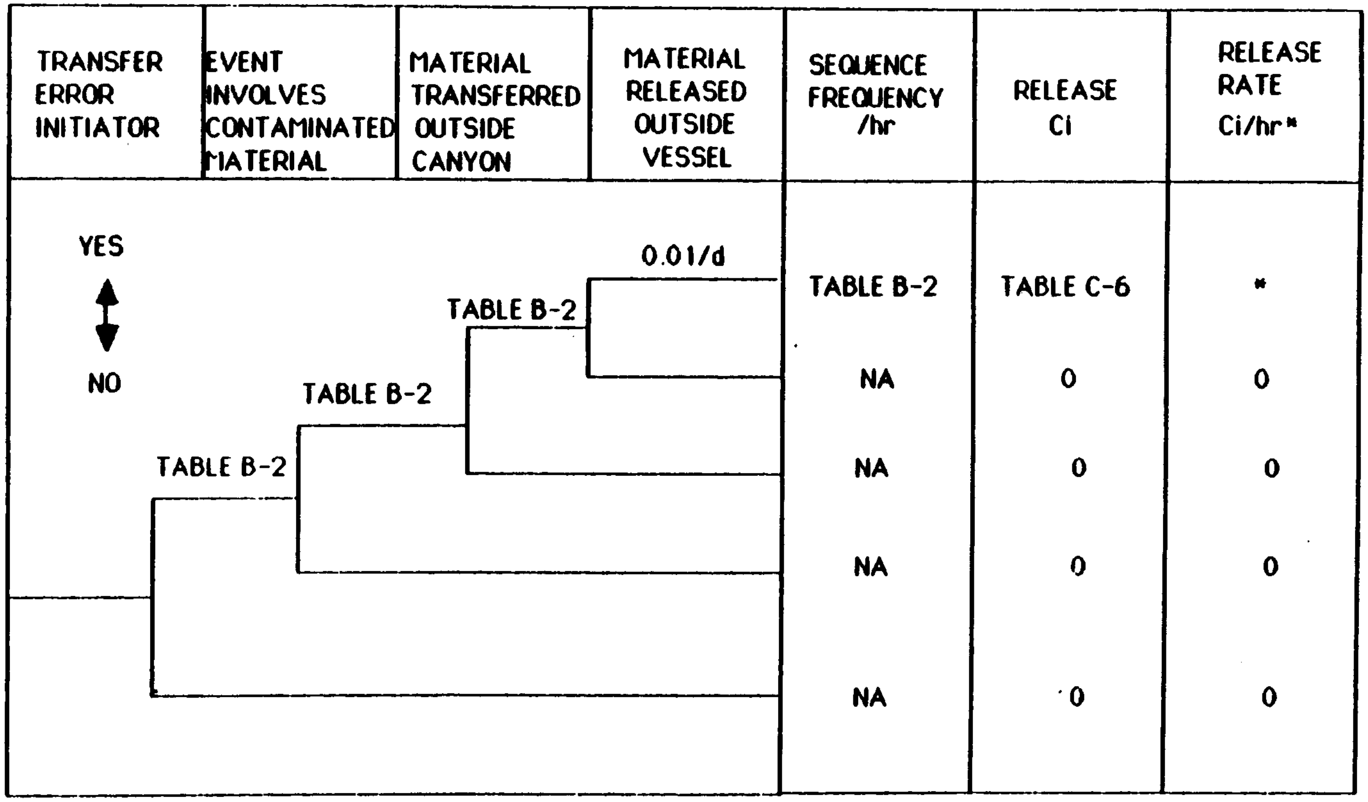
FIGURE B-4 Event Tree for Fire in Ion Exchange



* RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

SYSTEM	SEQUENCE FREQUENCY, /hr
DISSOLVING	1.5×10^{-9}
HEAD END	2.2×10^{-10}
HEAD END EVAPORATOR	1.2×10^{-8}
FIRST CYCLE	1.8×10^{-7}
2ND U CYCLE	1.8×10^{-7}
TOTAL	3.7×10^{-7}

FIGURE B-5 Event Tree for Criticality Release from the Stack

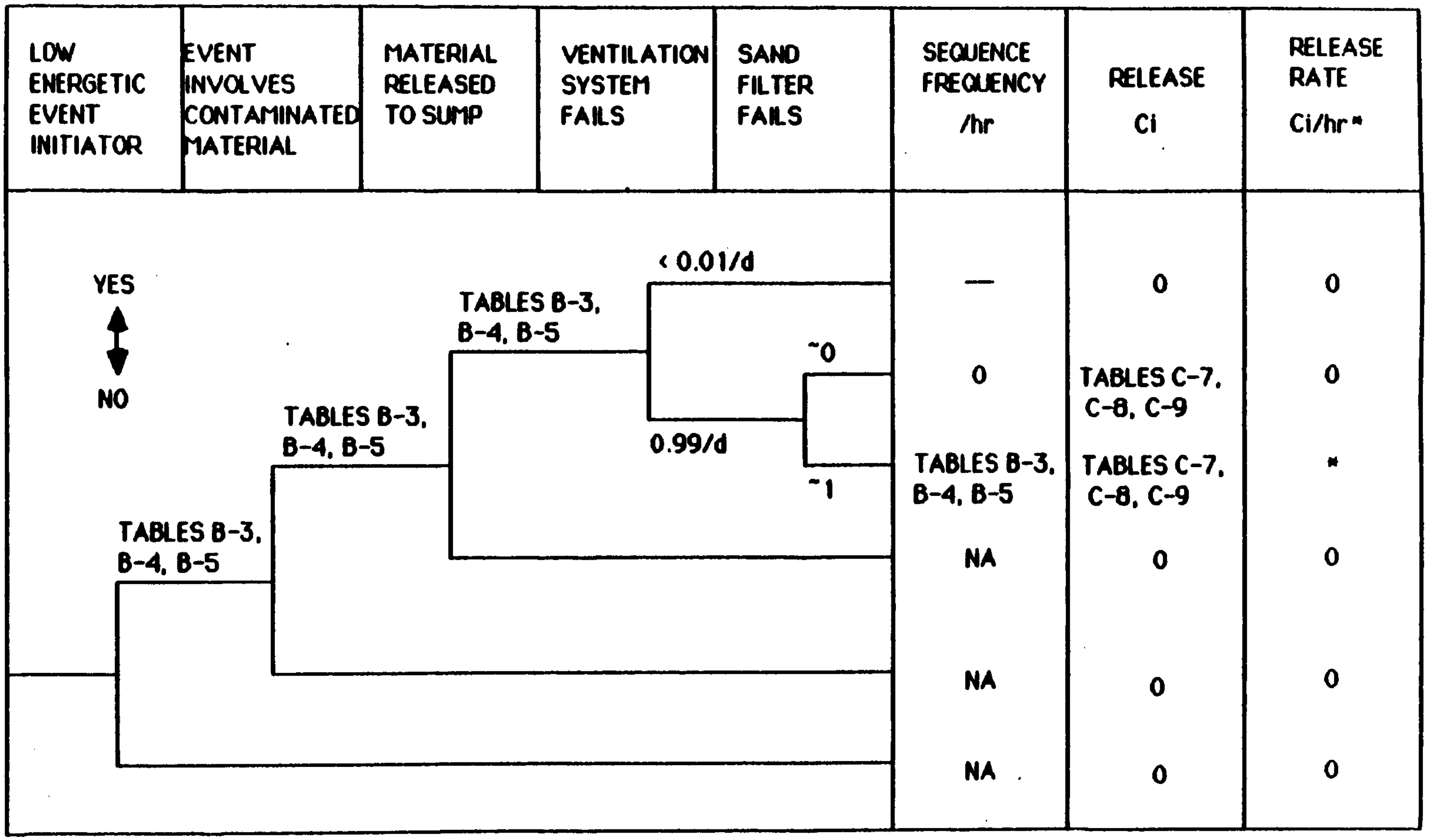


* RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

FIGURE B-6 Event Tree for Airborne 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	1.0E-05	5.0E-01	4.0E-02	1.0E-02	2.0E-09	1.5E-04	3.0E-13
Dissolver	1.00	4.0E-05	5.0E-01	4.0E-02	1.0E-02	8.0E-09	3.1E+00	2.5E-08
Head End	1.00	3.0E-05	1.0E+00	4.0E-02	1.0E-02	1.2E-08	2.2E+01	2.6E-07
First Cycle	0.45	1.4E-04	5.0E-01	4.0E-02	1.0E-02	1.3E-08	1.9E+01	2.4E-07
2nd U Cycle	0.15	1.4E-04	5.0E-01	4.0E-02	1.0E-02	4.2E-09	6.7E-09	2.8E-17
2nd Np Cycle	0.40	1.4E-04	5.0E-01	4.0E-02	1.0E-02	1.1E-08	1.2E-05	1.3E-13
HAW Evaporator	0.50	8.0E-05	1.0E+00	4.0E-02	1.0E-02	1.6E-08	2.1E+01	3.4E-07
LAW Evaporator	0.50	8.0E-05	1.0E+00	4.0E-02	1.0E-02	1.6E-08	8.8E-07	1.4E-14
Ion Exchange	1.00	2.3E-04	5.0E-01	4.0E-02	1.0E-02	4.6E-08	5.6E-02	2.6E-09
High Heat Waste	1.00	5.0E-05	1.0E+00	4.0E-02	1.0E-02	2.0E-08	2.1E+01	4.2E-07



* RELEASE RATE = SEQUENCE FREQUENCY (/hr) X RELEASE (Ci)

FIGURE B-7 Event Tree for Low Energetic Release from the Stack

TABLE B-3. Data for Quantification of Airborne Release from the Stack due to Transfer Error Event Tree

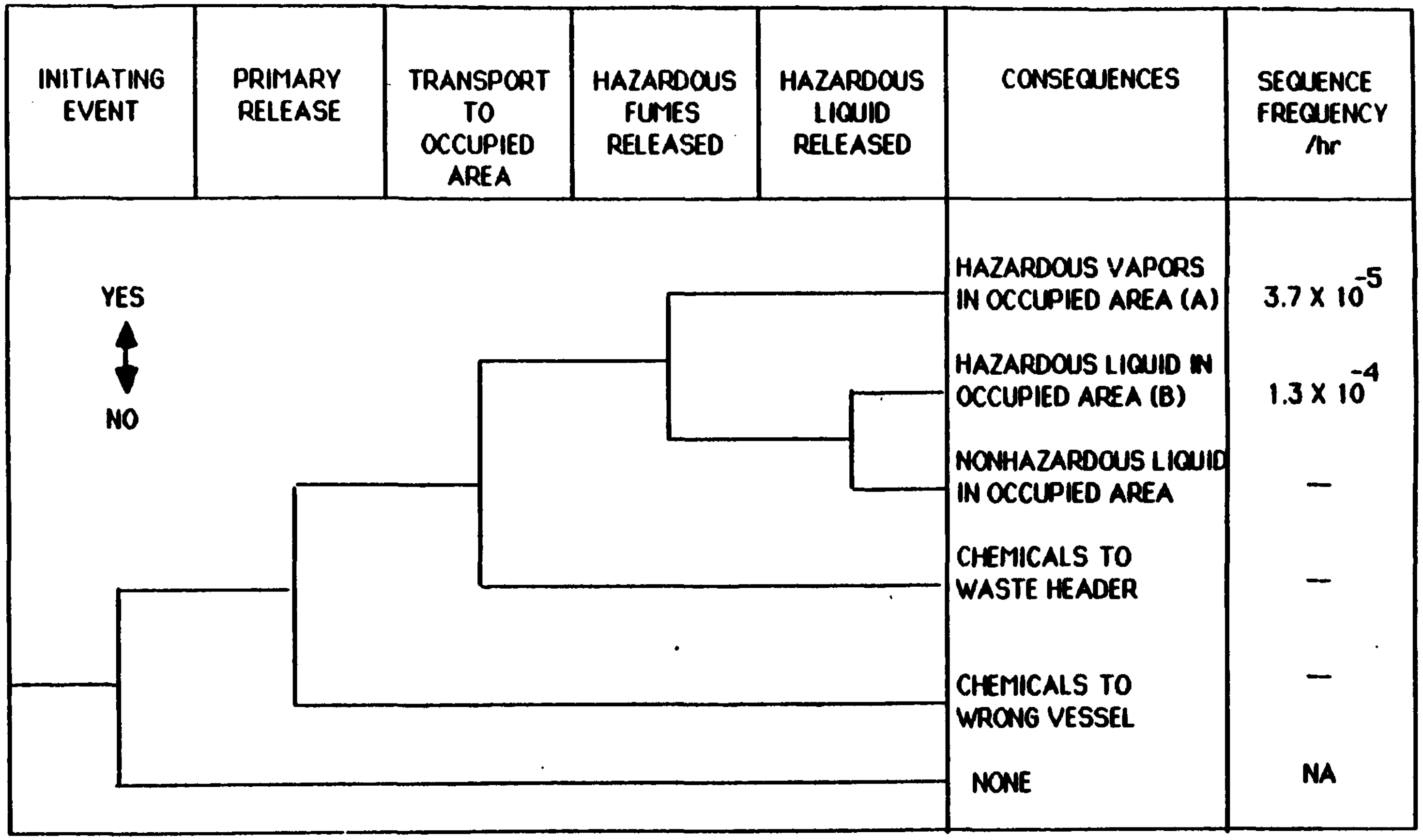
Initiator Location	Fraction	Initiator Frequency, /hr	Event Involves Contaminated Material, /d	Material Released to Sump, /d	Sequence Frequency, /hr	Release, Ci	Release Rate, Ci/hr
Decontamination	1.00	1.0E-05	5.0E-01	8.0E-02	4.0E-07	7.0E-07	2.8E-13
Dissolver	1.00	4.0E-05	5.0E-01	8.0E-02	1.6E-06	1.7E-01	2.7E-07
Head End	1.00	3.0E-05	1.0E+00	8.0E-02	2.4E-06	1.5E-01	3.6E-07
First Cycle	0.45	1.4E-04	5.0E-01	8.0E-02	2.5E-06	9.3E-02	2.3E-07
2nd U Cycle	0.15	1.4E-04	5.0E-01	8.0E-02	8.4E-07	3.3E-11	2.8E-17
2nd Np Cycle	0.40	1.4E-04	5.0E-01	8.0E-02	2.2E-06	5.9E-08	1.3E-13
HAW Evaporator	0.50	8.0E-05	1.0E+00	8.0E-02	3.2E-06	1.0E-01	3.3E-07
LAW Evaporator	0.50	8.0E-05	1.0E+00	8.0E-02	3.2E-06	4.3E-09	1.4E-14
Ion Exchange	1.00	2.3E-04	5.0E-01	8.0E-02	9.2E-06	2.8E-04	2.5E-09
High Heat Waste	1.00	5.0E-05	1.0E+00	8.0E-02	4.0E-06	1.0E-01	4.2E-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, /d	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	4.2E-07	2.1E-13
Dissolver	1.00	4.0E-05	1.0E+00	1.0E+00	4.0E-05	9.9E-02	4.0E-06
Head End	1.00	1.0E-05	1.0E+00	1.0E+00	1.0E-05	8.9E-02	8.9E-07
First Cycle	0.80	8.0E-05	6.0E-01	1.0E+00	3.8E-05	5.5E-02	2.1E-06
2nd U Cycle	0.07	8.0E-05	6.0E-01	1.0E+00	3.4E-06	1.9E-11	6.5E-17
2nd Np Cycle	0.13	8.0E-05	6.0E-01	1.0E+00	6.2E-06	3.5E-08	2.2E-13
HAW Evaporator	0.50	9.0E-05	1.0E+00	1.0E+00	4.5E-05	6.1E-02	2.7E-06
LAW Evaporator	0.50	9.0E-05	1.0E+00	1.0E+00	4.5E-05	2.5E-09	1.1E-13
Ion Exchange	1.00	1.0E-04	1.0E+00	1.0E+00	1.0E-04	5.4E-04	5.4E-08
High Heat Waste	1.00	6.0E-05	1.0E+00	1.0E+00	6.0E-05	6.1E-02	3.6E-06

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	8.0E-04	6.0E-01	1.0E+00	4.8E-04	1.4E-07	6.7E-11
Dissolver	1.00	6.2E-05	1.0E+00	1.0E+00	6.2E-05	1.4E-02	8.7E-07
Head End	1.00	5.0E-04	1.0E+00	1.0E+00	5.0E-04	1.3E-02	6.3E-06
First Cycle	0.32	3.0E-03	7.5E-01	1.0E+00	7.2E-04	7.6E-03	5.5E-06
2nd U Cycle	0.16	3.0E-03	7.5E-01	1.0E+00	3.6E-04	2.7E-12	9.8E-16
2nd Np Cycle	0.52	3.0E-03	7.5E-01	1.0E+00	1.2E-03	4.9E-09	5.7E-12
HAW Evaporator	0.50	7.0E-03	5.0E-01	1.0E+00	1.8E-03	8.6E-03	1.5E-05
LAW Evaporator	0.50	7.0E-03	5.0E-01	1.0E+00	1.8E-03	3.6E-10	6.3E-13
Ion Exchange	1.00	1.2E-03	1.0E+00	1.0E+00	1.2E-03	2.6E-04	3.1E-07
High Heat Waste	1.00	7.0E-05	1.0E+00	1.0E+00	7.0E-05	8.6E-03	6.0E-07



