AR TARGET SHEET

The following document was too large to scan as one unit, therefore it has been broken down into sections.

DOCUMENT # SD-HS-SAR-001

EDMC # 0035786

SECTION 3 OF 3

THIS PAGE INTENTIONALLY
LEFT BLANK

9413219.0939

PUREX FSAR

CHAPTER 9.0 ACCIDENT SAFETY ANALYSIS

9413219.0940

This page intentionally left blank.

CONTENTS

9.0 Accident Safety Analysis 9-1

9.1 Methodology of Analysis 9-1
9.1.1 Scoping 9-1
9.1.2 Preliminary Hazards Analysis 9-2
9.1.3 Identification of Key Risk Contributions from PHA 9-3
9.1.4 Fault Tree Analysis 9-21
9.1.5 Major Risk Contributors 9-21
9.1.6 Limitations 9-23
9.1.7 Consequence Evaluation 9-24

9.2 Analysis of PUREX Accident Scenarios 9-29
9.2.1 Uncontrolled Radiation Release Due to External Phenomena 9-30
9.2.2 Criticality in the Plutonium Oxide Production Facility 9-35
9.2.3 Dissolving Short Cooled Fuel 9-39
9.2.4 Uranium Metal Fire in Dissolver 9-48
9.2.5 Silver Reactor Explosion 9-50
9.2.6 Hydrogen Explosion in a Dissolver 9-59
9.2.7 1BX Column Criticality 9-64
9.2.8 Solvent Fire in H Cell 9-65
9.2.9 Solvent Fire in L Cell 9-70
9.2.10 Plutonium Release via the P&O Gallery 9-76
9.2.11 Fission Product Release via the P&O Gallery 9-86
9.2.12 Oxide Line Fire 9-91
9.2.13 Cooling Coil Leak with Release to 216-A-25 Pond 9-93
9.2.14 Personnel Overexposure During Fuel Charging 9-98

9.3 Evaluation of Results 9-105
9.3.1 Hazard Level 9-105
9.3.2 Risk Acceptability 9-105
9.3.3 Consequence Guidelines 9-109

9.4 Summary 9-109

9.5 References 9-109

FIGURES:
9-1. 200 East Population Grid 9-26
9-2. Hanford Site Population Grid 9-27

TABLES:
9-1. Probability Classes 9-4
9-2. Consequence Classes 9-4
9-3. PUREX Preliminary Hazards Analysis 9-5
9-4. Hanford Site Population Distribution from the PUREX Plant 9-25

TABLES: (Continued)

9-5. Offsite Distribution of Population Within 50-mi of the 200 Area Hanford Meteorological Tower by Population Grid Sector for 1980 9-28
9-6. Seismic Analysis Results 9-31
9-7. Wind and Tornado Resistance Evaluations 9-33
9-8. Occurrence Probability of Seismically-Induced Dispersions 9-35
9-9. Summary of Assumptions for Plutonium Oxide Production Facility Criticality Accident Scenario. 9-36
9-10. Adverse Meteorology Summary: N Cell Criticality, 8-h Release from Ground Level 9-37
9-11. Radionuclide Release at Building Envelope from N Cell Criticality 9-38
9-12 Doses Resulting from Airborne Releases Due to N Cell Criticality 9-40
9-13. Summary of Assumptions for Calculating Source Terms for the "Dissolution of Short-Cooled Fuel" Accident Scenario 9-43
9-14. Adverse Meteorology Summary: Short-Cooled Fuel 3-h Release from 200-ft-high Stack 9-44
9-15. Release of Building Envelope from Dissolution of Short-Cooled Fuel 9-45
9-16. Doses Resulting from Dissolution of Short-Cooled Fuel . . 9-47
9-17. Summary of Assumptions for Calculating Source Terms for the "Uranium Metal Fire in Dissolver" Accident Scenario 9-49
9-18. Adverse Meteorology Summary: Uranium Fire 8-h Release from the 200-ft-high Stack 9-51
9-19. Release at Building Envelope from a Uranium Metal Fire in a Dissolver 9-52
9-20. Doses Resulting from a Uranium Metal Fire in a Dissolver 9-53
9-21. Summary of Assumptions for Calculating Source Terms for the "Silver Reactor Explosion" Accident Scenario . . 9-55
9-22. Adverse Meteorology Summary: Silver Reactor Explosion 10-min Release from the 200-ft-high Stack 9-56
9-23. Release at Building Envelope from Silver Reactor Explosion 9-57
9-24. Doses Resulting from Silver Reactor Explosion 9-58
9-25. Summary of Assumptions for Calculating Source Terms for the "Hydrogen Explosion in a Dissolver" Accident Scenario 9-60
9-26. Adverse Meteorology Summary: Hydrogen Explosion 10-min Release from the 200-ft-high Stack 9-61
9-27. Release at Building Envelope from a Hydrogen Explosion in a Dissolver 9-62
9-28. Doses Resulting from a Hydrogen Explosion in a Dissolver 9-63
9-29. Summary of Assumptions for Calculating Source Terms for the "1BX Column Criticality" Accident Scenario . . . 9-66
9-30. Adverse Meteorology Summary: 1 BX Column Criticality 8-h Release from the 200-ft-high Stack 9-67

9413219.0942

TABLES: (Continued)

9-31. Radionuclide Release at Building Envelope from 1BX Column Criticality 9-68
9-32. Doses Resulting from 1BX Column Criticality 9-69
9-33. Summary of Assumptions for the "Solvent Fire in H Cell" Accident Scenario 9-71
9-34. Adverse Meteorology Summary: Solvent Fire H Cell 8-h Release from the 200-ft-high Stack 9-72
9-35. Release at Building Envelope from Solvent Fire in H Cell 9-73
9-36. Doses Resulting from Solvent Fire in H Cell 9-75
9-37. Summary of Assumptions for the "Solvent Fire in L Cell" Accident Scenario 9-77
9-38. Adverse Meteorology Summary: Solvent Fire L-Cell 20-min Release from the 200-ft-high Stack 9-77
9-39. Release at Building Envelope from Solvent Fire in L Cell 9-78
9-40. Doses Resulting from Solvent Fire in L Cell 9-79
9-41. Summary of Assumptions for the "Plutonium Release via the P&O Gallery" Accident Scenario 9-81
9-42. Adverse Meteorology Summary: Plutonium Release via P&O Gallery 20-min Release from Ground Level 9-82
9-43. Release at Building Envelope from P&O Gallery Blowback Involving Plutonium 9-83
9-44. Doses Resulting from Plutonium Release from Pipe and Operating Gallery 9-85
9-45. Summary of Assumptions for the "Fission Product Release via the P&O Gallery" Accident Scenario 9-87
9-46. Adverse Meteorology Summary: Fission Product Release via P&O Gallery, 15-min Release from Ground Level 9-87
9-47. Fission Product Release at Building Envelope from P&O Gallery Blowback 9-88
9-48. Doses Resulting from Fission Product Release from Pipe and Operating Gallery 9-92
9-49. Summary of Assumptions for the "Oxide Line Fire" Accident Scenario 9-94
9-50. Adverse Meteorology Summary: Oxide Line Fire 3-h Release from Ground Level 9-95
9-51. Release at Building Envelope from Oxide Line Fire 9-95
9-52. Doses Resulting from Oxide Line Fire 9-96
9-53. Summary of Assumptions for the Calculating Source Terms for the "Cooling Coil Leak with Release to 216-A-25 Pond" Accident Scenario 9-99
9-54. Adverse Meteorology Summary: Cooling Coil Leak, 8-d Release from Ground Level 9-100
9-55. Release to Atmosphere from Cooling Coil Leak to 216-A-25 Pond 9-101
9-56. Doses Resulting from Cooling Coil Leak to 216-A-25 Pond 9-103
9-57. Summary of Assumptions for the "Personnel Overexposure During Fuel Charging" Accident Scenario 9-104
9-58. Accident Analysis Summary 9-106
9-59. Hazard Level Criteria 9-108

9413219 0944

This page intentionally left blank.