INITIATING EVENT	INITIATOR FREQUENCY, /hr	FREQUENCY FOR SUBSEQUENT EVENTS, /d				SEQUENCE FREQUENCY, /hr A	SEQUENCE FREQUENCY, /hr 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

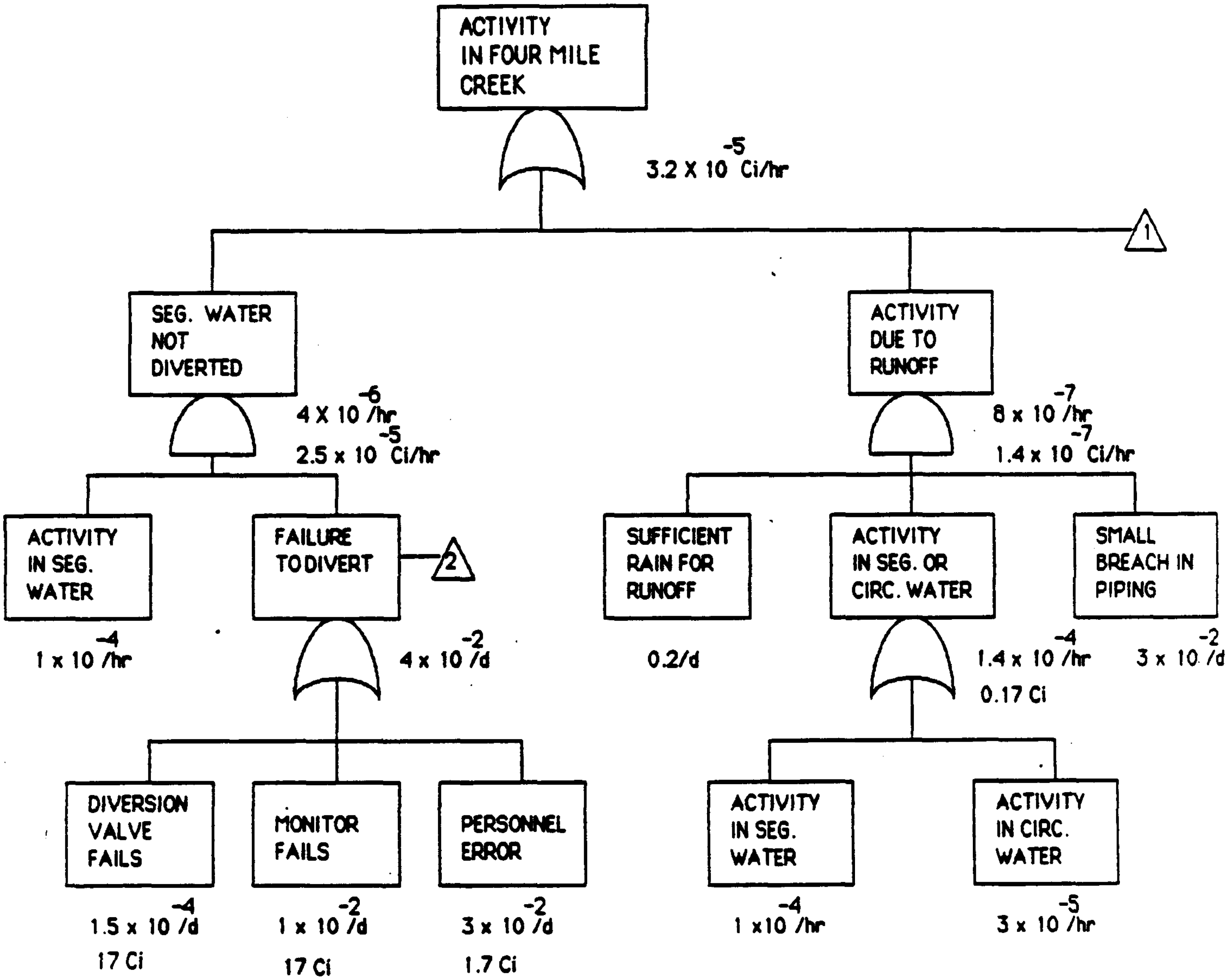


FIGURE B-9 Fault Tree for Liquid Releases to Four Mile Creek

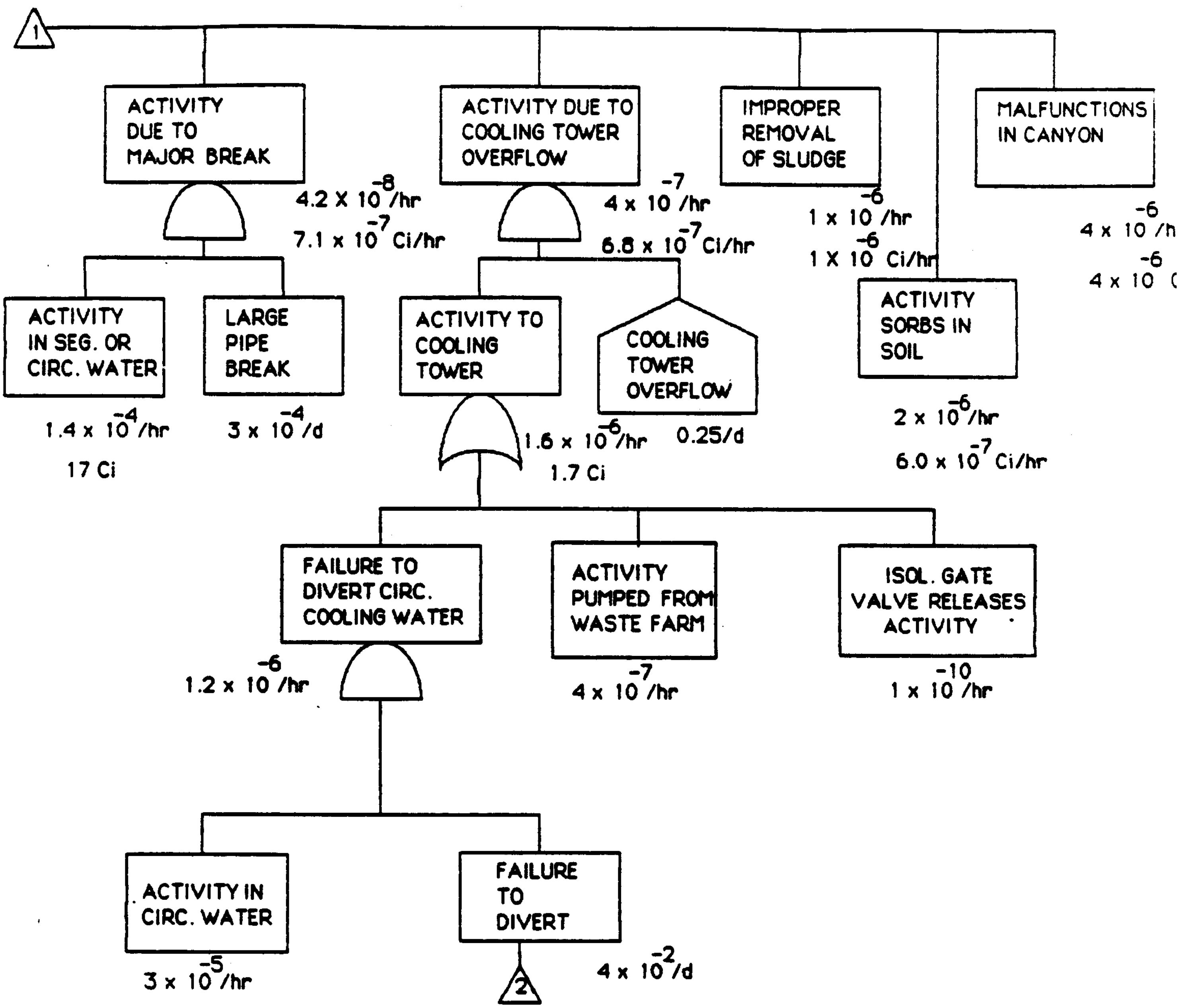


FIGURE B-9 Fault Tree for Liquid Releases to Four Mile Creek (continued)

APPENDIX C

H-CANYON SOURCE TERM CALCULATIONS

TABLE C-1. Curies of Specific Nuclides Released at Ground Level due to Earthquake (H-Canyon)

Nuclide	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Np Cycle	HAW Evaporator	LAW Evaporator	Ion Exchange	High Heat Waste
89Sr	2.41E-03	2.18E-03	1.87E-03	3.01E-13	3.43E-10	2.06E-03	5.32E-11	5.55E-07	2.06E-03
90Sr	1.66E-04	1.51E-04	1.31E-04	2.68E-14	1.16E-11	1.44E-04	3.68E-12	1.16E-07	1.44E-04
90Y	1.66E-04	1.51E-04	1.31E-04	2.68E-14	1.16E-11	1.44E-04	3.68E-12	1.16E-07	1.44E-04
91Y	3.73E-03	3.39E-03	2.93E-03	5.95E-13	2.67E-10	3.23E-03	8.46E-11	1.06E-06	3.23E-03
95Zr	3.39E-03	3.08E-03	2.71E-03	5.60E-13	2.40E-09	2.98E-03	7.09E-10	2.07E-06	2.98E-03
95Nb	9.83E-04	8.93E-04	7.67E-05	1.55E-13	3.38E-11	8.45E-05	1.17E-11	1.87E-06	8.45E-05
103Ru	8.14E-04	7.39E-04	4.06E-04	8.34E-14	6.23E-11	4.47E-04	1.77E-11	1.01E-08	4.47E-04
106Ru	6.10E-04	5.54E-04	3.16E-04	6.55E-14	4.45E-11	3.48E-04	1.36E-11	3.89E-08	3.48E-04
106Rh	6.10E-04	5.54E-04	4.74E-04	9.97E-14	4.23E-11	5.22E-04	1.34E-11	3.89E-06	5.22E-04
110Ag	1.46E-05	1.32E-05	1.15E-05	2.32E-15	1.02E-12	1.27E-05	3.27E-13	1.31E-07	1.27E-05
123Sn	1.49E-05	1.35E-05	1.17E-05	2.38E-15	1.02E-12	1.29E-05	3.27E-13	2.22E-09	1.29E-05
125Sb	1.90E-05	1.72E-05	1.47E-05	2.98E-15	1.29E-12	1.62E-05	4.09E-13	6.06E-08	1.62E-05
127Te	2.64E-05	2.40E-05	1.83E-05	4.17E-15	1.83E-12	2.01E-05	5.73E-13	7.57E-08	2.01E-05
129Te	1.49E-05	1.35E-05	1.17E-05	2.38E-15	1.02E-12	1.29E-05	3.27E-13	2.22E-09	1.29E-05
134Cs	3.73E-04	3.39E-04	2.93E-04	5.89E-14	2.54E-12	3.23E-04	8.05E-13	8.08E-08	3.23E-04
137Cs	6.78E-04	6.16E-04	4.06E-04	7.74E-14	3.47E-11	4.47E-04	1.11E-11	3.33E-07	4.47E-04
141Ce	7.46E-04	6.77E-04	5.87E-04	1.19E-13	5.34E-11	6.46E-04	1.64E-11	3.63E-07	6.46E-04
144Ce	8.82E-03	7.98E-03	6.77E-03	1.37E-12	5.79E-10	7.46E-03	1.91E-10	3.94E-06	7.46E-03
144Pr	8.82E-03	7.98E-03	6.77E-03	1.37E-12	5.79E-10	7.46E-03	1.91E-10	3.94E-06	7.46E-03
147Pm	9.49E-04	8.62E-04	7.44E-04	1.55E-13	6.68E-11	8.20E-04	2.18E-11	8.58E-07	8.20E-04
148Pm	9.87E-04	8.93E-04	1.31E-03	2.62E-15	1.16E-12	1.44E-03	1.36E-12	7.57E-08	1.44E-03
154Eu	1.56E-05	1.42E-05	1.22E-05	0.00E+00	0.00E+00	1.34E-05	0.00E+00	2.32E-09	1.34E-05
234U	4.07E-08	3.69E-08	2.93E-08	5.95E-13	4.45E-15	2.98E-12	6.82E-16	4.90E-16	2.98E-12
235U	5.43E-20	4.92E-10	4.06E-20	8.34E-15	4.45E-15	4.97E-14	1.36E-17	6.56E-17	4.97E-14
236U	4.75E-09	4.31E-09	3.83E-09	8.34E-14	4.45E-15	3.98E-13	9.55E-17	6.06E-17	3.98E-13
238U	1.36E-21	1.23E-11	1.06E-21	2.20E-16	4.45E-15	9.94E-16	2.73E-19	1.72E-16	9.94E-16
237Np	2.48E-09	2.25E-09	1.94E-09	1.57E-15	1.60E-12	1.99E-11	4.64E-15	2.22E-09	1.99E-11
238Pu	9.16E-06	8.31E-06	6.99E-06	2.86E-16	1.25E-11	7.71E-06	3.68E-13	2.52E-05	7.71E-06
239Pu	7.80E-08	7.08E-08	6.09E-08	2.50E-10	1.07E-13	6.71E-08	3.27E-14	7.07E-09	6.71E-08
240Pu	5.76E-08	5.23E-08	4.51E-08	2.38E-10	8.02E-14	4.97E-08	2.32E-14	4.34E-09	4.97E-08
241Pu	2.61E-05	2.37E-05	2.05E-05	8.34E-16	4.02E-11	2.26E-05	1.04E-11	4.80E-07	2.26E-05
242Pu	1.15E-20	1.05E-10	9.02E-11	3.57E-21	4.45E-15	9.94E-11	4.09E-17	4.80E-10	9.94E-11
Total	3.44E-02	3.12E-02	2.60E-02	5.85E-12	4.60E-09	2.87E-02	1.36E-09	4.53E-05	2.87E-02
Frequency, /hr	2.28E-08	2.28E-08	2.28E-08	2.28E-08	2.28E-08	2.28E-08	2.28E-08	2.28E-08	2.28E-08

TABLE C-2. Curies of Specific Nuclides Released from 200-ft Stack due to Externally Induced Failures (H-Canyon)

Nuclide	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Np Cycle	HAW Evaporator	LAW Evaporator	Ion Exchange	High Heat Waste
89Sr	2.36E-02	2.14E-02	1.84E-02	3.74E-12	3.36E-09	2.02E-02	5.21E-10	5.44E-06	2.02E-02
90Sr	1.63E-03	1.48E-03	1.28E-03	2.63E-13	1.14E-10	1.41E-03	3.61E-11	1.14E-06	1.41E-03
90Y	1.63E-03	1.48E-03	1.28E-03	2.63E-13	1.14E-10	1.41E-03	3.61E-11	1.14E-06	1.41E-03
91Y	3.66E-02	3.32E-02	2.87E-02	5.84E-12	2.62E-09	3.17E-02	8.29E-10	1.04E-05	3.17E-02
95Zr	3.32E-02	3.02E-02	2.65E-02	5.49E-12	2.36E-08	2.92E-02	6.95E-09	2.03E-05	2.92E-02
95Nb	9.64E-03	8.75E-03	7.52E-04	1.52E-12	3.32E-10	8.28E-04	1.15E-10	1.83E-05	8.28E-04
103Ru	7.98E-01	7.24E-01	3.98E-01	8.17E-11	6.11E-08	4.39E-01	1.74E-08	9.90E-06	4.39E-01
106Ru	5.98E-01	5.43E-01	3.10E-01	6.42E-11	4.36E-08	3.41E-01	1.34E-08	3.81E-05	3.41E-01
106Rh	5.98E-03	5.43E-03	4.64E-03	9.77E-13	4.15E-10	5.12E-03	1.31E-10	3.81E-05	5.12E-03
110Ag	1.43E-04	1.30E-04	1.13E-04	2.28E-14	1.00E-11	1.24E-04	3.21E-12	1.29E-06	1.24E-04
123Sn	1.46E-04	1.33E-04	1.15E-04	2.33E-14	1.00E-11	1.27E-04	3.21E-12	2.18E-08	1.27E-04
125Sb	1.86E-04	1.69E-04	1.44E-04	2.92E-14	1.27E-11	1.58E-04	4.01E-12	5.94E-07	1.58E-04
127Te	2.59E-04	2.35E-04	1.79E-04	4.09E-14	1.79E-11	1.97E-04	5.61E-12	7.42E-07	1.97E-04
129Te	1.46E-04	1.33E-04	1.15E-04	2.33E-14	1.00E-11	1.27E-04	3.21E-12	2.16E-08	1.27E-04
134Cs	3.66E-03	3.32E-03	2.87E-03	5.78E-13	2.49E-11	3.17E-03	7.89E-12	7.92E-07	3.17E-03
137Cs	6.65E-03	6.03E-03	3.98E-03	7.59E-13	3.40E-10	4.39E-03	1.08E-10	3.27E-06	4.39E-03
141Ce	7.31E-03	6.64E-03	5.75E-03	1.17E-12	5.24E-10	6.33E-03	1.60E-10	3.56E-06	6.33E-03
144Ce	8.64E-02	7.82E-02	6.63E-02	1.34E-11	5.67E-09	7.31E-02	1.87E-09	3.86E-05	7.31E-02
144Pr	8.64E-02	7.82E-02	6.63E-02	1.34E-11	5.67E-09	7.31E-02	1.87E-09	3.86E-05	7.31E-02
147Pm	9.31E-03	8.45E-03	7.30E-03	1.52E-12	6.55E-10	8.04E-03	2.14E-10	8.41E-06	8.04E-03
148Pm	9.67E-03	8.75E-03	1.28E-02	2.57E-14	1.14E-11	1.41E-02	1.34E-11	7.42E-07	1.41E-02
154Eu	1.53E-04	1.39E-04	1.19E-04	0.00E+00	0.00E+00	1.32E-04	0.00E+00	2.28E-08	1.32E-04
234U	3.99E-07	3.62E-07	2.87E-07	5.84E-12	4.36E-14	2.92E-11	6.68E-15	4.80E-15	2.92E-11
235U	5.32E-19	4.83E-09	3.98E-19	8.17E-14	4.36E-14	4.87E-13	1.34E-16	6.43E-16	4.87E-13
236U	4.65E-08	4.22E-08	3.76E-08	8.17E-13	4.36E-14	3.90E-12	9.36E-16	5.94E-16	3.90E-12
238U	1.33E-20	1.21E-10	1.04E-20	2.16E-15	4.36E-14	9.74E-15	2.67E-18	1.68E-15	9.74E-15
237Np	2.43E-08	2.20E-08	1.90E-08	1.54E-14	1.57E-11	1.95E-10	4.54E-14	2.18E-08	1.95E-10
238Pu	8.97E-05	8.14E-05	6.85E-05	2.80E-15	1.22E-10	7.55E-05	3.61E-12	2.47E-04	7.55E-05
239Pu	7.64E-07	6.94E-07	5.97E-07	2.45E-17	1.05E-12	6.58E-07	3.21E-13	6.93E-08	6.58E-07
240Pu	5.65E-07	5.13E-07	4.42E-07	2.33E-17	7.86E-13	4.87E-07	2.27E-13	4.26E-08	4.87E-07
241Pu	2.56E-04	2.32E-04	2.01E-04	8.17E-15	3.94E-10	2.22E-04	1.02E-10	4.70E-06	2.22E-04
242Pu	1.13E-19	1.03E-09	8.84E-10	3.50E-20	4.36E-14	9.74E-10	4.01E-16	4.70E-09	9.74E-10
Total	1.72E+00	1.56E+00	9.56E-01	2.02E-10	1.49E-07	1.05E+00	4.37E-08	4.92E-04	1.05E+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

TABLE C-3. Curies of Specific Nuclides Released from 200-ft Stack due to Fire (H-Canyon)

Nuclide	First Cycle	2nd U Cycle	2nd Np Cycle	Resin Release in Ion Exchange: Evaporation Release	Leak Release	Both Mechanisms^
89Sr	4.15E-02	6.16E-07	3.53E-05	3.26E-06	2.86E-06	3.21E-04
90Sr	2.90E-03	4.24E-08	1.19E-06	6.82E-07	5.98E-07	6.72E-05
90Y	2.90E-03	4.24E-08	1.19E-06	6.82E-07	5.98E-07	6.72E-05
91Y	6.50E-02	9.62E-07	2.75E-05	6.22E-06	5.46E-06	6.14E-04
95Zr	6.00E-02	9.04E-07	2.47E-04	1.22E-05	1.07E-05	1.20E-03
95Nb	1.75E-03	2.58E-07	3.48E-06	1.10E-05	9.61E-06	1.08E-03
103Ru	9.00E-03	1.35E-07	6.41E-06	5.93E-06	5.20E-06	5.84E-04
106Ru	7.00E-03	1.06E-07	4.58E-05	2.28E-05	2.00E-05	2.25E-03
106Rh	1.05E-02	1.54E-07	4.35E-06	2.28E-05	2.00E-05	2.25E-03
110Ag	2.55E-04	3.75E-09	1.05E-07	7.71E-07	6.75E-07	7.60E-05
123Sn	2.60E-04	3.85E-09	1.05E-07	1.30E-08	1.14E-08	1.29E-06
125Sb	3.25E-04	4.81E-09	1.32E-07	3.56E-07	3.12E-07	3.51E-05
127Te	4.05E-04	6.73E-09	1.88E-07	4.45E-07	3.90E-07	4.38E-05
129Te	2.60E-04	3.85E-09	1.05E-07	1.30E-08	1.14E-08	1.29E-06
134Cs	6.50E-03	7.52E-08	2.61E-07	4.74E-07	4.16E-07	4.68E-05
137Cs	9.00E-03	1.25E-07	3.57E-06	1.96E-06	1.71E-06	1.93E-04
141Ce	1.30E-02	1.92E-07	5.50E-06	2.13E-06	1.87E-06	2.10E-04
144Ce	1.50E-01	2.21E-06	5.95E-05	2.31E-05	2.03E-05	2.28E-03
144Pr	1.50E-01	2.21E-06	5.95E-05	2.31E-05	2.03E-05	2.28E-03
147Pm	1.65E-02	2.50E-07	6.87E-06	5.04E-06	4.42E-06	4.97E-04
148Pm	2.90E-02	4.23E-09	1.19E-07	4.45E-07	3.90E-07	4.38E-05
154Eu	2.70E-04	0.00E+00	0.00E+00	1.36E-08	1.20E-08	1.34E-06
234U	6.50E-07	9.62E-07	4.58E-08	0.00E+00	2.60E-16	2.94E-15
235U	9.00E-08	1.35E-08	4.58E-10	0.00E+00	2.60E-16	2.94E-15
236U	8.50E-08	1.35E-07	4.58E-20	0.00E+00	2.60E-16	2.94E-15
238U	2.35E-10	3.56E-10	4.58E-10	0.00E+00	2.60E-16	2.94E-15
237Np	4.30E-08	2.60E-09	1.65E-17	1.30E-08	1.14E-08	1.29E-06
238Pu	1.55E-04	4.62E-10	1.28E-06	1.48E-04	1.30E-04	1.46E-02
239Pu	1.35E-06	4.04E-12	1.09E-08	4.15E-08	3.64E-08	4.09E-06
240Pu	1.00E-06	3.85E-12	8.24E-09	2.55E-08	2.23E-08	2.51E-06
241Pu	4.55E-04	1.35E-09	3.66E-06	2.82E-06	2.47E-06	2.78E-04
242Pu	2.00E-09	5.77E-15	4.58E-10	0.00E+00	0.00E+00	0.00E+00
241Am	0.00E+00	0.00E+00	0.00E+00	2.82E-09	2.47E-09	2.78E-07
Total	5.77E-01	9.42E-06	5.13E-04	2.95E-04	2.58E-04	2.90E-02
Frequency, /hr	7.00E-08	7.00E-08	7.00E-08	3.60E-06	4.60E-07	4.06E-06

^^Weighted by frequency of occurrence

^Weighted average release from both mechanisms x 100 to account for higher partition factor for fires

TABLE C-4. Curies of Specific Nuclides Released from 200-ft Stack due to Uncontrolled Reactions (H-Canyon)

Nuclide	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Np Cycle	HAW Evaporator	LAW Evaporator	Ion Exchange	Explosion in Ion Exchange	High Heat Waste
89Sr	3.45E-03	3.25E-03	3.03E-03	2.60E-08	1.34E-06	3.20E-03	1.75E-07	8.33E-05	2.78E-04	3.20E-03
90Sr	2.38E-04	2.25E-04	2.11E-04	1.83E-09	4.57E-08	2.23E-04	1.21E-08	1.77E-05	5.89E-05	2.23E-04
90Y	2.38E-04	2.25E-04	2.11E-04	1.83E-09	4.57E-08	2.23E-04	1.21E-08	1.77E-05	5.89E-05	2.23E-04
91Y	5.35E-03	5.05E-03	4.75E-03	4.07E-08	1.04E-06	5.41E-03	2.79E-07	1.56E-04	5.21E-04	5.01E-03
95Zr	4.86E-03	4.59E-03	4.38E-03	3.83E-08	9.40E-06	4.62E-03	2.34E-06	3.12E-04	1.04E-03	4.62E-03
95Nb	1.41E-03	1.33E-03	1.24E-04	1.06E-08	1.32E-07	1.31E-04	3.87E-08	2.81E-04	9.37E-04	1.31E-04
103Ru	6.44E-02	5.79E-02	3.19E-02	5.70E-09	2.44E-07	3.55E-02	5.99E-08	1.56E-04	5.20E-04	3.55E-02
106Ru	4.83E-02	4.34E-02	2.48E-02	4.48E-09	1.74E-07	2.76E-02	4.61E-08	5.83E-04	1.94E-03	2.76E-02
106Rh	8.75E-04	8.26E-04	7.67E-04	6.51E-09	1.65E-07	8.09E-04	4.41E-08	5.83E-04	1.94E-03	8.09E-04
110Ag	2.09E-05	1.97E-05	1.86E-05	1.58E-10	4.00E-09	1.96E-05	1.08E-09	1.98E-05	6.59E-05	1.96E-05
123Sn	2.14E-05	2.02E-05	1.90E-05	1.63E-10	4.00E-09	2.00E-05	1.08E-09	3.23E-07	1.08E-06	2.00E-05
125Sb	2.72E-05	2.57E-05	2.37E-05	2.03E-10	5.05E-09	2.50E-05	1.35E-09	9.06E-06	3.02E-05	2.50E-05
127Te	3.80E-05	3.58E-05	2.96E-05	2.85E-10	7.13E-09	3.12E-05	1.89E-09	1.15E-05	3.82E-05	3.12E-05
129Te	2.14E-05	2.02E-05	1.90E-05	1.63E-10	4.00E-09	2.00E-05	1.08E-09	3.33E-07	1.11E-06	2.00E-05
134Cs	5.35E-04	5.05E-04	4.75E-04	4.03E-09	9.92E-09	5.01E-04	2.66E-09	1.25E-05	4.16E-05	5.01E-04
137Cs	9.73E-04	9.18E-04	6.57E-04	5.29E-09	1.36E-07	6.93E-04	3.64E-08	5.00E-05	1.67E-04	6.93E-04
141Ce	1.07E-03	1.01E-03	9.50E-04	8.14E-09	2.09E-07	1.00E-03	5.40E-08	5.41E-05	1.80E-04	1.00E-03
144Ce	1.26E-02	1.19E-02	1.09E-02	9.36E-08	2.26E-06	1.16E-02	6.30E-07	5.93E-04	1.98E-03	1.16E-02
144Pr	1.26E-02	1.19E-02	1.09E-02	9.36E-08	2.26E-06	1.16E-02	6.30E-07	5.91E-04	1.97E-03	1.16E-02
147Pm	1.36E-03	1.28E-03	1.20E-03	1.06E-08	2.61E-07	1.27E-03	7.20E-08	1.25E-04	4.17E-04	1.27E-03
148Pm	1.41E-03	1.33E-04	2.11E-03	1.79E-10	4.52E-09	2.23E-03	4.50E-09	1.15E-05	3.82E-05	2.23E-03
154Eu	2.23E-05	2.11E-05	1.98E-05	0.00E+00	0.00E+00	2.08E-05	0.00E+00	3.44E-05	1.15E-04	2.08E-05
234U	5.83E-08	5.51E-08	4.75E-08	4.07E-08	1.74E-11	2.36E-13	2.25E-12	0.00E+00	0.00E+00	2.36E-13
235U	7.78E-10	7.34E-10	6.57E-10	5.70E-10	1.74E-11	3.94E-14	4.50E-14	0.00E+00	0.00E+00	3.94E-14
236U	6.81E-09	6.43E-09	6.21E-09	5.70E-10	1.74E-11	3.15E-13	3.15E-13	0.00E+00	0.00E+00	3.15E-13
238U	1.94E-11	1.84E-11	1.71E-11	1.51E-11	1.74E-11	7.88E-16	9.00E-16	0.00E+00	0.00E+00	7.88E-16
237Np	3.55E-09	3.35E-09	3.14E-09	1.10E-10	6.26E-09	1.58E-11	1.53E-11	3.33E-07	1.11E-06	1.58E-11
238Pu	1.31E-06	1.24E-05	1.13E-06	1.95E-11	4.87E-08	1.19E-05	1.21E-09	3.77E-03	1.26E-02	1.19E-05
239Pu	1.12E-07	1.06E-07	9.86E-08	1.71E-13	4.17E-10	1.04E-07	1.08E-10	1.04E-06	3.47E-06	1.04E-07
240Pu	8.27E-08	7.80E-08	7.30E-08	1.63E-13	3.13E-10	7.70E-08	7.65E-11	6.46E-07	2.15E-06	7.70E-08
241Pu	3.74E-05	3.53E-05	3.32E-05	5.70E-11	1.39E-07	3.50E-05	3.42E-08	7.18E-05	2.39E-04	3.50E-05
242Pu	1.65E-10	1.56E-10	1.46E-10	2.44E-17	1.74E-11	1.54E-10	1.35E-13	0.00E+00	0.00E+00	1.54E-10
Total	1.60E-01	1.45E-01	9.75E-02	3.94E-07	1.79E-05	1.07E-01	4.48E-06	7.54E-03	2.51E-02	1.06E-01
Frequency, /hr	9.00E-05	2.00E-06	1.00E-05	1.00E-05	1.00E-05	3.90E-05	2.60E-05	9.00E-06	5.00E-08	2.00E-05

TABLE C-5. Curies of Specific Nuclides Released from 200-ft Stack due to Criticality (H-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	3.70E-07

TABLE C-6. Curies of Specific Nuclides Released at Ground Level due to Transfer Error to 211-H (H-Canyon)

Nuclide	Decontamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Np Cycle	HAW Evaporator	LAW Evaporator	Ion Exchange	High Heat Waste
89Sr	1.05E-05	4.74E-01	4.26E-01	3.65E-01	1.24E-10	2.73E-07	4.06E-01	1.05E-08	6.21E-04	4.06E-01
90Sr	7.31E-07	3.27E-02	2.94E-02	2.55E-02	8.71E-12	9.22E-09	2.84E-02	7.25E-10	1.30E-04	2.84E-02
90Y	7.31E-07	3.27E-02	2.94E-02	2.55E-02	8.71E-12	9.22E-09	2.84E-02	7.25E-10	1.30E-04	2.84E-02
91Y	1.64E-05	7.34E-01	6.60E-01	5.72E-01	1.94E-10	2.13E-07	6.36E-01	1.67E-08	1.19E-03	6.36E-01
95Zr	1.51E-05	6.68E-01	6.60E-01	5.28E-01	1.82E-10	1.91E-06	5.87E-01	1.40E-07	2.31E-03	5.87E-01
95Nb	4.29E-07	1.94E-01	1.74E-01	1.50E-02	5.03E-11	2.69E-08	1.66E-02	2.31E-09	2.09E-03	1.66E-02
103Ru	2.27E-06	1.60E-03	1.44E+01	7.92E+00	2.71E-09	4.96E-06	8.81E+00	3.49E-07	1.13E-03	8.81E+00
106Ru	1.77E-06	1.20E-03	1.08E-01	6.16E+00	2.13E-09	3.55E-06	6.85E+00	2.69E-07	4.35E-03	6.85E+00
106Rh	2.65E-06	1.20E-01	1.08E-03	9.24E-02	3.10E-11	3.37E-08	1.03E-01	2.63E-09	4.35E-03	1.03E-01
110Ag	6.43E-08	2.87E-03	2.58E-03	2.24E-03	7.55E-13	8.15E-10	2.50E-03	6.44E-11	1.47E-04	2.50E-03
123Sn	6.55E-08	2.94E-03	2.64E-03	2.29E-03	7.74E-13	8.15E-10	2.55E-03	6.44E-11	2.48E-06	2.55E-03
125Sb	8.19E-08	3.74E-03	3.36E-03	2.86E-03	9.68E-13	1.03E-09	3.18E-03	8.06E-11	6.48E-05	3.18E-03
127Te	1.02E-07	5.21E-03	4.68E-03	3.56E-03	1.36E-12	1.45E-09	3.96E-03	1.13E-10	8.47E-05	3.96E-03
129Te	6.55E-08	2.94E-03	2.64E-02	2.29E-03	7.74E-13	8.15E-10	2.55E-03	6.44E-11	2.48E-06	2.55E-03
134Cs	1.64E-06	7.34E-02	6.60E-01	5.72E-02	1.92E-11	2.02E-09	6.36E-02	1.58E-10	9.03E-05	6.36E-02
137Cs	2.27E-06	1.34E-01	1.20E-01	7.92E-02	2.52E-11	2.77E-08	8.81E-02	2.18E-09	3.73E-04	8.81E-02
141Ce	3.28E-06	1.47E-01	1.32E+00	1.14E-01	3.87E-11	4.25E-08	1.27E-01	3.22E-09	4.06E-04	1.27E-01
144Ce	3.78E-05	1.74E-02	1.56E+00	1.32E+00	4.45E-10	4.61E-07	1.47E+00	3.76E-08	4.40E-03	1.47E+00
144Pr	3.78E-05	1.74E-02	1.56E+00	1.32E+00	4.45E-10	4.61E-07	1.47E+00	3.76E-08	4.40E-03	1.47E+00
147Pm	4.16E-06	1.87E-01	1.68E-01	1.45E-01	5.03E-11	5.32E-08	1.62E-01	4.30E-09	9.60E-04	1.62E-01
148Pm	7.31E-06	1.94E-01	1.74E-01	2.55E-01	8.52E-13	9.22E-10	2.84E-01	2.69E-10	8.47E-05	2.84E-01
254Eu	6.81E-08	3.07E-03	2.76E-03	2.38E-03	0.00E+00	0.00E+00	2.64E-03	0.00E+00	2.60E-06	2.64E-03
234U	1.51E-14	8.01E-06	7.20E-06	5.72E-06	1.94E-10	3.55E-12	5.87E-10	1.34E-13	0.00E+00	5.87E-10
235U	2.52E-16	1.07E-07	9.60E-08	7.92E-08	2.71E-12	3.55E-12	9.79E-12	2.69E-15	0.00E+00	9.79E-12
236U	2.02E-15	1.00E-06	9.00E-07	7.50E-07	2.71E-11	3.55E-12	7.83E-11	1.88E-14	0.00E+00	7.83E-11
238U	6.18E-18	2.67E-19	2.40E-09	2.09E-09	7.16E-14	3.55E-12	2.40E-13	5.37E-17	0.00E+00	2.40E-13
237Np	1.01E-13	4.87E-07	4.38E-07	3.78E-07	5.23E-13	1.28E-09	3.92E-09	9.13E-13	2.48E-06	3.92E-09
238Pu	3.91E-08	1.80E-03	1.62E-03	1.36E-03	9.29E-14	9.93E-09	1.52E-03	7.25E-11	2.82E-02	1.52E-03
239Pu	3.40E-10	1.54E-05	1.38E-05	1.19E-05	8.13E-16	8.51E-11	1.32E-05	6.44E-12	7.90E-06	1.32E-05
240Pu	2.52E-10	1.14E-05	1.02E-05	8.80E-06	7.74E-16	6.83E-11	9.79E-06	4.57E-12	6.85E-06	9.79E-06
241Pu	1.15E-07	5.21E-03	4.68E-03	4.00E-03	2.71E-13	2.84E-08	4.45E-03	2.04E-09	5.36E-04	4.45E-03
242Pu	5.04E-13	2.27E-08	2.04E-08	1.76E-08	1.16E-12	3.55E-12	1.96E-08	8.06E-15	5.36E-07	1.96E-08
Total	1.45E-04	3.05E+00	2.21E+01	1.90E+01	6.69E-09	1.21E-05	2.11E+01	8.79E-07	5.61E-02	2.11E+01
Frequency, /hr	2.00E-09	8.80E-09	1.20E-08	1.30E-08	4.20E-09	1.10E-08	1.60E-08	1.60E-08	4.60E-08	2.00E-08