9.0 ACCIDENT SAFETY ANALYSIS

9.1 METHODOLOGY OF ANALYSIS

9.1.1 Scoping

Rockwell Hanford Operations (Rockwell) has performed a safety analysis of operation of the Plutonium-Uranium Reduction Extraction (PUREX) Plant. This analysis was performed understanding the limitations of the numerical estimates of the probabilities and consequences of accidents associated with any industry. The set of accidents chosen to represent the risk of PUREX operations is acknowledged to be incomplete. The intention of this analysis is that by careful selection of the accident set and conservative treatment of those selected, the results will approximate the true risk. To initiate the selection, the analysis was centered on the key safety related areas of the PUREX operation. Safety related areas were identified by three criteria that govern potential radiological consequences:

- Inventory of radioactive material
- Chemical and physical form of the radioactive material
- Presence of processing conditions that contain driving forces for potential accidents.

Based upon these criteria, the following areas were selected for analysis:

- Fuel receiving (includes handling and transferring)
- Fuel decladding and dissolving
- Feed preparations
- Solvent extraction
- Plutonium concentrators
- Plutonium oxide production
- Waste treatment (primarily high-level) within the PUREX Plant.

Obviously, accidents with no radiological consequences may occur during the operating life of the facility: fires, chemical reactions, falls, etc. In this respect PUREX shares the same risks as those presented by nonnuclear chemical processing facilities. These accidents are discussed, but they are not included in the total estimated risk of the facility. No significant hazards were identified which did not involve radiological consequences.

9.1.2 Preliminary Hazards Analysis

Every effort has been made to ensure that information regarding risk in key areas is as complete as possible; however, some hazards or potential accidents may not be reviewed because of simplification, oversight, or inability to conceive of all possible release sequences. There is no certain method for ensuring that all conceivable hazards are determined. It is important, however, that all hazards with the potential for significant impacts are understood and evaluated. The best method available for identifying these hazards is the application of thorough peer review by experienced personnel to all aspects of plant operations. This was accomplished in the preparation of the PUREX preliminary hazards analysis (PHA). A PHA postulates a specific component failure or initiation event and determines the corresponding effect on the overall system. The PUREX PHA required development of a detailed model of the PUREX system. The model of the PUREX Plant as analyzed is presented in Appendix C. It identifies all major process operations, all inputs and all outputs. Instantaneous inventories of radioactive materials are known for each segment or block of the operation, as well as common and potential sources of energy. For the hazards analysis, each component of the operation was reviewed for potential hazards, hazard causes, and engineered and administrative barriers. The PHA gives the following information for each process operations analyzed.

- System/Subsystem

  This is the general process area or vessel of concern, keyed to Figures C-1 through C-19.

- Potential Accidents

  Potential accidents which may occur in this area are identified here.

- Causal Factors

  Basic causes and contributing factors are given. When multiple failures are required, the word "and" is included.

- Barriers

  Identified here are physical systems or administrative actions that tend to prevent the accident from occurring, mitigate the potential consequences, or detect an impending accident.

- Characteristic Source Terms

  This is a qualitative description of the potential radioactive materials that may be dispersed.

- Category

  The assessed severity of consequences and probability of occurrence are identified.

- Controlling Operational Safety Requirement

  This section provides a "road map" from identified PUREX hazards to the operational safety requirement (OSR) that provides administrative control over the hazard. The OSR for PUREX are provided in Chapter 11.0 of this document. The OSR called out in this section may serve to prevent the accident or mitigate its consequences.

In addition to the review and analysis done by Rockwell personnel, an independent PHA was performed by Pacific Northwest Laboratory (PNL), as documented in Reference 9-1. Both evaluations identify similar risk-producing sequences.

9413219.0947

9.1.3 Identification of Key Risk Contributors from PHA

A screening process was used to limit the number of detailed analyses to those accidents that effectively determine the risk envelope for the PUREX Plant. The screening is based on a simplification of a more complete risk expression. The risk of a radionuclide release is a mathematical product of five terms such that:

$$\text{Risk} = A \times B \times C \times D \times E$$

where

A = probability of release sequence

B = release magnitude

C = measure of the radiological characteristic of the released material

D = measure of the environmental transport factor

E = measure of population at risk.

The total risk involved in operation of the facility is therefore the summation of this expression for all release sequences.

For the purpose of comparing and screening sequences for one facility at one site, the expressions C, D, and E may be held as constants, leaving only the requirement for the determination of A and B to determine risk. The PUREX PHA was reviewed by a team of highly experienced PUREX personnel with representation from Operations; Safety and Quality Assurance (S&QA); and Research and Engineering (R&E). Probability estimates were made on each accident sequence identified in the PHA based on the criteria in Table 9-1. Consequence estimates were made based on the criteria in Table 9-2. Using these terms, a risk factor was developed for each accident identified in the

PHA: Risk = A x B. The PUREX PHA is shown in Table 9-3. Dominant release sequences in each of the six following categories were selected for detailed evaluation:

- Breach of containment
- Confinement barrier failure
- Uncontrolled chemical reaction
- Nuclear excursion
- Extrinsic occurrences affecting plant operation
- Industrial hazards.

TABLE 9-1. Probability Classes.[a]

Probability level	Estimated probability (yr^{-1})	Description
3	1	Incident considered as part of normal operations. May occur several times during lifetime of facility.
2	10^{-2}	May occur sometime during lifetime of facility.
1	10^{-6}	Probably will not occur during lifetime of facility--DBA.[b]
0	10^{-8}	No reasonable scenario is conceivable.

[a]Based upon accident classifications in ANSI N287.(2)
[b]Design basis accident.

TABLE 9-2. Consequence Classes.*

Severity level	Descriptive word	Description
0	No	No onsite or offsite impact to people or environment.
1	Low	Minor onsite and negligible offsite impact to people or environment.
2	Moderate	Considerable potential onsite impact to people or environment. Minor potential offsite impact.
3	High	Potential for major onsite or offsite impact to people or environment.