TABLE C-7. Curies of Specific Nuclides Released from 200-ft Stack due to Transfer Error to Sumps (H-Canyon)

Nuclide	Decon-tamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Np Cycle	HAW Evaporator	LAW Evaporator	Ion Exchange	High Heat Waste
89Sr	5.13E-08	2.32E-03	2.09E-03	1.79E-03	6.07E-13	1.34E-09	1.99E-03	5.13E-11	3.04E-06	1.99E-03
90Sr	3.58E-09	1.60E-04	1.44E-04	1.25E-04	4.27E-14	4.52E-11	1.39E-04	3.55E-12	6.36E-07	1.39E-04
90Y	3.58E-09	1.60E-04	1.44E-04	1.25E-04	4.27E-14	4.52E-11	1.39E-04	3.55E-12	6.36E-07	1.39E-04
91Y	8.03E-08	3.60E-03	3.24E-03	2.80E-03	9.48E-13	1.04E-09	3.12E-03	8.16E-11	5.81E-06	3.12E-03
95Zr	7.41E-08	3.27E-03	2.94E-03	2.59E-03	8.92E-13	9.38E-09	2.88E-03	6.84E-10	1.13E-05	2.88E-03
95Nb	2.10E-09	9.49E-04	8.53E-04	7.33E-05	2.47E-13	1.32E-10	8.15E-05	1.13E-11	1.02E-05	8.15E-05
103Ru	1.11E-08	7.85E-02	7.06E-02	3.88E-02	1.33E-11	2.43E-08	4.32E-02	1.71E-09	5.53E-06	4.32E-02
106Ru	8.65E-09	5.89E-02	5.29E-02	3.02E-02	1.04E-11	1.74E-08	3.36E-02	1.32E-09	2.13E-05	3.36E-02
106Rh	1.30E-09	5.89E-04	5.29E-04	4.53E-04	1.52E-13	1.65E-10	5.04E-04	1.29E-11	2.13E-05	5.04E-04
110Ag	3.15E-10	1.41E-05	1.27E-05	1.10E-05	3.70E-15	4.00E-12	1.22E-05	3.16E-13	7.19E-07	1.22E-05
123Sn	3.21E-10	1.44E-05	1.29E-05	1.12E-05	3.79E-15	4.00E-12	1.25E-05	3.16E-13	1.22E-08	1.25E-05
125Sb	4.02E-10	1.83E-05	1.65E-05	1.40E-05	4.74E-15	5.04E-12	1.56E-05	3.95E-13	3.32E-07	1.56E-05
127Te	5.00E-10	2.55E-05	2.29E-05	1.75E-05	6.64E-15	7.12E-12	1.94E-05	5.53E-13	4.15E-07	1.94E-05
129Te	3.21E-10	1.44E-05	1.29E-05	1.12E-05	3.79E-15	4.00E-12	1.25E-05	3.16E-13	1.22E-08	1.25E-05
134Cs	8.03E-09	3.60E-04	3.24E-04	2.80E-04	9.39E-14	9.99E-12	3.12E-04	7.76E-13	4.43E-07	3.12E-04
137Cs	1.11E-08	6.54E-04	5.88E-04	3.88E-04	1.23E-13	1.36E-10	4.32E-04	1.07E-11	1.83E-06	4.32E-04
141Ce	1.61E-08	7.20E-04	6.47E-04	5.61E-04	1.90E-13	2.08E-10	6.24E-04	1.58E-11	1.99E-06	6.24E-04
144Ce	1.85E-07	8.51E-03	7.65E-03	6.47E-03	2.18E-12	2.26E-09	7.19E-03	1.84E-10	2.16E-05	7.19E-03
144Pr	1.85E-07	8.51E-03	7.65E-03	6.47E-03	2.18E-12	2.26E-09	7.19E-03	1.84E-10	2.16E-05	7.19E-03
147Pm	2.04E-08	9.16E-04	8.24E-04	7.11E-04	2.47E-13	2.61E-10	7.91E-04	2.11E-11	4.70E-06	7.91E-04
148Pm	3.58E-08	9.49E-04	8.53E-04	1.25E-03	4.17E-15	4.52E-12	1.39E-03	1.32E-12	4.15E-07	1.39E-03
254Eu	3.34E-10	1.51E-05	1.35E-05	1.16E-05	0.00E+00	0.00E+00	1.30E-05	0.00E+00	1.27E-08	1.30E-05
234U	7.41E-17	3.93E-08	3.53E-08	2.80E-08	9.48E-13	1.74E-14	2.88E-12	6.58E-16	0.00E+00	2.88E-12
235U	1.24E-18	5.23E-10	4.71E-10	3.88E-20	1.33E-14	1.74E-14	4.80E-14	1.32E-17	0.00E+00	4.80E-14
236U	9.88E-18	4.58E-09	4.12E-09	3.67E-09	1.33E-13	1.74E-14	3.84E-13	9.21E-17	0.00E+00	3.84E-13
238U	3.03E-20	1.31E-11	1.18E-11	1.01E-11	3.51E-16	1.74E-14	9.59E-16	2.63E-19	0.00E+00	9.59E-16
237Np	4.94E-16	2.39E-09	2.15E-09	1.85E-09	2.56E-15	6.25E-12	1.92E-11	4.47E-15	1.22E-08	1.92E-11
238Pu	1.92E-10	8.83E-06	7.94E-06	6.68E-06	4.55E-16	4.86E-11	7.43E-06	3.55E-13	1.38E-04	7.43E-06
239Pu	1.67E-12	7.52E-08	6.76E-08	5.82E-08	3.98E-18	4.17E-13	6.47E-08	3.16E-14	3.87E-08	6.47E-08
240Pu	1.24E-12	5.56E-08	5.00E-08	4.31E-08	3.79E-18	3.13E-13	4.80E-08	2.24E-14	2.38E-08	4.80E-08
241Pu	5.62E-10	2.52E-05	2.27E-05	1.96E-05	1.33E-15	1.39E-10	2.18E-05	1.00E-11	2.63E-06	2.18E-05
242Pu	2.47E-15	1.11E-10	1.00E-10	8.62E-11	5.69E-21	1.74E-14	9.59E-11	3.95E-17	2.63E-19	9.59E-11
Total	7.01E-07	1.69E-01	1.52E-01	9.32E-02	3.28E-11	5.92E-08	1.04E-01	4.30E-09	2.75E-04	1.04E-01
Frequency, /hr	4.00E-07	1.60E-06	2.40E-06	2.50E-06	8.40E-07	2.20E-06	3.20E-06	3.20E-06	9.20E-06	4.00E-06

TABLE C-8. Curies of Specific Nuclides Released from 200-ft Stack due to Overflow to Sumps (H-Canyon)

Nuclide	Decon-tamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Np Cycle	HAW Evaporator	LAW Evaporator	Ion Exchange	High Heat Waste
89Sr	3.01E-08	1.36E-03	1.22E-03	1.05E-03	3.56E-13	7.84E-10	1.17E-03	3.01E-11	5.99E-06	1.17E-03
90Sr	2.10E-09	9.41E-05	8.43E-05	7.31E-05	2.50E-14	2.65E-11	8.17E-05	2.09E-12	1.25E-06	8.17E-05
90Y	2.10E-09	9.41E-05	8.43E-05	7.31E-05	2.50E-14	2.65E-11	8.17E-05	2.09E-12	1.25E-06	8.17E-05
91Y	4.71E-08	2.11E-03	1.89E-03	1.64E-03	5.56E-13	6.11E-10	1.83E-03	4.79E-11	1.14E-05	1.83E-03
95Zr	4.35E-08	1.92E-03	1.72E-03	1.51E-03	5.22E-13	5.50E-09	1.69E-03	4.02E-10	2.23E-05	1.69E-03
95Nb	1.23E-09	5.57E-04	4.99E-04	4.29E-05	1.45E-13	7.74E-11	4.79E-05	6.64E-12	2.01E-05	4.79E-05
103Ru	6.52E-09	4.61E-02	4.13E-02	2.27E-02	7.78E-12	1.43E-08	2.53E-02	1.00E-09	1.09E-05	2.53E-02
106Ru	5.07E-09	3.46E-02	3.10E-02	1.77E-02	6.11E-12	1.02E-08	1.97E-02	7.72E-10	4.19E-05	1.97E-02
106Rh	7.61E-09	3.46E-04	3.10E-04	2.65E-04	8.89E-14	9.67E-11	2.96E-04	7.57E-12	4.19E-05	2.96E-04
110Ag	1.85E-10	8.26E-06	7.40E-06	6.43E-06	2.17E-15	2.34E-12	7.18E-06	1.85E-13	1.42E-06	7.18E-06
123Sn	1.88E-10	8.45E-06	7.57E-06	6.56E-06	2.22E-15	2.34E-12	7.32E-06	1.85E-13	2.40E-08	7.32E-06
125Sb	2.36E-10	1.08E-05	9.63E-06	8.20E-06	2.78E-15	2.95E-12	9.15E-06	2.32E-13	6.53E-07	9.15E-06
127Te	2.94E-10	1.50E-05	1.34E-05	1.02E-05	3.89E-15	4.17E-12	1.14E-05	3.24E-13	8.17E-07	1.14E-05
129Te	1.88E-10	8.45E-06	7.57E-06	6.56E-06	2.22E-15	2.34E-12	7.32E-06	1.85E-13	2.40E-08	7.32E-06
134Cs	4.71E-09	2.11E-04	1.89E-04	1.64E-04	5.50E-14	5.80E-12	1.83E-04	4.56E-13	8.71E-07	1.83E-04
137Cs	6.52E-09	3.84E-04	3.44E-04	2.27E-04	7.23E-14	7.94E-11	2.53E-04	6.26E-12	3.59E-06	2.53E-04
141Ce	9.42E-09	4.23E-04	3.78E-04	3.28E-04	1.11E-13	1.22E-10	3.66E-04	9.27E-12	3.92E-06	3.66E-04
144Ce	1.09E-07	4.99E-03	4.47E-03	3.78E-03	1.28E-12	1.32E-09	4.22E-03	1.08E-10	4.25E-05	4.22E-03
144Pr	1.09E-07	4.99E-03	4.47E-03	3.78E-03	1.28E-12	1.32E-09	4.22E-03	1.08E-10	4.25E-05	4.22E-03
147Pm	1.20E-08	5.38E-04	4.82E-04	4.16E-04	1.45E-13	1.53E-10	4.65E-04	1.24E-11	9.26E-06	4.65E-04
148Pm	2.10E-08	5.57E-04	4.99E-04	7.31E-04	2.45E-15	2.65E-12	8.17E-04	7.72E-13	8.17E-07	8.17E-04
254Eu	1.96E-10	8.84E-06	7.91E-06	6.81E-06	0.00E+00	0.00E+00	7.60E-06	0.00E+00	2.50E-08	7.60E-06
234U	4.35E-17	2.31E-08	2.06E-08	1.64E-08	5.56E-13	1.02E-14	1.69E-12	3.86E-16	0.00E+00	1.69E-12
235U	7.25E-19	3.07E-10	2.75E-10	2.27E-10	7.78E-15	1.02E-14	2.82E-14	7.72E-18	0.00E+00	2.82E-14
236U	5.80E-18	2.69E-09	2.41E-09	2.14E-09	7.78E-14	1.02E-14	2.25E-13	5.41E-17	0.00E+00	2.25E-13
238U	1.78E-20	7.68E-12	6.88E-12	5.93E-12	2.06E-16	1.02E-14	5.63E-16	1.55E-19	0.00E+00	5.63E-16
237Np	2.90E-16	1.40E-09	1.26E-09	1.08E-09	1.50E-15	3.66E-12	1.13E-11	2.63E-15	2.40E-08	1.13E-11
238Pu	1.12E-10	5.19E-06	4.64E-06	3.91E-06	2.67E-16	2.85E-11	4.36E-06	2.09E-13	2.72E-04	4.36E-06
239Pu	9.78E-13	4.42E-08	3.96E-08	3.40E-08	2.33E-18	2.44E-13	3.80E-08	1.85E-14	7.62E-08	3.80E-08
240Pu	7.25E-13	3.27E-08	2.92E-08	2.52E-08	2.22E-18	1.83E-13	2.82E-08	1.31E-14	4.68E-08	2.82E-08
241Pu	3.30E-10	1.48E-05	1.32E-05	1.15E-05	7.78E-16	8.12E-11	1.28E-05	5.87E-12	5.17E-06	1.28E-05
242Pu	1.45E-15	6.53E-11	5.85E-11	5.04E-11	3.33E-21	1.02E-14	5.63E-11	2.32E-17	5.17E-19	5.63E-11
Total	4.18E-07	9.93E-02	8.90E-02	5.45E-02	1.92E-11	3.47E-08	6.08E-02	2.53E-09	5.41E-04	6.08E-02
Frequency, /hr	5.00E-07	4.00E-05	1.00E-05	3.80E-05	3.40E-06	6.20E-06	4.50E-05	4.50E-05	1.00E-04	6.00E-05

TABLE C-9. Curies of Specific Nuclides Released from 200-ft Stack due to Leaks to Sumps (H-Canyon)

Nuclide	Decon-tamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Np Cycle	HAW Evaporator	LAW Evaporator	Ion Exchange	High Heat Waste
89Sr	1.01E-08	1.83E-04	1.71E-04	1.47E-04	5.01E-14	1.10E-10	1.66E-04	4.27E-12	2.86E-06	1.66E-04
90Sr	7.04E-10	1.33E-05	1.18E-05	1.03E-05	3.52E-15	3.73E-12	1.16E-05	2.95E-13	5.98E-07	1.16E-05
90Y	7.04E-10	1.33E-05	1.18E-05	1.03E-05	3.52E-15	3.73E-12	1.16E-05	2.95E-13	5.98E-07	1.16E-05
91Y	1.58E-08	2.99E-04	2.65E-04	2.30E-04	7.83E-14	8.60E-11	2.59E-04	6.78E-12	5.46E-06	2.59E-04
95Zr	1.46E-08	2.72E-04	2.41E-04	2.12E-04	7.36E-14	7.74E-10	2.39E-04	5.69E-11	1.07E-05	2.39E-04
95Nb	4.12E-10	7.89E-05	6.99E-05	6.01E-06	2.04E-14	1.09E-11	6.78E-06	9.41E-13	9.61E-06	6.78E-06
103Ru	2.18E-09	6.53E-03	5.79E-03	3.18E-03	1.10E-12	2.01E-09	3.59E-03	1.42E-10	5.20E-06	3.59E-03
106Ru	1.70E-09	4.90E-03	4.34E-03	2.47E-03	8.61E-13	1.43E-09	2.79E-03	1.09E-10	2.00E-05	2.79E-03
106Rh	2.55E-09	4.90E-05	4.34E-05	3.71E-05	1.25E-14	1.36E-11	4.19E-05	1.07E-12	2.00E-05	4.19E-05
110Ag	6.19E-11	1.17E-06	1.04E-06	9.01E-07	3.05E-16	3.30E-13	1.02E-06	2.63E-14	6.75E-07	1.02E-06
123Sn	6.31E-11	1.20E-06	1.06E-06	9.19E-07	3.13E-16	3.30E-13	1.04E-06	2.63E-14	1.14E-08	1.04E-06
125Sb	7.89E-11	1.52E-06	1.35E-06	1.15E-06	3.92E-16	4.16E-13	1.30E-06	3.28E-14	3.12E-07	1.30E-06
127Te	9.83E-11	2.12E-06	1.88E-06	1.43E-06	5.48E-16	5.88E-13	1.62E-06	4.60E-14	3.90E-07	1.62E-06
129Tm	6.31E-11	1.20E-06	1.06E-06	9.19E-07	3.13E-16	3.30E-13	1.04E-06	2.63E-14	1.14E-08	1.04E-06
134Cs	1.58E-09	2.99E-05	2.65E-05	2.38E-05	7.75E-15	8.17E-13	2.59E-05	6.46E-14	4.16E-07	2.59E-05
137Cs	2.18E-09	5.44E-05	4.82E-05	3.18E-05	1.02E-14	1.12E-11	3.59E-05	8.86E-13	1.71E-06	3.59E-05
141Ce	3.15E-09	5.98E-05	5.30E-05	4.60E-05	1.57E-14	1.72E-11	5.18E-05	1.31E-12	1.87E-06	5.18E-05
144Ce	3.64E-08	7.07E-04	6.27E-04	5.30E-04	1.80E-13	1.86E-10	5.98E-04	1.53E-11	2.03E-05	5.98E-04
144Pr	3.64E-08	7.07E-04	6.27E-04	5.30E-04	1.80E-13	1.86E-10	5.98E-04	1.53E-11	2.03E-05	5.98E-04
147Pm	4.00E-09	7.62E-05	6.75E-05	5.83E-05	2.04E-14	2.15E-11	6.58E-05	1.75E-12	4.42E-06	6.58E-05
148Pm	7.04E-09	7.89E-05	6.99E-05	1.03E-04	3.45E-16	3.73E-13	1.16E-04	1.09E-13	3.90E-07	1.16E-04
154Eu	6.55E-11	1.25E-06	1.11E-06	9.54E-07	0.00E+00	0.00E+00	1.08E-06	0.00E+00	1.20E-08	1.08E-06
234U	1.46E-17	3.26E-09	2.89E-09	2.30E-09	7.83E-14	1.43E-15	2.39E-13	5.47E-17	2.60E-16	2.39E-13
235U	2.43E-19	4.33E-11	3.86E-11	3.18E-11	1.10E-15	1.43E-15	3.99E-15	1.09E-18	2.60E-16	3.99E-15
236U	1.94E-18	3.81E-10	3.38E-10	3.00E-10	1.10E-14	1.43E-15	3.19E-14	7.66E-18	2.60E-16	3.19E-14
238U	5.94E-21	1.09E-12	9.64E-13	8.31E-13	2.90E-17	1.43E-15	7.98E-17	2.19E-20	2.60E-16	7.98E-17
237Np	9.74E-17	1.99E-10	1.76E-10	1.52E-10	2.11E-16	5.16E-13	1.60E-12	3.72E-16	1.14E-08	1.60E-12
238Pu	1.76E-11	7.34E-07	6.51E-07	5.48E-07	3.76E-17	4.02E-12	6.18E-07	2.95E-14	1.30E-04	6.18E-07
239Pu	3.28E-13	6.26E-09	5.55E-09	4.77E-09	3.29E-19	3.44E-14	5.38E-09	2.63E-15	3.64E-08	5.38E-09
240Pu	2.43E-13	4.62E-09	4.10E-09	3.53E-09	3.13E-19	2.58E-14	3.99E-09	1.86E-15	2.23E-08	3.99E-09
241Pu	1.10E-10	2.09E-06	1.86E-06	1.61E-06	1.10E-16	1.15E-11	1.81E-06	8.31E-13	2.47E-06	1.81E-06
242Pu	4.85E-16	9.25E-12	8.20E-12	7.07E-12	4.70E-22	1.43E-15	7.98E-12	3.28E-18	2.47E-09	7.98E-12
Total	1.40E-07	1.41E-02	1.25E-02	7.64E-03	2.71E-12	4.89E-09	8.62E-03	3.58E-10	2.58E-04	8.62E-03
Frequency, /hr	4.80E-04	6.20E-05	5.00E-04	7.20E-04	3.60E-04	1.20E-03	1.80E-03	1.80E-03	1.20E-03	7.00E-05

TABLE C-10. Curies of Specific Nuclides Released to Retention Basin due to Coil Failures (H-Canyon)

Nuclide	Decontamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Np Cycle	HAW Evaporator	LAW Evaporator	High Heat Waste
89Sr	2.20E-02	1.21E+00	1.21E+00	1.41E+00	4.94E-10	9.03E-07	1.41E+00	5.93E-08	1.41E+00
90Sr	4.60E-03	8.30E-02	8.30E-02	9.86E-02	3.45E-11	6.31E-08	9.86E-02	4.14E-09	9.86E-02
90Y	0.00E+00	8.30E-02	8.30E-02	9.86E-02	3.45E-11	6.31E-08	9.86E-02	4.14E-09	9.86E-02
91Y	4.00E-02	1.87E+00	1.87E+00	2.21E+00	7.74E-10	1.41E-06	2.21E+00	9.28E-08	2.21E+00
95Zr	5.40E-03	1.70E+00	1.70E+00	2.04E+00	7.14E-10	1.31E-06	2.04E+00	8.57E-08	2.04E+00
95Nb	1.02E-02	4.93E-01	4.93E-01	5.78E-02	2.02E-11	3.70E-08	5.78E-02	2.43E-09	5.78E-02
103Ru	7.80E-03	4.08E-01	4.08E-01	3.06E-01	1.07E-10	1.96E-07	3.06E-01	1.29E-08	3.06E-01
106Ru	5.80E-03	3.06E-01	3.06E-01	2.38E-01	8.33E-11	1.52E-07	2.38E-01	1.00E-08	2.38E-01
106Rh	0.00E+00	3.06E-01	3.06E-01	3.57E-01	1.25E-10	2.28E-07	3.57E-01	1.50E-08	3.57E-01
110Ag	0.00E+00	7.31E-03	7.31E-03	8.67E-03	3.03E-12	5.55E-09	8.67E-03	3.64E-10	8.67E-03
123Sn	0.00E+00	7.48E-03	7.48E-03	8.84E-03	3.09E-12	5.66E-09	8.84E-03	3.71E-10	8.84E-03
125Sb	0.00E+00	9.52E-03	9.52E-03	1.11E-02	3.87E-12	7.07E-09	1.11E-02	4.64E-10	1.11E-02
127Te	0.00E+00	4.76E-03	4.76E-03	1.34E-02	4.69E-12	8.58E-09	1.34E-02	5.63E-10	1.34E-02
129Te	0.00E+00	7.48E-03	7.48E-03	8.84E-03	3.09E-12	5.66E-09	8.84E-03	3.71E-10	8.84E-03
134Cs	6.40E-03	1.87E-01	1.87E-01	2.21E-01	7.74E-11	1.41E-07	2.21E-01	9.28E-09	2.21E-01
137Cs	5.00E-03	3.40E-01	3.40E-01	3.06E-01	1.07E-10	1.96E-07	3.06E-01	1.29E-08	3.06E-01
141Ce	0.00E+00	3.74E-01	3.74E-01	4.42E-01	1.55E-10	2.83E-07	4.42E-01	1.86E-08	4.42E-01
144Ce	8.40E-02	4.42E+00	4.42E+00	5.10E+00	1.79E-09	3.26E-06	5.10E+00	2.14E-07	5.10E+00
144Pr	0.00E+00	4.42E+00	4.42E+00	5.10E+00	1.79E-09	3.26E-06	5.10E+00	2.14E-07	5.10E+00
147Pm	8.20E-03	4.76E-01	4.76E-01	5.61E-01	1.96E-10	3.59E-07	5.61E-01	2.36E-08	5.61E-01
148Pm	0.00E+00	4.93E-01	4.93E-01	9.86E-01	3.45E-10	6.31E-07	9.86E-01	4.14E-08	9.86E-01
154Eu	0.00E+00	7.82E-03	7.82E-03	9.18E-03	3.21E-12	5.88E-09	9.18E-03	3.86E-10	9.18E-03
234U	6.00E-11	2.04E-06	2.04E-06	2.21E-05	7.74E-15	1.41E-11	2.04E-09	8.57E-17	2.04E-09
235U	6.00E-13	2.72E-07	2.72E-07	3.06E-07	1.07E-16	1.96E-13	3.40E-11	1.43E-18	3.40E-11
236U	8.00E-12	2.38E-06	2.38E-06	2.89E-06	1.01E-15	1.85E-12	2.72E-10	1.14E-17	2.72E-10
238U	2.00E-14	6.80E-09	6.80E-09	7.99E-09	2.80E-18	5.11E-15	6.80E-13	2.86E-20	6.80E-13
237Np	0.00E+00	1.24E-06	1.24E-06	1.46E-06	5.11E-16	9.34E-13	1.36E-09	5.71E-17	1.36E-09
238Pu	1.62E-04	4.59E-03	4.59E-03	5.27E-03	1.84E-12	3.37E-09	5.27E-03	2.21E-10	5.27E-03
239Pu	1.74E-06	3.91E-05	3.91E-05	4.59E-05	1.61E-14	2.94E-11	4.59E-05	1.93E-12	4.59E-05
240Pu	1.40E-06	2.89E-05	2.89E-05	3.40E-05	1.19E-14	2.18E-11	3.40E-05	1.43E-12	3.40E-05
241Pu	3.20E-07	1.31E-02	1.31E-02	1.55E-02	5.43E-12	9.92E-09	1.55E-02	6.51E-10	1.55E-02
242Pu	0.00E+00	5.78E-08	5.78E-08	6.80E-08	2.38E-17	4.35E-14	6.80E-08	2.86E-15	6.80E-08
Total	2.00E-01	1.72E+01	1.72E+01	1.96E+01	6.86E-09	1.26E-05	1.96E+01	8.24E-07	1.96E+01
Frequency, /hr	1.50E-06	2.60E-05	1.60E-05	2.40E-06	2.40E-06	2.40E-06	3.40E-05	3.40E-05	2.40E-06

TABLE C-11. Curies of Specific Nuclides Released to Four Mile Creek due to Coil Failures (H-Canyon)

Nuclide	Decontamination	Dissolver	Head End	First Cycle	2nd U Cycle	2nd Np Cycle	HAW Evaporator	LAW Evaporator	High Heat Waste
89Sr	2.20E-02	1.21E+00	1.21E+00	1.41E+00	4.94E-10	9.03E-07	1.41E+00	5.93E-08	1.41E+00
90Sr	4.60E-03	8.30E-02	8.30E-02	9.86E-02	3.45E-11	6.31E-08	9.86E-02	4.14E-09	9.86E-02
90Y	0.00E+00	8.30E-02	8.30E-02	9.86E-02	3.45E-11	6.31E-08	9.86E-02	4.14E-09	9.86E-02
91Y	4.00E-02	1.87E+00	1.87E+00	2.21E+00	7.74E-10	1.41E-06	2.21E+00	9.28E-08	2.21E+00
95Zr	5.40E-03	1.70E+00	1.70E+00	2.04E+00	7.14E-10	1.31E-06	2.04E+00	8.57E-08	2.04E+00
95Nb	1.02E-02	4.93E-01	4.93E-01	5.78E-02	2.02E-11	3.70E-08	5.78E-02	2.43E-08	5.78E-02
103Ru	7.80E-03	4.08E-01	4.08E-01	3.06E-01	1.07E-10	1.96E-07	3.06E-01	1.29E-08	3.06E-01
106Ru	5.80E-03	3.06E-01	3.06E-01	2.38E-01	8.33E-11	1.52E-07	2.38E-01	1.00E-08	2.38E-01
106Rh	0.00E+00	3.06E-01	3.06E-01	3.57E-01	1.25E-10	2.28E-07	3.57E-01	1.50E-08	3.57E-01
110Ag	0.00E+00	7.31E-03	7.31E-03	8.67E-03	3.03E-12	5.55E-09	8.67E-03	3.64E-10	8.57E-03
123Sn	0.00E+00	7.48E-03	7.48E-03	8.84E-03	3.09E-12	5.66E-09	8.84E-03	3.71E-10	8.34E-03
125Sb	0.00E+00	9.52E-03	9.52E-03	1.11E-02	3.87E-12	7.07E-09	1.11E-02	4.64E-10	1.11E-02
127Te	0.00E+00	4.76E-03	4.76E-03	1.34E-02	4.69E-12	8.58E-09	1.34E-02	5.63E-10	1.34E-02
129Te	0.00E+00	7.48E-03	7.48E-03	8.84E-03	3.09E-12	5.66E-09	8.84E-03	3.71E-10	8.84E-03
134Cs	6.40E-03	1.87E-01	1.87E-01	2.21E-01	7.74E-11	1.41E-07	2.21E-01	9.28E-09	2.21E-01
137Cs	5.00E-03	3.40E-01	3.40E-01	3.06E-01	1.07E-10	1.96E-07	3.06E-01	1.29E-08	3.06E-01
141Ce	0.00E+00	3.74E-01	3.74E-01	4.42E-01	1.55E-10	2.83E-07	4.42E-01	1.86E-08	4.42E-01
144Ce	8.40E-02	4.42E+00	4.42E+00	5.10E+00	1.79E-09	3.26E-06	5.10E+00	2.14E-07	5.10E+00
144Pr	0.00E+00	4.42E+00	4.42E+00	5.10E+00	1.79E-09	3.26E-06	5.10E+00	2.14E-07	5.10E+00
147Pm	8.20E-03	4.76E-01	4.76E-01	5.61E-01	1.96E-10	3.59E-07	5.61E-01	2.36E-08	5.61E-01
148Pm	0.00E+00	4.93E-01	4.93E-01	9.86E-01	3.45E-10	6.31E-07	9.86E-01	4.14E-08	9.86E-01
154Eu	0.00E+00	7.82E-03	7.82E-03	9.18E-03	3.21E-12	5.88E-09	9.18E-03	3.86E-10	9.18E-03
234U	6.00E-11	2.04E-06	2.04E-06	2.21E-05	7.74E-15	1.41E-11	2.04E-09	8.57E-17	2.04E-09
235U	6.00E-13	2.72E-07	2.72E-07	3.06E-07	1.07E-16	1.96E-13	3.40E-11	1.43E-18	3.40E-11
236U	8.00E-12	2.38E-06	2.38E-06	2.89E-06	1.01E-15	1.85E-12	2.72E-10	1.14E-17	2.72E-10
238U	2.00E-14	6.80E-09	6.80E-09	7.99E-09	2.80E-18	5.11E-15	6.80E-13	2.86E-20	6.80E-13
237Np	0.00E+00	1.24E-06	1.24E-06	1.46E-06	5.11E-16	9.34E-13	1.36E-09	5.71E-17	1.36E-09
238Pu	1.62E-04	4.59E-03	4.59E-03	5.27E-03	1.84E-12	3.37E-09	5.27E-03	2.21E-10	5.27E-03
239Pu	1.74E-06	3.91E-05	3.91E-05	4.59E-05	1.61E-14	2.94E-11	4.59E-05	1.93E-12	4.59E-05
240Pu	1.40E-06	2.89E-05	2.89E-05	3.40E-05	1.19E-14	2.18E-11	3.40E-05	1.43E-12	3.40E-05
241Pu	3.20E-07	1.31E-02	1.31E-02	1.55E-02	5.43E-12	9.92E-09	1.55E-02	6.51E-10	1.55E-02
242Pu	0.00E+00	5.78E-08	5.78E-08	6.80E-08	2.38E-17	4.35E-14	6.80E-08	2.86E-15	6.80E-08
Total	2.00E-01	1.72E+01	1.72E+01	1.96E+01	6.86E-09	1.26E-05	1.96E+01	8.24E-07	1.96E+01
Frequency, /hr	6.00E-08	1.04E-06	6.40E-07	9.60E-08	9.60E-08	9.60E-08	1.36E-06	1.36E-06	9.60E-08