*Based upon classes defined in DOE Order 5481.1A.(3)

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 1 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
1.4 Charge fuel to dissolver	Dissolution of short-cooled fuel, with release of short-lived FP[a]	DOG[b] system not designed to accommodate short-lived FP and Lack of control of material charged to dissolvers resulting in charge of green fuel	Administrative control by procedure on material brought to PUREX Radiation monitor verification of fuel age Heat sensor in rail car carrier	Short-lived FP gases including, Kr, Xe, and I_2	1	2	11.2.1 11.3.2
	Spilling canister of fuel elements	Crane hook or canister fails (bottom breaks out), or Operator error causes fall, or Brake on crane fails	Crane on preventative maintenance Training of crane operator Procedures		2	0	11.2.4 11.3.5
	High personnel exposure	Charging of fuel in presence of personnel	Administrative control dispatcher check-in/check-out		1	1	11.3.5
2.0 Fuel decladding and dissolution	Discharge from tank into leaking coils and then to cribs or pond: o Discharge to crib from TK-A3, -B3, -C3, and -E1 o Discharge to pond from TK-E3, -E5, -D3, -D4, and -D5	Coil failure, and Suckback occurrence, and Failure to divert to catch basin	Coils normally at positive pressure Bypasses on certain cooling water shut-off valves to maintain pressure Radiation monitor causes automatic diversion of contaminated water to retention basin 216-A-42	Dissolver solution released to crib Dissolver/metathesis decladding solution released in cooling pond	2 1	0 1	11.3.4 11.3.5
	Jetting of dissolver solution to an open nozzle or line in process cell	Crane operator failure complete assignment, or Operator failure, or Gasket failure	Administrative procedures including lock and tag system Current cell jumper position log	Doclad solution aerosol to A-E cells Metathesis solution to A-E cells Dissolver fuel solution aerosol to A-E cells	2	0	11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 2 of 16)