APPENDIX D

H-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/YR			DOSE, PERSON-REM				
DISSOLVER	3.43E-02	2.00E-04	5.96E-06	2.98E-02	3.72E-02	9.04E-01	1.44E-01	1.15E-01	3.76E-05	4.75E-01
HEAD END	3.11E-02	2.00E-04	5.40E-06	2.70E-02	3.37E-02	8.19E-01	1.31E-01	1.04E-01	3.41E-05	4.30E-01
FIRST CYCLE	2.60E-02	2.00E-04	4.54E-06	2.27E-02	2.87E-02	6.92E-01	1.10E-01	8.75E-02	2.70E-05	3.58E-01
2ND U CYCLE	5.84E-12	2.00E-04	4.40E-15	2.20E-11	6.19E-12	3.90E-10	1.65E-11	8.70E-11	5.31E-15	1.34E-09
2ND Np CYCLE	4.59E-09	2.00E-04	6.08E-12	3.04E-08	3.65E-09	1.08E-06	1.28E-07	1.36E-07	1.37E-11	1.30E-07
HAW EVAPORATOR	2.86E-02	2.00E-04	5.02E-06	2.51E-02	3.19E-02	7.63E-01	1.22E-01	9.65E-02	3.40E-05	3.96E-01
LAW EVAPORATOR	1.36E-09	2.00E-04	3.06E-13	1.53E-09	1.13E-09	5.49E-08	6.10E-09	6.21E-09	4.06E-12	1.64E-08
ION EXCHANGE	4.52E-05	2.00E-04	9.88E-06	4.94E-02	4.45E-05	1.83E+00	2.28E-01	2.06E-01	2.42E-08	1.64E-01
HIGH HEAT WASTE	2.86E-02	2.00E-04	5.02E-06	2.51E-02	3.19E-02	7.63E-01	1.22E-01	9.65E-02	3.40E-05	3.96E-01
TOTAL			3.58E-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			DOSE, MREM				
DISSOLVER	3.43E-02	2.00E-04	1.31E-06	6.57E-03	4.34E-03	1.94E-01	3.08E-02	2.20E-02	7.18E-06	1.30E-01
HEAD END	3.11E-02	2.00E-04	1.19E-06	5.96E-03	3.93E-03	1.76E-01	2.79E-02	1.99E-02	6.52E-06	1.17E-01
FIRST CYCLE	2.60E-02	2.00E-04	1.01E-06	5.03E-03	3.36E-03	1.49E-01	2.35E-02	1.68E-02	5.19E-06	9.79E-02
2ND U CYCLE	5.84E-12	2.00E-04	1.33E-15	6.67E-12	7.20E-13	1.16E-10	3.74E-12	2.02E-11	1.01E-15	3.68E-10
2ND Np CYCLE	4.59E-09	2.00E-04	1.14E-12	5.69E-09	4.12E-10	2.03E-07	2.25E-08	2.22E-08	2.57E-12	3.52E-08
HAW EVAPORATOR	2.86E-02	2.00E-04	1.11E-06	5.55E-03	3.73E-03	1.64E-01	2.59E-02	1.85E-02	6.52E-06	1.08E-01
LAW EVAPORATOR	1.36E-09	2.00E-04	6.54E-14	3.27E-10	1.29E-10	1.14E-08	1.20E-09	1.13E-09	7.63E-13	4.35E-09
ION EXCHANGE	4.52E-05	2.00E-04	1.80E-06	8.98E-03	5.17E-06	3.37E-01	3.95E-02	3.32E-02	4.59E-09	4.51E-02
HIGH HEAT WASTE	2.86E-02	2.00E-04	1.11E-06	5.55E-03	3.73E-03	1.64E-01	2.59E-02	1.85E-02	6.52E-06	1.08E-01
TOTAL			7.53E-06							

TABLE D-3. Dose and Risk to the 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	3.43E-02	2.00E-04	2.16E-06	1.08E-02	1.59E-02	3.32E-01	5.26E-02	4.32E-02	6.31E-06	1.48E-01
HEAD END	3.11E-02	2.00E-04	1.97E-06	9.83E-03	1.44E-02	3.01E-01	4.77E-02	3.92E-02	5.72E-06	1.34E-01
FIRST CYCLE	2.60E-02	2.00E-04	1.65E-06	8.24E-03	1.23E-02	2.54E-01	4.01E-02	3.30E-02	4.57E-06	1.11E-01
2ND U CYCLE	5.84E-12	2.00E-04	1.36E-15	6.79E-12	2.64E-12	1.22E-10	5.86E-12	3.03E-11	8.90E-16	4.13E-10
2ND Np CYCLE	4.59E-09	2.00E-04	2.34E-12	1.17E-08	1.57E-09	4.14E-07	4.97E-08	5.39E-08	2.22E-12	4.06E-08
HAW EVAPORATOR	2.86E-02	2.00E-04	1.82E-06	9.08E-03	1.37E-02	2.80E-01	4.43E-02	3.64E-02	5.72E-06	1.23E-01
LAW EVAPORATOR	1.36E-09	2.00E-04	1.13E-13	5.63E-10	4.88E-10	2.04E-08	2.29E-09	2.39E-09	6.58E-13	5.21E-09
ION EXCHANGE	4.52E-05	2.00E-04	3.84E-06	1.92E-02	1.89E-05	7.05E-01	8.96E-02	8.22E-02	4.00E-09	5.05E-02
HIGH HEAT WASTE	2.86E-02	2.00E-04	1.82E-06	9.08E-03	1.37E-02	2.80E-01	4.43E-02	3.64E-02	5.72E-06	1.23E-01
TOTAL			1.32E-05							

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	1.72E+00	8.76E-03	2.13E-03	2.43E-01	1.81E+00	6.84E+00	1.07E+00	1.11E+00	2.49E-03	2.34E+01
HEAD END	1.56E+00	8.76E-03	1.93E-03	2.20E-01	1.64E+00	6.21E+00	9.71E-01	1.01E+00	2.26E-03	2.12E+01
FIRST CYCLE	9.56E-01	8.76E-03	1.58E-03	1.80E-01	1.00E+00	5.20E+00	8.18E-01	7.86E-01	1.30E-03	1.29E+01
2ND U CYCLE	2.02E-10	8.76E-03	1.45E-12	1.65E-10	2.09E-10	2.91E-09	1.22E-10	6.73E-10	2.66E-13	1.21E-08
2ND Np CYCLE	1.49E-07	8.76E-03	1.98E-09	2.26E-07	1.39E-07	7.97E-06	9.41E-07	1.02E-06	2.71E-10	2.41E-06
HAW EVAPORATOR	1.05E+00	8.76E-03	1.73E-03	1.98E-01	1.10E+00	5.74E+00	9.01E-01	8.66E-01	1.44E-03	1.42E+01
LAW EVAPORATOR	[illegible]	8.76E-03	1.04E-10	1.19E-08	4.22E-08	4.10E-07	4.52E-08	5.20E-08	7.84E-11	5.06E-07
ION EXCHANGE	4.92E-04	8.76E-03	3.21E-03	3.66E-01	4.17E-04	1.35E+01	1.69E+00	1.52E+00	2.08E-07	1.22E+00
HIGH HEAT WASTE	1.05E+00	8.76E-03	1.73E-03	1.98E-01	1.10E+00	5.74E+00	9.01E-01	8.66E-01	1.44E-03	1.42E+01
TOTAL			1.23E-02							

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	BONE	LIVER	KIDNEY	THYROID	LUNG
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR			DOSE, MREM				
DISSOLVER	1.72E+00	8.76E-03	2.79E-04	3.18E-02	1.18E-01	8.48E-01	1.30E-01	1.32E-01	2.84E-04	3.63E+00
HEAD END	1.56E+00	8.76E-03	2.52E-04	2.88E-02	1.07E-01	7.69E-01	1.18E-01	1.20E-01	2.58E-04	3.30E+00
FIRST CYCLE	9.56E-01	8.76E-03	2.04E-04	2.33E-02	6.56E-02	6.43E-01	9.95E-02	9.13E-02	1.49E-04	2.01E+00
2ND U CYCLE	2.02E-10	8.76E-03	2.51E-13	2.87E-11	1.37E-11	4.92E-10	1.58E-11	8.97E-11	3.03E-14	1.89E-09
2ND Np CYCLE	1.49E-07	8.76E-03	2.13E-10	2.43E-08	9.04E-09	8.59E-07	9.48E-08	9.63E-08	3.08E-11	3.74E-07
HAW EVAPORATOR	1.05E+00	8.76E-03	2.25E-04	2.57E-02	7.22E-02	7.08E-01	1.10E-01	1.01E-01	1.64E-04	2.21E+00
LAW EVAPORATOR	4.37E-08	8.76E-03	1.29E-11	1.47E-09	2.74E-09	4.91E-08	5.10E-09	5.66E-09	8.90E-12	8.74E-08
ION EXCHANGE	4.92E-04	8.76E-03	3.33E-04	3.80E-02	2.76E-05	1.42E+00	1.67E-01	1.40E-01	2.36E-08	1.91E-01
HIGH HEAT WASTE	1.05E+00	8.76E-03	2.25E-04	2.57E-02	7.22E-02	7.08E-01	1.10E-01	1.01E-01	1.64E-04	2.21E+00
TOTAL			1.52E-03							

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
ONSITE POPULATION			PERSON-REM/YR			DOSE, PERSON-REM				
DISSOLVER	1.72E+00	8.76E-03	3.70E-04	4.22E-02	3.71E-01	1.20E+00	1.88E-01	1.97E-01	6.95E-04	3.50E+00
HEAD END	1.56E+00	8.76E-03	3.35E-04	3.82E-02	3.37E-01	1.09E+00	1.71E-01	1.79E-01	6.31E-04	3.18E+00
FIRST CYCLE	9.56E-01	8.76E-03	2.73E-04	3.12E-02	2.06E-01	9.17E-01	1.44E-01	1.40E-01	3.64E-04	1.94E+00
2ND U CYCLE	2.02E-10	8.76E-03	2.15E-13	2.46E-11	4.30E-11	4.39E-10	2.10E-11	1.13E-10	7.42E-14	1.80E-09
2ND Np CYCLE	1.49E-07	8.76E-03	3.67E-10	4.19E-08	2.87E-08	1.47E-06	1.77E-07	1.95E-07	7.54E-11	3.62E-07
HAW EVAPORATOR	1.05E+00	8.76E-03	3.01E-04	3.44E-02	2.27E-01	1.01E+00	1.58E-01	1.54E-01	4.02E-04	2.13E+00
LAW EVAPORATOR	4.37E-08	8.76E-03	1.83E-11	2.09E-09	8.69E-09	7.33E-08	8.19E-09	9.50E-09	2.18E-11	8.51E-08
ION EXCHANGE	4.92E-04	8.76E-03	5.99E-04	6.84E-02	8.53E-05	2.52E+00	3.20E-01	2.93E-01	5.79E-08	1.81E-01
HIGH HEAT WASTE	1.05E+00	8.76E-03	3.01E-04	3.44E-02	2.27E-01	1.01E+00	1.58E-01	1.54E-01	4.02E-04	2.13E+00
TOTAL	TOTAL		2.18E-03							

TABLE D-7. Dose and Risk to the Offsite Population from Stack Release due to Fire

UNIT OPERATION	TOTAL Ci RELEASED	FREQUENCY, /YR	RISK
OFFSITE POPULATION			PERSON-REM/YR
FIRST CYCLE	5.77E-01	6.13E-04	2.34E-04
2ND U CYCLE	9.42E-06	6.13E-04	1.65E-08
2ND Np CYCLE	5.13E-04	6.13E-04	1.29E-06
ION EXCHANGE	2.90E-02	3.56E-02	7.60E-01
TOTAL			7.68E-01

UNIT OPERATION	TOTAL BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
	DOSE, PERSON-REM						
FIRST CYCLE	3.81E-01	4.83E-01	1.16E+01	1.85E+00	1.47E+00	4.56E-04	6.01E+00
2ND U CYCLE	2.69E-05	7.56E-06	4.78E-04	2.02E-05	1.07E-04	6.33E-09	1.64E-03
2ND Np CYCLE	2.10E-03	3.76E-04	7.74E-02	9.46E-03	8.72E-03	1.07E-06	1.12E-02
ION EXCHANGE	2.16E+01	2.46E-02	7.99E+02	1.00E+02	9.01E+01	1.23E-05	7.18E+01

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			DOSE, MREM				
FIRST CYCLE	5.77E-01	6.13E-04	2.95E-05	4.81E-02	3.22E-02	1.42E+00	2.25E-01	1.60E-01	5.24E-05	9.36E-01
2ND U CYCLE	9.42E-06	6.13E-04	2.86E-09	4.66E-06	5.01E-07	8.08E-05	2.61E-06	1.41E-05	7.23E-10	2.57E-04
2ND Np CYCLE	5.13E-04	6.13E-04	1.38E-07	2.25E-04	2.44E-05	8.34E-03	9.53E-04	8.22E-04	1.20E-07	1.73E-03
ION EXCHANGE	2.90E-02	3.56E-02	7.97E-02	2.24E+00	1.63E-03	8.42E+01	9.86E+00	8.29E+00	1.39E-06	1.13E+01
TOTAL			7.97E-02							

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
ONSITE POPULATION			PERSON-REM/YR			DOSE, PERSON-REM				
FIRST CYCLE	5.77E-01	6.13E-04	4.08E-05	6.66E-02	9.92E-02	2.05E+00	3.24E-01	2.67E-01	1.28E-04	8.99E-01
2ND U CYCLE	9.42E-06	6.13E-04	2.46E-09	4.01E-06	1.55E-06	7.23E-05	6.46E-06	1.79E-05	1.77E-09	2.43E-04
2ND Np CYCLE	5.13E-04	6.13E-04	2.39E-07	3.89E-04	7.76E-05	1.43E-02	1.78E-03	1.67E-03	2.95E-07	1.68E-03
ION EXCHANGE	2.90E-02	3.56E-02	1.44E-01	4.04E+00	5.04E-03	1.49E+02	1.89E+01	1.73E+01	3.42E-06	1.07E+01
TOTAL			1.44E-01							

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	1.60E-01	7.88E-01	1.31E-02	1.66E-02	1.63E-01	3.44E-01	7.55E-02	7.25E-02	2.18E-04	2.05E+00
HEAD END	1.45E-01	1.75E-02	5.62E-04	3.21E-02	1.47E-01	9.35E-01	1.48E-01	1.36E-01	1.94E-04	1.91E+00
FIRST CYCLE	9.75E-02	8.76E-02	1.19E-03	1.36E-02	9.76E-02	2.95E-01	6.50E-02	5.50E-02	1.21E-04	1.20E+00
2ND U CYCLE	3.94E-07	8.76E-02	9.20E-08	1.05E-06	3.17E-07	1.88E-05	8.59E-07	4.21E-06	2.76E-10	6.24E-05
2ND Np CYCLE	1.79E-05	8.76E-02	7.81E-06	8.91E-05	1.08E-05	3.15E-03	3.74E-04	3.97E-04	4.07E-08	3.83E-04
HAW EVAPORATOR	1.07E-01	3.42E-01	1.04E-02	3.03E-02	1.07E-01	8.99E-01	1.42E-01	1.25E-01	1.33E-04	1.37E+00
LAW EVAPORATOR	4.48E-06	2.28E-01	8.68E-07	3.81E-06	2.71E-06	1.37E-04	1.52E-05	1.54E-05	1.02E-08	4.09E-05
ION EXCHANGE	7.54E-03	7.88E-02	4.40E-01	5.58E+00	6.39E-03	2.06E+02	2.58E+01	2.33E+01	3.22E-06	1.86E+01
EXPLOSION IN ION EXCHANGE	2.51E-02	4.38E-04	8.19E-03	1.87E+01	2.13E-02	6.90E+02	8.63E+01	7.77E+01	1.07E-05	6.20E+01
HIGH HEAT WASTE	1.06E-01	1.75E-01	5.31E-03	3.03E-02	1.07E-01	8.99E-01	1.42E-01	1.25E-01	1.33E-04	1.37E+00
TOTAL			4.79E-01							

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	1.60E-01	7.88E-01	2.02E-03	2.56E-03	1.06E-02	5.40E-02	1.11E-02	1.00E-02	2.49E-05	3.18E-01
HEAD END.	1.45E-01	1.75E-02	7.22E-05	4.12E-03	9.59E-03	1.15E-01	1.80E-02	1.56E-02	2.21E-05	2.97E-01
FIRST CYCLE	9.75E-02	8.76E-02	1.84E-04	2.10E-03	6.41E-03	4.61E-02	9.51E-03	7.53E-03	1.38E-05	1.87E-01
2ND U CYCLE	3.94E-07	8.76E-02	1.58E-08	1.80E-07	2.10E-08	3.14E-06	1.11E-07	5.52E-07	3.15E-11	9.77E-06
2ND Np CYCLE	1.79E-05	8.76E-02	8.32E-07	9.50E-06	6.95E-07	3.39E-04	3.77E-05	3.70E-05	4.57E-09	5.93E-05
HAW EVAPORATOR	1.07E-01	3.42E-01	1.33E-03	3.89E-03	7.04E-03	1.11E-01	1.73E-02	1.41E-02	1.51E-05	2.13E-01
LAW EVAPORATOR	4.48E-06	2.28E-01	1.06E-07	4.64E-07	1.75E-07	1.62E-05	1.71E-06	1.60E-06	1.14E-09	6.19E-06
ION EXCHANGE	7.54E-03	7.98E-02	4.56E-02	5.79E-01	4.24E-04	2.17E+01	2.55E+00	2.14E+00	3.65E-07	2.91E+00
EXPLOSION IN ION EXCHANGE	2.51E-02	4.38E-04	8.50E-04	1.94E+00	1.42E-03	7.26E+01	8.51E+00	7.15E+00	1.22E-06	9.74E+00
HIGH HEAT WASTE	1.06E-01	1.75E-01	6.82E-04	3.89E-03	7.04E-03	1.11E-01	1.73E-02	1.41E-02	1.51E-05	2.13E-01
TOTAL			5.08E-02							

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
ONSITE POPULATION			PERSON-REM/YR			DOSE, PERSON-REM				
DISSOLVER	1.60E-01	7.88E-01	2.11E-03	2.67E-03	3.35E-02	5.45E-02	1.22E-02	1.19E-02	6.08E-05	3.07E-01
HEAD END	1.45E-01	1.75E-02	9.79E-05	5.59E-03	3.02E-02	1.65E-01	2.59E-02	2.44E-02	5.41E-05	2.86E-01
FIRST CYCLE	9.75E-02	8.76E-02	1.91E-04	2.18E-03	2.01E-02	4.68E-02	1.05E-02	9.12E-03	3.37E-05	1.80E-01
2ND U CYCLE	3.94E-07	8.76E-02	1.38E-08	1.57E-07	6.49E-08	2.85E-06	1.47E-07	7.10E-07	7.69E-11	9.27E-06
2ND Np CYCLE	1.79E-05	8.76E-02	1.45E-06	1.65E-05	2.24E-06	5.83E-04	7.03E-05	7.61E-05	1.12E-08	5.76E-05
HAW EVAPORATOR	1.07E-01	3.42E-01	1.80E-03	5.28E-03	2.20E-02	1.59E-01	2.50E-02	2.25E-02	3.71E-05	2.05E-01
LAW EVAPORATOR	4.48E-06	2.28E-01	1.54E-07	6.77E-07	5.61E-07	2.44E-05	2.74E-06	2.87E-06	2.81E-09	6.25E-06
ION EXCHANGE	7.54E-03	7.88E-02	8.20E-02	1.04E+00	1.31E-03	3.84E+01	4.88E+00	4.48E+00	8.97E-07	2.76E+00
EXPLOSION IN ION EXCHANGE	2.51E-02	4.38E-04	1.53E-03	3.49E+00	4.36E-03	1.28E+02	1.63E+01	1.50E+01	2.99E-06	9.21E+00
HIGH HEAT WASTE	1.06E-01	1.75E-01	9.25E-04	5.28E-03	2.20E-02	1.59E-01	2.50E-02	2.25E-02	3.71E-05	2.05E-01
TOTAL			8.86E-02							