/stem/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Control-ling OSR
					Probability	Consequence	
Fuel decladding and dissolution (Continued)	NH_3 or H_2 explosion in dissolver or DOG	Radiolytic or chemical generation of H_2 or chemical generation of NH_3	Ensure presence of ammonium nitrate in decladding solution H_2 and NH_3 monitor, administrative controls		1	1	11.2.2 11.2.3 11.3.5
Metathesis process	U metal fire in dissolver melts through dissolver releasing liquid in A, B, or C Cell	Metal heating due to FP decay and exothermic U reactions, and Makeup water omitted, and U melting point higher than dissolver stainless steel	Long time span required to sufficiently heat fuel to cause fire Administrative control on length of time fuel can remain uncovered	FP aerosol in DOG system and in A, B, or C Cells	1	2	11.2.4 11.3.5
7 Centrifugation of decladding waste solution	A missile generated by failure of centrifuge penetrates a line or vessel containing dissolver solution (such as TK-E6)	Unbalanced operation of centrifuge	Safety margin in centrifuge design	Dissolver solution aerosol filling E cell	1	0	11.3.5
8 Liquid overflow to TK-E5	NH_3 explosion in TK-E5	NH_3 is a normal product of operations	Water vapor barrier TK-E5	Dissolver solution aerosol filling E cell	1	0	11.2.2
11 Centrifugation of decladding waste to metathesis	A missile generated by failure of centrifuge penetrates a line or vessel containing dissolver solution (such as TK-E6)	Unbalanced operation of centrifuge	Safety margin in centrifuge design	Dissolver solution aerosol, filling E cell	1	0	11.3.5
Fuel dissolution	Criticality in a dissolver containing dissolved fuel	Addition of metathesis solutions from TK-D1 causing precipitation of dissolver solution and Double charge of fuel or miscommunication as to enrichment of fuel in dissolver	Administrative control that includes a supervised lock and tag for KOH addition to dissolver Administrative controls of dissolver charge as a function of fuel enrichment	Dissolver aerosol and volatile short half-life isotopes to DOG system	1	1	11.2.4 11.3.1 11.3.5
	U metal fire in dissolver		See Analogous Accident in Section 2.2[c]		1	2	11.2.4 11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 3 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
2.4 Jet solution to holding tanks TK-D3, -D4, and -D5	Criticality due to KOH-triggered precipitation of feed solution	Addition to methathesis solution to tank from the dissolver or TK-D2 while tank is full or nearly full and Excess plutonium inventory in affected tanks	Administrative procedure control Supervised lock and tag system	Dissolver aerosol and short half-life isotopes and gases including Kr, Xe, and I_2	1	1	11.3.5 11.3.1
	Rupture of tanks or connecting piping due to cell block impact	See Analogous Accident in Section 14.0		Dissolved metal solution aerosol in D cell			11.3.5
2.4 Feed solution holding tanks TK-D3, -D4, and -D5	H_2 explosion in tank atmosphere	Radiolytic H_2 formation, and Lack of air dilution, and Source of ignition	Administrative control to provide air dilution to tank Lack of readily identifiable ignition source Explosion needs to blow off vessel covers	Dissolved metal solution aerosol in D cell	1	1	11.3.3 11.2.3 11.3.5
2.5 Feed makeup tank TK-E6	H_2 explosion in tank atmosphere	Radiolytic H_2 formation, and Lack of air dilution, and Source of ignition	Administrative control to provide air dilution to tank Lack of readily identifiable ignition source Explosion needs to blow off vessel covers	Dissolved metal solution aerosol in D cell	1	1	11.3.3 11.2.3 11.3.5
	Criticality in TK-E6	TK-E6 not geometrically safe, and An excessive amount of Pu jetted from TK-L11 or N cell to TK-E6 because operator error or erroneous laboratory results, and Both agitators fail in TK-E6	Procedures limiting transfers of TK-L11 or N cell to TK-E6 plus a required laboratory analysis Criticality limits are such as to require at least double batching to create a criticality $Cd(NO_3)_2$ required for transfer of concentrated Pu solution from TK-L11 or N cell to TK-E6 Redundant agitators in TK-E6	Dissolver aerosol and short half-life isotopes	1	1	11.3.1 11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 4 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
2.8 Centrifugation of either the decladding waste or metathesis solution	A missile generated by the failure of the centrifuge penetrates a line or vessel containing dissolver solution (such as TK-E6)	Unbalanced operation of the centrifuge	Safety in centrifuge design	Dissolver solution aerosol filling E cell	1	0	11.3.5
3.1 NH_3 scrubbing with H_2O during declad[2] (T-A3-3, B3-3, C3-3)	NH_3 explosion in electric heater Azide explosion in silver reactor	Failure to turn on condenser cooling water and scrubber water, and Low silver reactor temperature allows formation of explosive compound, and NH_3 reactions with silver forms potentially explosive compounds, and Restoration of heat to silver reactor	Alarm sounds if dissolver is in declad stage with low H_2O flow to NH_3 scrubber NH_3 detector installed in the line to alarm on high NH_3 flow past NH_3 scrubber Silver reactor maintained at operating temperature during all phases of dissolver operations to preclude formation of azides Low temperature alarm on silver reactor	^{131}I from silver reactor out through stack	1	1	11.2.2 11.3.5
5.0 Backcycle waste concentrator	Leakage from TK-F10 or TK-J1 to cooling coils and then to cooling pond	See Analogous Accident in Section 2.0		Backcycle waste released to cooling pond			11.3.4 11.3.5
5.2 Backcycle waste concentrator E-H4	Red oil explosion blows aerosol into cell	Excessive organic in concentrator caused by flooding or defective interface controller on either 2A or 2D column, or the operation of the TK-F10 agitator and Excessive temperatures in concentrator caused by failure of steam PRV and steam pop-off valve	Administrative and operational control of columns Control of steam pressures including PRV and pop-offs on the steam supply Temperature alarms and interlock on concentrator Explosion must be sufficient to blow seal pot	Aerosol formation of backcycle solutions in H cell	1	0	11.2.5 11.3.5 11.2.6
6.0 Decontamination and partitioning cycle	Dropping of cell cover block into H cell and rupture of feed tank or connecting line	See Analogous Accident in Section 14.0		HA column feed aerosol filling H cell			11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 5 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
6.1 HA column feed tank (TK-H1)	H_2 explosion in tank atmosphere	See Analogous Accident in Section 2.4		Dissolved feed aerosol in H cell			11.3.5 11.3.3 11.2.3
	Discharge from tank into leaking cooling coils and then to cooling ponds	See Analogous Accident in Section 2.0		Feed solution released in cooling pond			
6.3 Partitioning feed tank (TK-J3)	Organic fire in H or J cell sump from HAP line leakage/misconnection, TK-J3, or 1BXF feed line	Loss of organic to cell sump and A source of ignition (the probability of fire can be enhanced by overheating of solvent)	Administrative control over the materials in sumps A lack of a readily identifiable source of ignition Two high-level alarms in sump Light water fire suppression in H, J, K, G, and R cells	A Pu and FP aerosol filling H or J cell	1	0	11.2.5 11.3.5
6.4 Pu-U partitioning column	1BX column criticality	Excessive ferrous ions present in column and Excessive nitrite in column organic feed causing reflux	Procedures on means of recovery from loss of partitioning Laboratory analyses of both ferrous concentration and 1BX makeup Neutron monitors positioned along column	Short-lived FP offgasing to vessel vent system	1	1	11.3.1 11.3.5
7.0 Final Pu cycle	A solvent fire involving loaded Pu solvent	A solvent leak to the cell floor and Source of ignition	Administrative control over the materials collected in sumps Lack of readily identifiable source of ignition Fire fog system Low solvent inventory	Pu aerosol filling L cell	1	1	11.2.5
	Discharge from TK-L3 to cooling coil and then to cooling pond	See Analogous Accident in Section 2.0		Pu nitrate solution in cooling pond			11.3.4 11.3.5
	Criticality in L cell	Leak of Pu solution to floor from vessel or connecting piping, and	Procedure control of sump contents Product loadout monitors to detect gross leakage	Pu aerosol with volatile fission products in L cell	1	1	11.3.1 11.3.5 11.2.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 6 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
7.0 Final Pu cycle (Continued)	Criticality in L cell (continued)	The flooding of L cell, and No periodic cell flush or visual inspection	Level alarms Pu leak to cell floor forms slab configuration				
7.1 Pu cycle feed tank (TK-J5)	Criticality in J cell sump	Filling of J cell sump with solution from TK-J5 (2AF) and Presence of a precipitating agent in the sump	Sump monitor Administrative control over contents of sumps along with control of material added to J cell sump Operational depth of Pu on cell floor may not exceed stated limits	Pu aerosol plus FP gases from criticality in J cell	1	1	11.3.1 11.2.5 11.3.5
7.5 Air pulser to 3A column	Blowback into P&O[e] gallery to environment	Leak in pulser system in P&O gallery	Routine pressure checks of pulsers Filtered exhaust Glovebox around air pulser	3A column solution aerosol to P&O gallery and then to environment through white room stack	2	1	11.3.5 11.3.3
7.6 Air pulser to 3B column	Blowback into P&O gallery to environment	Leak in pulser system in P&O gallery	Routine pressure checks of pulsers Filtered exhaust Glovebox around air pulser	3A column solution aerosol to P&O gallery and then to environment through white room stack	2	1	11.3.5 11.3.3
7.7 Pu stripper (T-L6)	Pu concentration red oil explosion	Excessive organic in Pu stripper caused by 3B column interface controller failure and Excessive temperature in the vessel	Administrative control of column operations Control of the steam pressure with PRV and pop-off valves Temperature controls on vessel	Pu aerosol in L cell	1	0	11.2.5 11.3.5 11.2.6