TABLE D-13. Dose and Risk to the Offsite Population from Stack Release due to Criticality

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	4.76E+04	1.31E-05	1.94E-06	1.48E+00	1.71E+00	1.04E+00	2.40E+00	3.31E+00	1.97E+02	1.23E+00
HEAD END	4.76E+04	1.93E-06	2.85E-06	1.48E+00	1.71E+00	1.04E+00	2.40E+00	3.31E+00	1.97E+02	1.23E+00
HEAD END EVAPORATOR	4.76E+04	1.05E-04	1.56E-04	1.48E+00	1.71E+00	1.04E+00	2.40E+00	3.31E+00	1.97E+02	1.23E+00
FIRST CYCLE	4.76E+04	1.56E-03	2.31E-03	1.48E+00	1.71E+00	1.04E+00	2.40E+00	3.31E+00	1.97E+02	1.23E+00
2ND U CYCLE	4.76E+04	1.56E-03	2.31E-03	1.48E+00	1.71E+00	1.04E+00	2.40E+00	3.31E+00	1.97E+02	1.23E+00
TOTAL			4.79E-03							

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	1.31E-05	1.73E-05	1.32E+00	1.25E+00	1.44E+00	1.65E+00	1.93E+00	6.77E+01	1.08E+00
HEAD END	4.76E+04	1.93E-06	2.54E-06	1.32E+00	1.25E+00	1.44E+00	1.65E+00	1.93E+00	6.77E+01	1.08E+00
HEAD END EVAPORATOR	4.76E+04	1.05E-04	1.39E-04	1.32E+00	1.25E+00	1.44E+00	1.65E+00	1.93E+00	6.77E+01	1.08E+00
FIRST CYCLE	4.76E+04	1.56E-03	2.06E-03	1.32E+00	1.25E+00	1.44E+00	1.65E+00	1.93E+00	6.77E+01	1.08E+00
2ND U CYCLE	4.76E+04	1.56E-03	2.06E-03	1.32E+00	1.25E+00	1.44E+00	1.65E+00	1.93E+00	6.77E+01	1.08E+00
TOTAL			4.28E-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	1.31E-05	4.34E-05	3.30E+00	3.35E+00	3.36E+00	3.88E+00	4.36E+00	7.41E+01	3.07E+00
HEAD END	4.76E+04	1.93E-06	6.36E-06	3.30E+00	3.35E+00	3.36E+00	3.88E+00	4.36E+00	7.41E+01	3.07E+00
HEAD END EVAPORATOR	4.76E+04	1.05E-04	3.47E-04	3.30E+00	3.35E+00	3.36E+00	3.88E+00	4.36E+00	7.41E+01	3.07E+00
FIRST CYCLE	4.76E+04	1.56E-03	5.15E-03	3.30E+00	3.35E+00	3.36E+00	3.88E+00	4.36E+00	7.41E+01	3.07E+00
2ND U CYCLE	4.76E+04	1.56E-03	5.15E-03	3.30E+00	3.35E+00	3.36E+00	3.88E+00	4.36E+00	7.41E+01	3.07E+00
TOTAL			1.07E-02							

TABLE D-16. Dose and Risk to the Offsite Population from Ground Level Release due to Transfer Error to 211-H

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	1.45E-04	1.75E-05	2.23E-09	1.27E-04	1.61E-04	3.87E-03	6.16E-04	4.89E-04	1.51E-07	2.00E-03
DISSOLVER	3.05E+00	7.01E-05	3.06E-04	4.37E+00	2.37E+00	1.50E+02	1.79E+01	1.63E+01	6.54E-03	2.88E+01
HEAD END	2.21E+01	1.05E-04	7.02E-04	6.68E+00	1.21E+01	1.61E+02	2.77E+01	2.14E+01	6.72E-02	1.16E+02
FIRST CYCLE	1.90E+01	1.14E-04	5.38E-04	4.72E+00	2.63E+01	1.37E+02	2.15E+01	2.07E+01	3.42E-02	3.40E+02
2ND U CYCLE	6.69E-09	3.68E-05	3.58E-13	9.72E-09	9.17E-09	2.17E-07	1.65E-08	3.98E-08	1.16E-11	5.37E-07
2ND Np CYCLE	1.21E-05	9.64E-05	2.33E-09	2.42E-05	1.50E-05	8.52E-04	1.01E-04	1.09E-04	2.90E-08	2.60E-04
HAW EVAPORATOR	2.11E+01	1.40E-04	7.39E-04	5.27E+00	2.92E+01	1.53E+02	2.40E+01	2.30E+01	3.79E-02	3.78E+02
LAW EVAPORATOR	8.79E-07	1.40E-04	4.42E-11	3.15E-07	1.12E-06	1.09E-05	1.20E-06	1.38E-06	2.07E-09	1.50E-05
ION EXCHANGE	5.61E-02	4.03E-04	2.22E-02	5.52E+01	6.29E-02	2.04E+03	2.56E+02	2.30E+02	3.12E-05	1.84E+02
HIGH HEAT WASTE	2.11E+01	1.75E-04	9.23E-04	5.27E+00	2.92E+01	1.53E+02	2.40E+01	2.30E+01	3.79E-02	3.78E+02
TOTAL			2.55E-02							

TABLE D-17. Dose and Risk to the Maximum Individual from Ground Level Release due to Transfer Error to 211-H

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	1.45E-04	1.75E-05	4.92E-10	2.81E-05	1.88E-05	8.30E-04	1.31E-04	9.37E-05	2.89E-08	5.47E-04
DISSOLVER	3.05E+00	7.01E-05	5.59E-05	7.98E-01	2.77E-01	2.89E+01	3.17E+00	2.70E+00	1.24E-03	7.76E+00
HEAD END	2.21E+01	1.05E-04	1.37E-04	1.30E+00	1.34E+00	3.47E+01	5.96E+00	4.14E+00	1.28E-02	3.11E+01
FIRST CYCLE	1.90E+01	1.14E-04	1.22E-04	1.07E+00	3.02E+00	2.96E+01	4.59E+00	4.21E+00	6.53E-03	9.26E+01
2ND U CYCLE	6.69E-09	3.68E-05	9.68E-14	2.63E-09	1.05E-09	5.36E-08	3.06E-09	8.50E-09	2.22E-12	1.47E-07
2ND Np CYCLE	1.21E-05	9.64E-05	4.38E-10	4.55E-06	1.70E-06	1.61E-04	1.78E-05	1.81E-05	5.51E-09	7.06E-05
HAW EVAPORATOR	2.11E+01	1.40E-04	1.68E-04	1.20E+00	3.35E+00	3.30E+01	5.11E+00	4.69E+00	7.24E-03	1.03E+02
LAW EVAPORATOR	8.79E-07	1.40E-04	9.59E-12	6.84E-08	1.27E-07	2.28E-06	2.37E-07	2.63E-07	3.94E-10	4.07E-06
ION EXCHANGE	5.61E-02	4.03E-04	4.03E-03	1.00E+01	7.30E-03	3.77E+02	4.42E+01	3.71E+01	5.91E-06	5.05E+01
HIGH HEAT WASTE	2.11E+01	1.75E-04	2.10E-04	1.20E+00	3.35E+00	3.30E+01	5.11E+00	4.69E+00	7.24E-03	1.03E+02
TOTAL			4.72E-03							

TABLE D-18. Dose and Risk to the Onsite Population from Ground Level Release due to Transfer Error to 211-H

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	1.45E-04	1.75E-05	8.06E-10	4.60E-05	6.87E-05	1.42E-03	2.24E-04	1.85E-04	2.55E-08	6.22E-04
DISSOLVER	3.05E+00	7.01E-05	1.19E-04	1.70E+00	1.02E+00	5.71E+01	6.96E+00	6.48E+00	1.07E-03	9.04E+00
HEAD END	2.21E+01	1.05E-04	2.67E-04	2.54E+00	5.20E+00	5.91E+01	1.00E+01	8.03E+00	1.10E-02	3.68E+01
FIRST CYCLE	1.90E+01	1.14E-04	1.92E-04	1.69E+00	1.12E+01	5.00E+01	7.82E+00	7.62E+00	5.61E-03	1.06E+02
2ND U CYCLE	6.69E-09	3.68E-05	1.18E-13	3.20E-09	3.91E-09	7.51E-08	6.30E-09	1.44E-08	1.90E-12	1.66E-07
2ND Np CYCLE	1.21E-05	9.64E-05	8.97E-10	9.31E-06	6.40E-06	3.27E-04	3.94E-05	4.33E-05	4.72E-09	8.09E-05
HAW EVAPORATOR	2.11E+01	1.40E-04	2.64E-04	1.88E+00	1.25E+01	5.58E+01	8.73E+00	8.50E+00	6.22E-03	1.18E+02
LAW EVAPORATOR	8.79E-07	1.40E-04	1.61E-11	1.15E-07	4.78E-07	4.04E-06	4.50E-07	5.23E-07	3.38E-10	4.69E-06
ION EXCHANGE	5.61E-02	4.03E-04	8.62E-03	2.14E+01	2.67E-02	7.89E+02	1.00E+02	9.19E+01	5.14E-06	5.66E+01
HIGH HEAT WASTE	2.11E+01	1.75E-04	3.29E-04	1.88E+00	1.25E+01	5.58E+01	8.73E+00	8.50E+00	6.22E-03	1.18E+02
TOTAL			9.79E-03							

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	7.01E-07	3.50E-03	1.65E-09	4.71E-07	5.96E-07	1.43E-05	2.28E-06	1.81E-06	5.63E-10	7.41E-06
DISSOLVER	1.69E-01	1.40E-02	3.35E-04	2.39E-02	1.78E-01	6.74E-01	1.05E-01	1.10E-01	2.45E-04	2.30E+00
HEAD END	1.52E-01	2.10E-02	4.52E-04	2.15E-02	1.60E-01	6.06E-01	9.49E-02	9.86E-02	2.20E-04	2.07E+00
FIRST CYCLE	9.32E-02	2.19E-02	3.83E-04	1.75E-02	9.77E-02	5.08E-01	7.98E-02	7.66E-02	1.27E-04	1.26E+00
2ND U CYCLE	3.28E-11	7.36E-03	1.99E-13	2.70E-11	3.39E-11	4.74E-10	2.00E-11	1.10E-10	4.33E-14	1.96E-09
2ND Np CYCLE	5.92E-08	1.93E-02	1.73E-09	8.96E-08	5.56E-08	3.15E-06	3.74E-07	4.04E-07	1.08E-10	9.62E-07
HAW EVAPORATOR	1.04E-01	2.80E-02	5.47E-04	1.95E-02	1.09E-01	5.64E-01	8.87E-02	8.52E-02	1.42E-04	1.40E+00
LAW EVAPORATOR	4.30E-09	2.80E-02	3.28E-11	1.17E-09	4.15E-09	4.03E-08	4.45E-09	5.11E-09	7.71E-12	5.57E-08
ION EXCHANGE	2.75E-04	8.06E-02	1.64E-02	2.04E-01	2.33E-04	7.55E+00	9.45E-01	8.52E-01	1.16E-07	6.79E-01
HIGH HEAT WASTE	1.04E-01	3.50E-02	6.83E-04	1.95E-02	1.09E-01	5.64E-01	8.87E-02	8.52E-02	1.42E-04	1.40E+00
TOTAL			1.88E-02							

TABLE D-20. Dose and Risk to the Maximum Individual 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
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR			DOSE, MREM				
DECONTAMINATION	7.01E-07	3.50E-03	2.08E-10	5.94E-08	3.97E-08	1.76E-06	2.78E-07	1.98E-07	6.47E-11	1.15E-06
DISSOLVER	1.69E-01	1.40E-02	4.39E-05	3.13E-03	1.16E-02	8.34E-02	1.28E-02	1.30E-02	2.80E-05	3.58E-01
HEAD END	1.52E-01	2.10E-02	5.91E-05	2.81E-03	1.04E-02	7.50E-02	1.15E-02	1.17E-02	2.51E-05	3.21E-01
FIRST CYCLE	9.32E-02	2.19E-02	4.97E-05	2.27E-03	6.39E-03	6.27E-02	9.70E-03	8.90E-03	1.45E-05	1.96E-01
2ND U CYCLE	3.28E-11	7.36E-03	3.43E-14	4.66E-12	2.22E-12	8.01E-11	2.58E-12	1.46E-11	4.94E-15	3.07E-10
2ND Np CYCLE	5.92E-08	1.93E-02	1.85E-10	9.60E-09	3.61E-09	3.39E-07	3.76E-08	3.81E-08	1.22E-11	1.49E-07
HAW EVAPORATOR	1.04E-01	2.80E-02	7.09E-05	2.53E-03	7.11E-03	6.97E-02	1.08E-02	9.90E-03	1.62E-05	2.18E-01
LAW EVAPORATOR	4.30E-09	2.80E-02	4.06E-12	1.45E-10	2.70E-10	4.82E-09	5.01E-10	5.56E-10	8.75E-13	8.61E-09
ION EXCHANGE	2.75E-04	8.06E-02	1.71E-03	2.12E-02	1.54E-05	7.96E-01	9.32E-02	7.83E-02	1.31E-08	1.07E-01
HIGH HEAT WASTE	1.04E-01	3.50E-02	8.87E-05	2.53E-03	7.11E-03	6.97E-02	1.08E-02	9.90E-03	1.62E-05	2.10E-01
TOTAL			2.02E-03							

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
ONSITE POPULATION			PERSON-REM/YR			DOSE, PERSON-REM				
DECONTAMINATION	7.01E-07	3.50E-03	2.88E-10	8.23E-08	1.22E-07	2.54E-06	4.01E-07	3.30E-07	1.58E-10	1.11E-06
DISSOLVER	1.69E-01	1.40E-02	5.82E-05	4.15E-03	3.66E-02	1.19E-01	1.85E-02	1.94E-02	6.84E-05	3.45E-01
HEAD END	1.52E-01	2.10E-02	7.84E-05	3.73E-03	3.28E-02	1.07E-01	1.67E-02	1.74E-02	6.15E-05	3.10E-01
FIRST CYCLE	9.32E-02	2.19E-02	6.66E-05	3.04E-03	2.01E-02	8.94E-02	1.40E-02	1.36E-02	3.55E-05	1.89E-01
2ND U CYCLE	3.28E-11	7.36E-03	2.96E-14	4.02E-12	6.97E-12	7.17E-11	3.43E-12	1.84E-11	1.21E-14	2.91E-10
2ND Np CYCLE	5.92E-08	1.93E-02	3.20E-10	1.66E-08	1.14E-08	5.83E-07	7.02E-08	7.72E-08	3.00E-11	1.44E-07
HAW EVAPORATOR	1.04E-01	2.80E-02	9.47E-05	3.38E-03	2.23E-02	9.95E-02	1.56E-02	1.52E-02	3.96E-05	2.10E-01
LAW EVAPORATOR	4.30E-09	2.80E-02	5.77E-12	2.06E-10	8.55E-10	7.20E-09	8.05E-10	9.34E-10	2.14E-12	8.38E-09
ION EXCHANGE	2.75E-04	8.06E-02	3.08E-03	3.82E-02	4.77E-05	1.41E+00	1.79E-01	1.64E-01	3.23E-08	1.01E-01
HIGH HEAT WASTE	1.04E-01	3.50E-02	1.18E-04	3.38E-03	2.23E-02	9.95E-02	1.56E-02	1.52E-02	3.96E-05	2.10E-01
TOTAL			3.49E-03							

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	4.18E-07	4.38E-03	1.21E-09	2.76E-07	3.50E-07	8.39E-06	1.34E-06	1.06E-06	3.32E-10	4.36E-06
DISSOLVER	9.93E-02	3.50E-01	4.94E-03	1.41E-02	1.04E-01	3.96E-01	6.20E-02	6.45E-02	1.44E-04	1.35E+00
HEAD END	8.90E-02	8.76E-02	1.10E-03	1.26E-02	9.36E-02	3.54E-01	5.54E-02	5.77E-02	1.29E-04	1.21E+00
FIRST CYCLE	5.45E-02	3.33E-01	3.43E-03	1.03E-02	5.72E-02	2.97E-01	4.67E-02	4.49E-02	7.44E-05	7.38E-01
2ND U CYCLE	1.92E-11	2.98E-02	4.71E-13	1.58E-11	1.99E-11	2.78E-10	1.17E-11	6.43E-11	2.54E-14	1.15E-09
2ND Np CYCLE	3.47E-08	5.43E-02	2.85E-09	5.25E-08	3.26E-08	1.85E-06	2.19E-07	2.37E-07	6.34E-11	5.64E-07
HAW EVAPORATOR	6.08E-02	3.94E-01	4.49E-03	1.14E-02	6.37E-02	3.31E-01	5.21E-02	5.01E-02	8.30E-05	8.22E-01
LAW EVAPORATOR	2.53E-09	3.94E-01	2.70E-10	6.85E-10	2.43E-09	2.37E-08	2.61E-09	3.00E-09	4.52E-12	3.26E-08
ION EXCHANGE	5.41E-04	8.76E-01	3.53E-01	4.03E-01	4.59E-04	1.49E+01	1.86E+00	1.68E+00	2.29E-07	1.34E+00
HIGH HEAT WASTE	6.08E-02	5.26E-01	5.99E-03	1.14E-02	6.37E-02	3.31E-01	5.21E-02	5.01E-02	8.30E-05	8.22E-01
TOTAL			3.73E-01							