	Suckback into P&O gallery from tube bundles	Failure of a tube bundle and Leak in P&O gallery	Routine pressure checks of tube bundles Pressurized air is an automatic backup for low steam pressure in tube bundle Filtered exhaust	Pu solution aerosol from TK-L6 in P&O gallery white room to white room stack	2	1	11.3.5 11.3.3
	Sample gallery ventilation HEPA[f] filter failure	Fire or Improper installation or Moisture or Damaged filter frames	Minimal combustible loading Small inventory available for release DOP[g] testing on installation and annually thereafter	Pu solution aerosol in samples	1	0	11.3.3 11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 7 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
7.7 Pu stripper (T-L6) (Continued)	Criticality in L cell	Slow leak from vessel to hot surface then agglomerating to slowly form critical mass (cell flooding will reduce mass required)	Pu accountability system Periodic cell flush Molarity and composition of solution inhibits evaporation to dryness. Cell sump design Routine visual inspection of cell Preventative maintenance	Volatile FP in L cell atmosphere	1	1	11.3.1 11.2.5 11.3.5
7.8 Pu product concentrator (E-L7-1)	Pu concentrator in red oil explosion	Excessive organic in Pu concentrator caused by flooding of 3B column and Excessive temperature in vessel	Administrative control of column operations Control of steam pressure with PRV and pop-off valves Temperature controls on vessel	Pu aerosol in L cell	1	0	11.2.5 11.2.6 11.3.5
	Blowback into P&O gallery from tube bundles	Failure of a tube bundle and Leak in P&O gallery	Routine pressure checks of tube bundles Aerosol formation required for spread of contamination Pressurized air is automatic backup to low steam pressure in tube bundle	Pu solution aerosol from E-L7-1 in P&O gallery white room to white room stack	2	1	11.3.5 11.3.3
	Criticality in L cell	Slow leak from vessel to hot surface then agglomerating to slowly form critical mass (cell flooding will reduce mass required), and No routine visual inspection of cell, and No preventative vessel maintenance, and No periodic cell flush-out	Pu accountability system for large leaks Vessel integrity	FP aerosol and gases in L cell	1	1	11.3.1 11.2.5 11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 8 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
7.13 PR room series HEPA filters to PR stack	Loss of filtration capability resulting in spread of Pu aerosol from stack	Aerosol formation in rooms or vessels and Loss of filters efficiency caused by damaged filters/filter frames or incorrect installation	Two-stage HEPA filters in series PR room stack monitor DOP testing will be performed on HEPA filters upon installation and then annually	Pu aerosol	1	2	11.3.3 11.3.5
7.14 Product loadout to PR bottles from TK-L13-10	Spill of a PR can on either the dock or elevator thereby releasing Pu aerosol	Loose PR bottle lid, and Tipping of PR can, and No filtering of aerosol formed	Procedural control on how cans are handled Procedure instructing the bottle lids to be closed tightly Not performed routinely	Ground-level release Pu aerosol in building ventilation system or on loadout dock	2	1	11.3.5
8.0 Organic recovery system 1	Discharge from TK-G1, -G2, or -G8 into cooling pond	- See Analogous Accident in Section 2.0		Organic wash wastes released in cooling pond			11.3.4 11.3.5
	Major organic fire	Organic leakage to G cell and Ignition prior to removal	Administrative control of material in sumps Lack of ignition sources Firefighting system, including tunnel spray Large conflagration required to affect filters Redundant sump level alarms	Aerosol formation resulting from burning of solvent to main exhaust filters Zr-Nb and Ru	1	1	11.2.5 11.3.5
12.6 Final facility filtering (deep bed)	Fire in final filter	NH_4NO_3 and organic in filter and Source of ignition	Lack of ignition source All NH_3 streams processed separately and resulting NH_3 vent releases are discharged through a separate stack Concentration of NH_4NO_3 too low to support combustion	Aerosol of deep bed filter activity	0	2	11.3.3 11.2.2 11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 9 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
12.6A Final facility filtering (HEPA) (deep beds assumed to removed from service)	Fire in HEPA filter	NH_4NO_3 and organic in filter and Source of ignition	Lack of ignition source All NH_3 streams processed separately and resulting NH vent releases are discharged through a separate stack Concentration of NH_4NO_3 too low to support combustion Water seal on HEPA HEPA filters in series resulting in reduced activity and less NH_4NO_3 in final stage	Aerosol of HEPA filter activity (negligible)	1	2	11.3.3 11.2.2
	Sudden blockage of filter by water seal resulting in air reversal from canyon	Water seal provided to permit HEPA filter isolation	Administrative procedures	Canyon air dispersed into normally clean areas	2	1	11.3.5
12.6B	Smoke plugging of deep-bed filters	Solvent fire in canyon cell	Firefighting system, including tunnel spray		1	2	11.3.3 11.3.5 11.2.5
14.0 Waste concentration and treatment	Discharge from TK-F18 or -F26 into cooling coils to 216-A-25 pond	See Analogous Accident in Section 2.0		IWW waste released in cooling pond			11.3.4 11.3.5
	Discharge from TK-F16, -F18, and -F15 to crib	See Analogous Accident in Section 2.0		IWW waste released to 216-A-30 crib			11.3.4 11.3.5
	Jetting of IWW solution to an open nozzle or line in F cell	Crane operator failure to complete assignment, or Process operator error, or Gasket failure	Administrative procedures which includes lock and tag system Current cell jumper position log	An aerosol of IWW filling F cell	2	0	11.3.3 11.3.5
	Dropping of cell block into F cell and rupture of process vessel or line	Crane/block bale failure or Operator error	Crane braking system Cell and block geometry Administrative controls used to govern cell cover block movement	Aerosol of IWW filling F cell	1	0	11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 10 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
14.1 Waste concentration (TK-F6, -F26, -F15, or -F16)	Release in P&O gallery from suckback from jets Suckback from leaky tube bundles Blowback in instrumentation lines	Failure of jet gang valves to blow steam out of line and Instrument line leakage	Maintenance on jet gang valves Testing of tube bundles at every plant outage Pressurized air is automatic backup for low steam pressure in tube bundle	IWW solution aerosol released to P&O gallery	2	1	11.3.3 11.3.5
14.2 Waste concentration (TK-F6-1)	Red oil explosion spreads contamination in F cell	Excessive organic in IWW concentrator (HA column flood) and Excessive temperature due to failure of the PRV and SRV on steam line and High temperature steam shut off	Administrative control of column operation IWW concentrator temperature control Decant concentrator feed tank to remove organic Organic detection system on feed tank	Aerosol of IWW filling F cell	1	0	11.2.5 11.2.6 11.3.5
14.5 Waste denitration (TK-F15)	H_2 explosion in tank atmosphere	Excessive radiolytic H_2. and Lack of air dilution and Source of ignition	Vessel vent system provides air dilution to tank atmosphere Source of ignition not readily identifiable	IWW aerosol formation in F cell	1	1	11.2.3 11.3.3 11.3.5
14.6 Sump waste receiver (TK-F18) (TK-F18)	Criticality in TK-F18	Collecting of excess Pu containing solution leaked to sumps and Failure to acidify TK-F18	Control of leaks by maintenance Procedures for not adding caustic to TK-F18 until laboratory sample results are known Sample taken of sump before shipment to TK-F18 Sample taken in TK-F18	FP aerosol and gases in TK-F18	1	1	11.3.1 11.2.5 11.3.5
14.7 Waste neutralization (TK-F16)	H_2 explosion in tank atmosphere	See Analogous Accident in Section 14.6					11.2.3 11.3.3 11.3.5
15.0 PuO_2 conversion	Nuclear criticality in N or L cell sumps	Transfer of solutions to full tanks (TK-M3, -M4, -M5, or -M6) and overflow to sump via -M12 and Leaking tank	Administrative controls including lock and tag, accountability for transfers, and fissile mass and depth limits Proper operating procedures	Aerosols of Pu solutions plus short-lived FP gases into N cell glovebox and L cell	1	1	11.3.1 11.2.5 11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 11 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
15.0 PuO_2 conversion (continued)			Criticality drain in N cell Liquid-level instruments and alarms on tanks and sumps				
	Spill sample from Pu nitrate storage tank TK-M3, -M4, -M5, and -M6	Procedural error	Administrative controls Proper operating procedures Filtration of glovebox air	Small amount of Pu solution in work area Aerosols of Pu solution into N cell air	2	0	11.3.5 11.3.3
	Criticality in N or H cell vessel sumps and piping	Low acid concentration permits Pu polymer formation resulting in precipitation and hold up of Pu at concentration exceeding 450 g/L	Closed loop cooling of all vessels with 1.2M acid coolant No water addition normally possible Dilution normally available only with 1.2M acid condensates 1M acid PUREX operating procedures nuclear materials accountability Fissile mass within single vessel insufficient for criticality	Aerosols of Pu solutions/sludges plus short-lived FP gases into VOG and N cell glovebox	1	2	11.3.1 11.3.5
	Fire in oxide conversion facility	Accumulation of combustible solvent or solid waste and Source of ignition	Wet-pipe sprinkler in N cell. Administrative controls on accumulation of combustible material. Design of cell doors minimizes possibility of solvent reaching N cell	Aerosol of PuO_2	1	2	11.2.5 11.3.5
15.3 Prereduction	Pressurization of tank, forcing overflow of Pu solution to glovebox and into instrument lines	H_2O_2 added too fast, or H_2O_2 is too concentrated or Insufficient agitation in tank and Leaking instrument lines	Administrative controls including analysis of H_2O_2 composition HEPA filtration of glovebox and room Proper operating procedures	Aerosols of Pu solution to N cell operating area via instrument lines	2	0	11.2.9 11.3.5 11.3.3

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 12 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
15.5 Vacuum drum filter	Nuclear criticality at or near drum filter	Leak/spill of solids or liquid followed by evaporation, permitting Pu buildup in glovebox or on outside of equipment	Administrative controls of inventory solution depth Housekeeping and periodic inspections of TV and periscope Proper operating procedures Filtration of glovebox atmosphere	Aerosols of Pu solutions/sludges, plus short-lived FP gases into VOG and N cell gloveboxes	1	2	11.3.1 11.3.5 11.3.3
	Nuclear criticality at or near calciner	Bridging of filter cake or Filter cake fails to fall into calciner, or Calcine backs up to vacuum drum filter	Adequate design Proper operating procedures, including vital observations by television monitors Manual assist available if needed Administrative controls Filtration of air in vessel systems, in cell, and in canyon air	Aerosols of Pu sludge and powder plus short-lived FP gases into N cell glovebox	1	2	11.3.1 11.3.5 11.3.3
15.6 Calciner	Fire or red oil explosion in calcincer	Excessive organic feed to calciner	Normally good operation of 3B column and/or early detection of 3B column problems Normally good operation of steam stripping E-L6 TK-L9 and M cell samples analyzed for organic Proper operating procedures, including analysis for phosphorus in TK-L9 samples Filtration of VOG and canyon air	PuO_2 dust into N cell glovebox	1	0	11.2.9 11.2.5 11.2.6 11.3.5 11.3.3
15.7.1 Rework material load-in	Spill outside hood during transfer	Operator fails while hand carrying or Transfer containment seal fails	Double plastic bag and taped lid on can or closed cap on jar	PuO_2, $Pu(C_2O_4)_2$, floor sweep	3	0	11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 13 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
15.7.1 Rework material load-in (continued)	Overpressurization of container resulting in rupture and spill	Damp material sealed in can or unvented bottle	Double plastic bag, material outside of glovebox for very short time	PuO_2, $Pu(C_2O_4)_2$, floor sump	2	0	11.2.9
15.7.4 Dissolvers	Spill in glovebox	Overcharging of HNO_3	Operating procedures, Pyrex glass allows visibility of calibrated levels	Acid solution and solids spill into glovebox	2	0	11.3.5
		Pressurization of dissolver	Pressure relief device on dissolver	Aerosol liquid and solids in glovebox	2	0	
		Leaking containers	Administrative control		2	0	
		Excessive air sparge and Loss of vacuum	Restriction orifices, administrative control	Aerosol, liquid, and solids in glovebox Powder in glovebox Pu solutions on floor	2 2 2	0 0 0	
		Overbatching any tank	Sump alarm Vacuum trap receives any overflow; automatic shut off of vacuum when liquid detected in vacuum trap				
15.7.5 Rework, transfer, filter, and receiver	Spill from filter into glovebox	Component mechanical failure	Sump alarm	Liquid spill in glovebox	2	0	11.3.5
15.7.6 Dryer	Fire in glovebox	Electrical malfunction of dryer	Administrative procedures, Halon fire system High temperature cut-off on dryer	Aerosol, liquid, and solids in glovebox	2	0	11.3.3 11.3.5
15.7.7 Rework storage	Leak	TK-N53 failure or Line failure or Pump leak	Catch pan with level detection and alarm vacuum transfer	Liquid $Pu(NO_3)_4$ in acid solution on pipe chase floor	2	0	11.3.5 11.2.5
	Undesirable chemical reaction Temperature and pressure excursions	Wrong chemical substituted for AMN (also applies to rework receiver tank)	Administrative procedure		2	0	11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 14 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
15.7.8 Transfer to TK-L11	Spill in canyon cell	Component mechanical failure	Administrative procedure coordination between control rooms and operators Welded piping Sump alarm	WF error, 15 L of recycle Pu nitrate 90 L of Pu nitrate	2	0	11.3.5
N-6 glovebox	Hand injury due to puncture wound	Glove is cut by broken glass or sharp object	Operating procedures, training All glass vessels covered with screen	Surface contamination in glovebox	2	0	11.3.5
	Glove puncture during hood cleanup	Sharp object in hood	Good housekeeping, training	Surface contamination in glovebox	2	0	11.3.5
	Fire on floor of glovebox	Combustible material in glovebox (rags, plastic, etc.)	Good housekeeping	Spills from process such dissolver feed	2	0	11.3.5
15.8 Rotary scalper screen	Nuclear criticality in rotary scalper screen	Design limit of 3 g/cm^3 for Pu in solid form exceeded	Tests show 3 g/cm^3 is higher than can be obtained The equipment is not normally capable of producing solids of this high density Administrative control of inventory (would need too long to fill calciner blender)	Short-lived FP and noble gases released into VOG and N cell glovebox	1	2	11.3.1 11.3.5
15.9 PuO_2 blender	PuO_2 blender leakage resulting in powder release	Drop blender, breaking slide valve and glovebox enclosure or Incomplete closure of slide valve	Proper operating procedures Adequate design of blender Slide valves interlocked to prevent removal of blender while they are open	PuO_2 dust in air in glovebox and operating area	1	0	11.3.5
15.10 Product loadout	PuO_2 container rupture from excessive pressure	Excess moisture or nitrate in product or Excess moisture in glovebox air	Undried PuO_2 chunks are prescreened and removed Proper operating procedures Calciner temperature and screw speed continuously monitored	PuO_2 dust in glovebox air and operating area air	2	0	11.2.9 11.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 15 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
15.10 Product loadout (continued)			Dual dryers of glovebox air Dew point of glovebox air is monitored Results of loss-on-ignition are normally known before covering Filtration of glovebox, cell, and canyon air				
	PuO_2 shipping container and product cans crushed or spilled	Accident while handling or loading or Improper container closure	Proper operating procedures Administrative controls Adequate design of containers Filtration of glovebox and operating area air	PuO_2 dust in air in glovebox and operating area	1	0	B1.3.3 B1.3.5
15.13 Filtrate holding	Nuclear criticality in TK-E6	Excessive Pu to filtrate, or $H_2C_2O_4$ pump fails, or Insufficient $H_2C_2O_4$ in strike solution, or Inadequate filtration of Pu, or Failure of TK-N7 agitator, or Inadequate heating in TK-N13 leaves $Pu(C_2O_4)_2$ in -N15 or -N16 for recycle to -E6	Proper operating procedures Volume and weight of additions to TK-N35 or -N36 are measured, and composition is analyzed Administrative controls Inspection of solutions TK-N7 and -N11 Filtration of VOS and canyon air N-13 is designed for adequate residence time	Aerosols of Pu solutions plus short-lived FP gases into VOS and ventilation air	1	1	B1.3.1 B1.3.5