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)			MREM/YR			DOSE, MREM				
DECONTAMINATION	4.10E-07	4.30E-03	1.53E-10	3.49E-08	2.34E-08	1.03E-06	1.63E-07	1.16E-07	3.81E-11	6.79E-07
DISSOLVER	9.93E-02	3.50E-01	6.45E-04	1.84E-03	6.81E-03	4.90E-02	7.54E-03	7.66E-03	1.64E-05	2.10E-01
HEAD END	8.90E-02	8.76E-02	1.44E-04	1.64E-03	6.10E-03	4.39E-02	6.74E-03	6.85E-03	1.47E-05	1.88E-01
FIRST CYCLE	5.45E-02	3.33E-01	4.43E-04	1.33E-03	3.74E-03	3.67E-02	5.68E-03	5.22E-03	8.50E-06	1.15E-01
2ND U CYCLE	1.92E-11	2.98E-02	8.13E-14	2.73E-12	1.30E-12	4.68E-11	1.51E-12	8.57E-12	2.89E-15	1.74E-10
2ND Np CYCLE	3.47E-08	5.43E-02	3.06E-10	5.63E-09	2.11E-09	1.99E-07	2.21E-08	2.23E-08	7.20E-12	8.74E-08
HAW EVAPORATOR	6.08E-02	3.94E-01	5.83E-04	1.48E-03	4.17E-03	4.09E-02	6.33E-03	5.82E-03	9.47E-06	1.28E-01
LAW EVAPORATOR	2.53E-09	3.94E-01	3.35E-11	8.49E-11	1.58E-10	2.83E-09	2.90E-10	3.27E-10	5.12E-13	5.04E-09
ION EXCHANGE	5.41E-04	2.76E-01	3.66E-02	4.18E-02	3.04E-05	1.57E+00	1.84E-01	1.54E-01	2.59E-08	2.10E-01
HIGH HEAT WASTE	6.08E-02	5.26E-01	7.78E-04	1.48E-03	4.17E-03	4.09E-02	6.33E-03	5.82E-03	9.47E-06	1.28E-01
TOTAL			3.92E-02							

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
ONSITE POPULATION			PERSON-REM/YR			DOSE, PERSON-REM				
DECONTAMINATION	4.18E-07	4.38E-03	2.11E-10	4.82E-08	7.20E-08	1.48E-06	2.35E-07	1.93E-07	9.32E-11	6.53E-07
DISSOLVER	9.93E-02	3.50E-01	8.55E-04	2.44E-03	2.15E-02	6.97E-02	1.09E-02	1.14E-02	4.02E-05	2.03E-01
HEAD END	8.90E-02	8.76E-02	1.91E-04	2.18E-03	1.92E-02	6.23E-02	9.74E-03	1.02E-02	3.60E-05	1.81E-01
FIRST CYCLE	5.45E-02	3.33E-01	5.93E-04	1.78E-03	1.18E-02	5.24E-02	8.19E-03	7.99E-03	2.08E-05	1.10E-01
2ND U CYCLE	1.92E-11	2.98E-02	7.00E-14	2.35E-12	4.08E-12	4.19E-11	2.01E-12	1.08E-11	7.07E-15	1.71E-10
2ND Np CYCLE	3.47E-08	5.43E-02	5.28E-10	9.73E-09	6.71E-09	3.41E-07	4.12E-08	4.53E-08	1.76E-11	8.46E-08
HAW EVAPORATOR	6.08E-02	3.94E-01	7.84E-04	1.99E-03	1.31E-02	5.84E-02	9.14E-03	8.91E-03	2.32E-05	1.23E-01
LAW EVAPORATOR	2.53E-09	3.94E-01	4.77E-11	1.21E-10	5.01E-10	4.23E-09	4.73E-10	5.49E-10	1.26E-12	4.91E-09
ION EXCHANGE	5.41E-04	8.76E-01	6.60E-02	7.53E-02	9.39E-05	2.77E+00	3.52E-01	3.23E-01	6.36E-08	1.99E-01
HIGH HEAT WASTE	6.08E-02	5.26E-01	1.05E-03	1.99E-03	1.31E-02	5.84E-02	9.14E-03	8.91E-03	2.32E-05	1.23E-01
TOTAL			6.94E-02							

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	1.40E-07	4.20E+00	3.89E-07	9.24E-08	1.17E-07	2.81E-06	4.48E-07	3.56E-07	1.11E-10	1.46E-06
DISSOLVER	1.41E-02	5.43E-01	1.06E-03	1.96E-03	1.31E-02	5.59E-02	8.75E-03	9.00E-03	2.30E-06	1.81E-01
HEAD END	1.25E-02	4.38E+00	7.58E-03	1.73E-03	1.31E-02	4.96E-02	7.72E-03	8.08E-03	1.79E-05	1.70E-01
FIRST CYCLE	7.64E-03	6.31E+00	9.08E-03	1.44E-03	7.99E-03	4.16E-02	6.54E-03	6.29E-03	1.04E-05	1.03E-01
2ND U CYCLE	2.71E-12	3.15E+00	7.03E-12	2.23E-12	2.81E-12	3.92E-11	1.65E-12	9.08E-12	3.58E-15	1.62E-10
2ND Np CYCLE	4.89E-09	1.05E+01	7.79E-08	7.41E-09	4.57E-09	2.61E-07	3.09E-08	3.34E-08	8.92E-12	7.93E-08
HAW EVAPORATOR	8.62E-03	1.58E+01	2.00E-02	1.27E-03	7.95E-03	4.04E-02	4.89E-03	5.65E-03	1.17E-05	1.02E-01
LAW EVAPORATOR	3.58E-10	1.58E+01	2.08E-04	1.32E-05	3.64E-04	4.61E-04	6.93E-10	7.48E-10	3.24E-10	2.00E-03
ION EXCHANGE	2.58E-04	1.05E+01	2.02E+00	1.92E-01	2.19E-04	7.12E+00	8.91E-01	8.02E-01	1.09E-07	6.40E-01
HIGH HEAT WASTE	8.62E-03	6.13E-01	7.79E-04	1.27E-03	7.95E-03	4.04E-02	4.89E-03	5.65E-03	1.17E-05	1.02E-01
TOTAL			2.06E+00							

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	1.40E-07	4.20E+00	4.92E-08	1.17E-08	7.82E-09	3.45E-07	5.46E-08	3.90E-08	1.28E-11	2.27E-07
DISSOLVER	1.41E-02	5.43E-01	1.39E-04	2.56E-04	8.64E-04	6.93E-03	1.06E-03	1.07E-03	2.63E-07	2.83E-02
HEAD END	1.25E-02	4.38E+00	1.00E-03	2.29E-04	8.55E-04	6.15E-03	9.38E-04	9.58E-04	2.04E-06	2.64E-02
FIRST CYCLE	7.64E-03	6.31E+00	1.18E-03	1.87E-04	5.23E-04	5.14E-03	7.96E-04	7.31E-04	1.19E-06	1.60E-02
2ND U CYCLE	2.71E-12	3.15E+00	1.21E-12	3.85E-13	1.83E-13	6.61E-12	2.13E-13	1.21E-12	4.08E-16	2.53E-11
2ND Np CYCLE	4.89E-09	1.05E+01	8.35E-09	7.94E-10	2.97E-10	2.80E-08	3.11E-09	3.15E-09	1.01E-12	1.23E-08
HAW EVAPORATOR	8.62E-03	1.58E+01	2.27E-03	1.44E-04	5.19E-04	4.55E-03	5.07E-04	6.08E-04	1.34E-06	1.59E-02
LAW EVAPORATOR	3.58E-10	1.58E+01	3.97E-05	2.52E-06	2.45E-05	8.77E-05	7.77E-11	8.22E-11	3.61E-11	3.16E-04
ION EXCHANGE	2.58E-04	1.05E+01	2.10E-01	2.00E-02	1.45E-05	7.50E-01	8.78E-02	7.38E-02	1.24E-08	1.01E-01
HIGH HEAT WASTE	8.62E-03	6.13E-01	8.83E-05	1.44E-04	5.19E-04	4.55E-03	5.07E-04	6.08E-04	1.34E-06	1.59E-02
TOTAL			2.15E-01							

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	1.40E-07	4.20E+00	6.81E-08	1.62E-08	2.41E-08	4.97E-07	7.87E-08	6.48E-08	3.12E-11	2.18E-07
DISSOLVER	1.41E-02	5.43E-01	1.84E-04	3.38E-04	2.68E-03	9.85E-03	1.54E-03	1.59E-03	6.43E-07	2.70E-02
HEAD END	1.25E-02	4.38E+00	1.30E-03	2.97E-04	2.69E-03	8.74E-03	1.36E-03	1.43E-03	4.99E-06	2.54E-02
FIRST CYCLE	7.64E-03	6.31E+00	1.58E-03	2.50E-04	1.64E-03	7.34E-03	1.15E-03	1.12E-03	2.91E-06	1.54E-02
2ND U CYCLE	2.71E-12	3.15E+00	1.05E-12	3.32E-13	5.76E-13	5.92E-12	2.83E-13	1.52E-12	9.98E-16	2.41E-11
2ND Np CYCLE	4.89E-09	1.05E+01	1.44E-08	1.37E-09	9.42E-10	4.82E-08	5.81E-09	6.38E-09	2.48E-12	1.19E-08
HAW EVAPORATOR	8.62E-03	1.58E+01	3.66E-03	2.32E-04	1.64E-03	7.35E-03	9.09E-04	1.03E-03	3.27E-06	1.53E-02
LAW EVAPORATOR	3.58E-10	1.58E+01	2.93E-05	1.86E-06	7.46E-05	6.49E-05	1.56E-10	1.66E-10	8.89E-11	2.99E-04
ION EXCHANGE	2.58E-04	1.05E+01	3.78E-01	3.60E-02	4.49E-05	1.32E+00	1.68E-01	1.54E-01	3.05E-08	9.50E-02
HIGH HEAT WASTE	8.62E-03	6.13E-01	1.42E-04	2.32E-04	1.64E-03	7.35E-03	9.09E-04	1.03E-03	3.27E-06	1.53E-02
TOTAL			3.85E-01							

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	3.64E-03	1.04E-01	2.76E-02	1.44E-01	1.80E-01	3.03E-01	5.84E+01	3.53E-03
DISSOLVER-Ru VOLATILIZATION	4.00E-01	8.76E-02	1.09E-03	1.24E-02	8.94E-01	9.86E-02		1.70E-01		1.26E+01
HEAD END-Ru VOLATILIZATION	4.50E-02	8.76E-02	1.23E-04	1.40E-03	1.01E-01	1.11E-02		1.91E-02		1.42E+00
MAXIMUM INDIVIDUAL (CHILD)			MREM/YR			DOSE, MREM				
DISSOLVER-SHORT COOLED FUELS	1.65E+00	3.50E-02	5.54E-04	1.58E-02	2.02E-03	2.75E-02	2.75E-02	4.48E-02	9.16E+00	4.19E-04
DISSOLVER-Ru VOLATILIZATION	4.00E-01	8.76E-02	2.03E-04	2.32E-03	5.89E-02	1.86E-02		2.52E-02		1.96E+00
HEAD END-Ru VOLATILIZATION	4.50E-02	8.76E-02	2.29E-05	2.61E-04	6.62E-03	2.10E-03		2.83E-03		2.21E-01
ONSITE POPULATION			PERSON-REM/YR			DOSE, PERSON-REM				
DISSOLVER-SHORT COOLED FUELS	1.65E+00	3.50E-02	6.10E-04	1.74E-02	6.03E-03	2.12E-02	2.96E-02	5.01E-02	9.54E+00	1.01E-03
DISSOLVER-Ru VOLATILIZATION	4.00E-01	8.76E-02	1.53E-04	1.78E-03	1.83E-01	1.38E-02		2.68E-02		1.87E+00
HEAD END-Ru VOLATILIZATION	4.50E-02	8.76E-02	1.73E-05	1.97E-04	2.86E-02	1.56E-03		3.01E-03		2.11E-01

TABLE D-29. Dose and Risk to the Offsite Population from Liquid Release 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	1.88E-05	3.50E-02	9.00E-02	2.89E-01	5.00E-02	1.95E-02	4.09E-05	6.66E-03	6.20E-05
DISSOLVER	1.72E+01	9.11E-03	2.21E-02	2.43E+00	6.10E+00	5.20E+00	3.99E+00	1.32E+00	3.31E-03	4.51E-01	3.52E-03
HEAD END	1.72E+01	5.61E-03	1.36E-02	2.43E+00	6.10E+00	5.20E+00	3.99E+00	1.32E+00	3.31E-03	4.51E-01	3.52E-03
FIRST CYCLE	1.96E+01	8.41E-04	2.13E-03	2.53E+00	2.57E+00	5.63E+00	4.07E+00	1.34E+00	3.34E-03	4.59E-01	3.49E-03
2ND U CYCLE	6.86E-09	8.41E-04	7.44E-13	8.85E-10	9.00E-10	1.97E-09	1.42E-09	4.70E-10	1.17E-12	1.61E-10	1.22E-12
2ND Np CYCLE	1.25E-05	8.41E-04	1.36E-09	1.62E-06	1.65E-06	3.61E-06	2.60E-06	8.59E-07	2.14E-09	2.94E-07	2.23E-09
RAW EVAPORATOR	1.96E+01	1.19E-02	3.01E-02	2.53E+00	2.57E+00	5.63E+00	4.07E+00	1.34E+00	3.34E-03	4.59E-01	3.49E-03
LAW EVAPORATOR	8.23E-07	1.19E-02	1.27E-09	1.06E-07	1.00E-07	2.37E-07	1.71E-07	5.64E-08	1.40E-10	1.93E-08	1.47E-10
HIGH HEAT WASTE	1.96E+01	8.41E-04	2.13E-03	2.53E+00	2.57E+00	5.63E+00	4.07E+00	1.34E+00	3.34E-03	4.59E-01	3.49E-03
TOTAL			7.02E-02								

TABLE D-30. Dose and Risk to the Maximum Individual from Liquid Release 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	1.59E-06	3.02E-03	8.34E-03	2.06E-02	1.66E-02	5.31E-03	7.40E-06	1.91E-03	6.72E-06
DISSOLVER	1.72E+01	9.11E-03	1.07E-03	2.05E-01	5.65E-01	1.40E+00	1.12E+00	3.60E-01	5.02E-04	1.29E-01	4.56E-04
HEAD END	1.72E+01	5.61E-03	1.15E-03	2.05E-01	5.65E-01	1.40E+00	1.12E+00	3.60E-01	5.02E-04	1.29E-01	4.56E-04
FIRST CYCLE	1.96E+01	8.41E-04	1.80E-04	2.14E-01	1.55E-01	1.45E+00	1.15E+00	3.64E-01	5.21E-04	1.31E-01	4.51E-04
2ND U CYCLE	6.86E-09	8.41E-04	6.31E-14	7.50E-11	5.41E-11	5.07E-10	4.01E-10	1.27E-10	1.82E-13	4.57E-11	1.50E-13
2ND Np CYCLE	1.25E-05	8.41E-04	1.15E-10	1.37E-07	9.89E-08	9.26E-07	7.33E-07	2.33E-07	3.34E-10	8.36E-08	2.89E-10
HAW EVAPORATOR	1.96E+01	1.19E-02	2.55E-03	2.14E-01	1.55E-01	1.45E+00	1.15E+00	3.64E-01	5.21E-04	1.31E-01	4.51E-04
LAW EVAPORATOR	8.23E-07	1.19E-02	1.07E-10	9.00E-09	6.49E-09	6.08E-08	4.81E-08	1.53E-08	2.19E-11	5.49E-09	1.89E-11
HIGH HEAT WASTE	1.96E+01	8.41E-04	1.80E-04	2.14E-01	1.55E-01	1.45E+00	1.15E+00	3.64E-01	5.21E-04	1.31E-01	4.51E-04
TOTAL			5.93E-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-5

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: 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/ft^2. 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 1CU 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 1CU 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.

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 NO_2 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, CO_2 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 NOx 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 SAFETY

General 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-5, SAFETY ANALYSIS - 200 AREA SAVANNAH RIVER PLANT, H-CANYON OPERATIONS

The H-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 H-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 2.0×10^{-5}/hr (0.18/year) or about one per six years, with the most likely injury being an acid burn to the skin. The evaluated frequency for hazardous vapor injury in CFP is $<1.4 \times 10^{-5}$/hr (0.12/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 Guidelines[1] with the Military Standard 882B[2] (MIL-STD-882B), chemical hazards in the H-Canyon Operations Area are assessed as IIIE (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 30 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 (2.8×10^{-6}/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 2×10^{-5}/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 H-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.