TABLE 9-3. PUREX Preliminary Hazards Analysis. (Sheet 16 of 16)

System/subsystem	Potential accidents	Causal factors	Barriers	Characteristic source term	Category		Controlling OSR
					Probability	Consequence	
15.17 HEPA filtration	Loss of glovebox primary HEPA	Fire or Improper installation or Moisture or Damaged HEPA frames	Halon Fire Suppression System. HEPA DOP tested after installation. Glovebox supplied with air Two HEPAs in series downstream	PuO_2 dust into glovebox exhaust system	1	0	11.3.3 11.3.5

[a]Fission product.
[b]Dissolver offgas.
[c]Probabilities and consequences are the same for analogous accidents.
[d]Pressure relief valve.
[e]Pipe and operating.
[f]High-efficiency particulate air.
[g]Dioctyl phthalate.
[h]Steam relief valve.
[i]Vessel offgas.
[j]Aluminum nitrate nonahydrate.
[k]Weight factor.

9.1.4 Fault Tree Analysis

A PHA is developed from individual cases or hazards to a general conclusion of how the facility is affected; therefore, evaluation of facility risk depends upon accurate identification of all potential hazards associated with the process analyzed. An additional system for methodical identification of potential hazards is the use of fault tree analysis. A fault tree is a graphic model of the various combinations of system or component failures that will result in the occurrence of a specific or "top" event. The top event of the PUREX fault tree (Fig. D-1) is specified as "above normal release of radioactivity from the PUREX operation". The fault tree branches down to all failures that may result in this event. The top tier of the fault tree identifies all potential pathways of releases, such as the product handling and removal (PR) room stack, main exhaust stack, cooling water pond, P&O gallery exhaust, HEPA filter malfunctions, and outside spills. Lower tiers identify failures which could contribute to releases via these pathways. In most cases, the failures thus identified are those evaluated in the PHA. Both methods identify similar potential hazards, confirming the position that all major risk contributors have been identified.

Accidents identified as major risk contributors are analyzed in detailed scenarios. Scenario development includes determination of source terms, releases, and potential doses to onsite and offsite personnel. In addition, the fault trees are quantified to provide an estimate of probability of occurrence of accidents that may result in an "above normal release of radioactivity". The PUREX fault trees are provided in Appendix D.

9.1.5 Major Risk Contributors

9.1.5.1 Breach of Containment. Containment is taken to mean an absolute barrier. Containment in a fuels reprocessing facility is to be distinguished from containment in a nuclear power plant where an absolute physical structure (containment vessel) is often implied. At PUREX, containment commonly is not absolute due to vents and sampling penetrations. Examples of containment barriers are the tanks, piping, tube bundles, and product containers. Breach of containment is defined as transport of radioactive contamination through a containment barrier. Barrier failure may occur as the result of age, deterioration, design flaws, corrosion, or mechanical disruption.

The dominant risk contributor in this category is a failure of the TK-D5 cooling coil, resulting in a release to the 216-A-25 cooling water pond.

9.1.5.2 Confinement Barriers Fail, are Compromised, or are Circumvented. Confinement barriers are those which interact with various systems. Examples include offgas filter systems, liquid waste monitoring and diversion systems, and ventilation systems which direct transport of radioactive contaminants. Confinement barriers fail, are compromised, or are circumvented as a result of system malfunction, operator error, or operating

conditions. Using an offgas filter as an example, failure would consist of disruption of the filter media with resultant loss of decontamination efficiency. Release of inadequately filtered exhaust may exceed specified limits.

The dominant risk contributors in this category are:

1. Dissolution of short cooled fuel

2. A P&O gallery release involving FP or plutonium.

9.1.5.3 Uncontrolled Chemical Reactions. Historically, unusual occurrences which fall under this definition result from loss of control of normal processes or unwanted chemical reactions (nonradioactive chemicals). In the PUREX and plutonium denitration processes, a number of conditions require constant control. These include volatile chemical reactions and byproducts, large inventories of high-potential energy chemicals, and segregation and isolation of incompatible chemicals.

The dominant risk contributors in this category are:

1. Fire involving solvent loaded with FP or plutonium

2. Silver reactor explosion

3. Hydrogen explosion in a dissolver

4. Uranium metal fire in a dissolver

5. Fire in the plutonium oxide production facility.

9.1.5.4 Nuclear Excursion. Although the presence of plutonium provides a potential for a nuclear chain reaction, nuclear criticality has not occurred at PUREX. Prevention of criticality is based on a policy requiring that at least two unlikely, independent, and concurrent equipment failures or operating errors must occur before a criticality accident is possible in those parts of the plant where heavy shielding would mitigate the personnel dose from a criticality event to an acceptable level. In the absence of such shielding, at least three failures or changes must occur before a criticality accident is possible. Violations of this policy, which allow one of the independent failures or changes to occur, are considered to be unusual occurrences or offnormal events, depending on the severity of the incident.

The dominant risk contributors in this category are:

1. 1BX column criticality

2. Criticality in the plutonium oxide production facility.

9413219.0966

9.1.5.5 Extrinsic Occurrences Affecting Plant Operation. These are defined as those event sequences occurring external to the plant and its processes, but affecting plant containment and confinement barriers and continuity of plant operations. Two classes of occurrences are considered: (1) loss or impairment of utilities and services, and (2) natural forces events. In general, interruption of utilities and services is considered to be one consequence of severe natural forces events.

The dominant risk contributors in this category are:

1. Seismic event, either a Hanford Regional Historical Earthquake (HRHE) or Safe Shutdown Earthquake (SSE)

2. Tornado.

9.1.5.6 Industrial Hazards. The dominant risk contributor in this category is a severe overexposure in canyon during loading of fuel elements.

9.1.6 Limitations

There are conceivable accidents which are not included because they represent a very low risk relative to these identified in the PHA. In addition, the PHA and fault tree analysis provides an analysis only of the risk contributors which occur during process operations, as this configuration provides the maximum source term. Although several accident scenarios were maximized by postulating process delays or short-term shutdown, accidents which may occur during shutdown and extended maintenance or accidents associated with interfacing facilities outside the separations building were not explicitly considered in this analysis. Uncontrolled releases from the following areas are not being considered because of low risk potential:

- R or U cells
- Lab rooms and hoods
- Ammonia scrubber failure
- Sample gallery rooms.

This analysis recognizes that radiological release may take place by liquid (aquatic) pathways, however, this mode is not being treated. Atmospheric pathways, involving possible lung ingestion and cloud immersion, are the primary contributors to dose consequence.

Sabotage and materials diversion has been excluded from the analysis because of the stringent controls and isolation from the public.

9.1.7 Consequence Evaluation

For this analysis, consequences are evaluated only for direct risks, or dose to the affected population. It must be recognized however, that additional, indirect risks and losses may result from accidents of the nature of those analyzed. Such indirect consequences may take the form of property damage or loss, environmental degradation, public or political reaction, or lost performance. The evaluation of such indirect consequences is outside the scope of this document.

Dose consequences resulting from a given radionuclide release are directly dependent on proximity of personnel to the release point, and the dose factor of the specific radionuclide released.

The onsite population distribution used in the evaluation of PUREX Plant accidents was developed from information provided by site contractor emergency coordinators, paycheck distribution records, and other references. The site population is subject to constant change and any census figures should be regarded as approximate. Hanford Site population distribution is given in Table 9-4 and organized by compass direction and distance, as shown on Figures 9-1 and 9-2. Onsite population does not include the PUREX Plant workforce. Offsite population distribution was taken from Reference 9-4 and is given in Table 9-5, again organized by compass heading and distance.

Dose consequences resulting from acute, or accidental, radionuclide releases were determined using the DACRIN(5), PABLM(6), and SUBDOSA(7) computer codes, as well as some HP-41CV calculator programs which facilitate preparation of input for the computer codes.

9.1.7.1 DACRIN. The computer code DACRIN was used to compute inhalation doses for all accidents. DACRIN inputs include particle diameter (1.0 μm assumed), breathing rate (350 cc/s for acute, and 250 cc/s for chronic assumed), wind speed and downwind distance (this permits decay of short-lived nuclides in flight), intake time (assumed to be the duration of the accident), total dose time (1 yr and 50 yr were assumed). The DACRIN code can compute integrated exposures to maximum individuals, but does so without considering ground deposition loss or building wake effects. For this reason, the HP-41CV programs "X/Q" - Rev 4, and "X/Q" - Pasquill, were employed. The ground deposition speed was 0 cm/s for inert gases, 0.1 cm/s for most particulates and 1.0 cm/s for iodine. These are also the values used internally by PABLM to determine ground deposition amount. Building wake effects were included for ground-level releases only (450 m^2 was the effective building cross-sectional area). In addition, the HP-41CV programs computed sector-averaged X/Q values which are weighted by the populations present in each sector segment, and totaled for all accidents except the criticalities. For the criticalities, the population weighted X/Q values at each distance were used separately, and dose results totaled. The DACRIN code uses the ICRP Lung Model and computes organ dose using ICRP Publication 2 methodology for each input nuclide and requested organ.

9413219.0968

TABLE 9-4. Hanford Site Population* Distribution from the PUREX Plant.

Compass heading	Range (ft)											Range (mi)													
	0 to 600	600 to 1,200	1,200 to 1,800	1,800 to 2,400	2,400 to 3,000	3,000 to 3,600	3,600 to 4,200	4,200 to 4,800	4,800 to 5,400	5,400 to 6,000	6,000 ft to 3 mi	3 to 4	4 to 5	5 to 6	6 to 7	7 to 8	8 to 9	9 to 10	10 to 11	11 to 12	12 to 13	13 to 14	14 to 15	15 to 16	16 to 17
North																									
NNE			3																						
NE			107																						
ENE																									
East																									
ESE																			4,613	6,336					
SE																			1,436						2,777
SSE																									
South																									
SSW																									
SW																									
WSW									290																
West								260	630		56		217												
WNW								129		300		8	1,150												
NW						38																			
NNW															55										

*Excluding personnel working within the PUREX Plant.

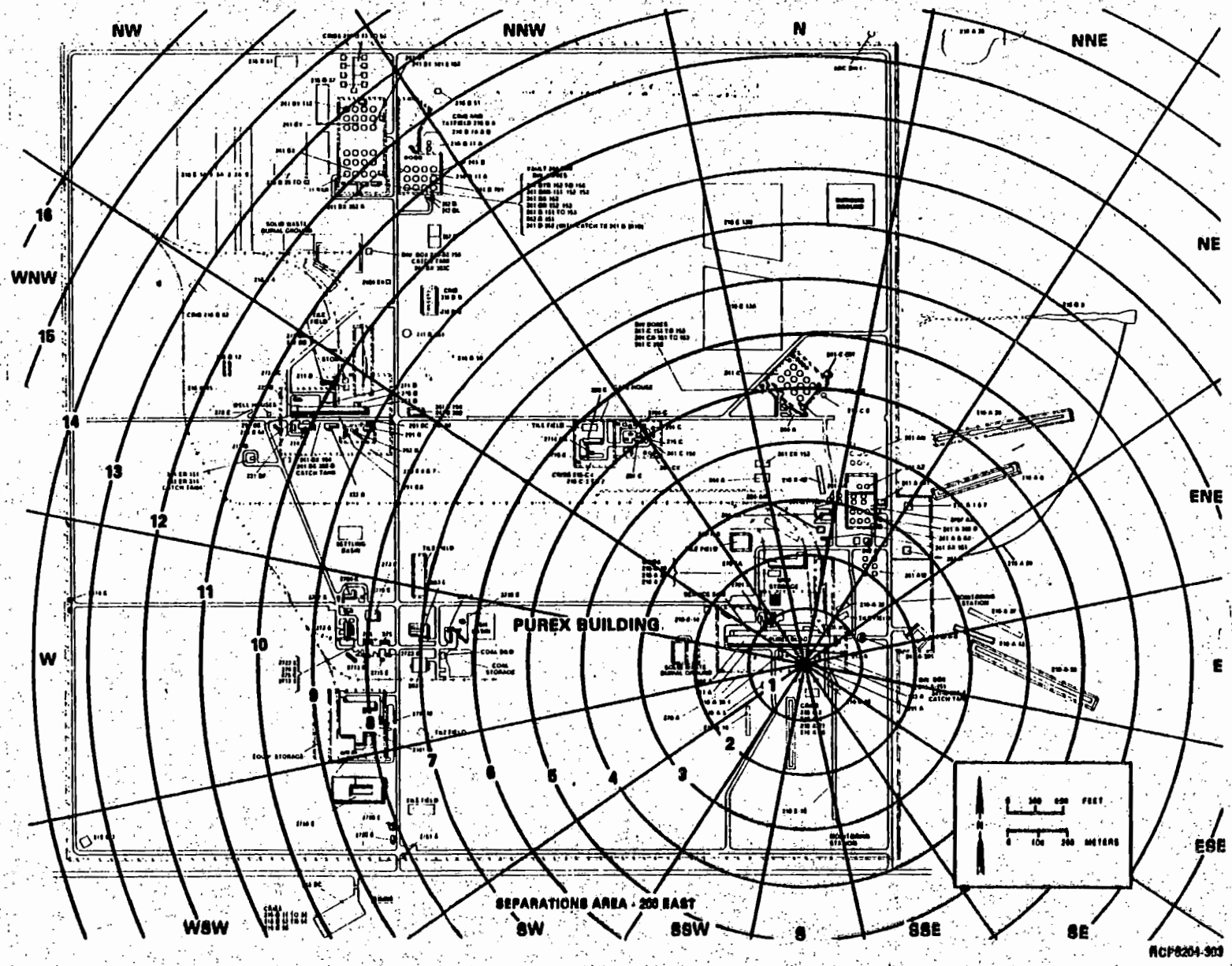


FIGURE 9-1. 200 East Population Grid.

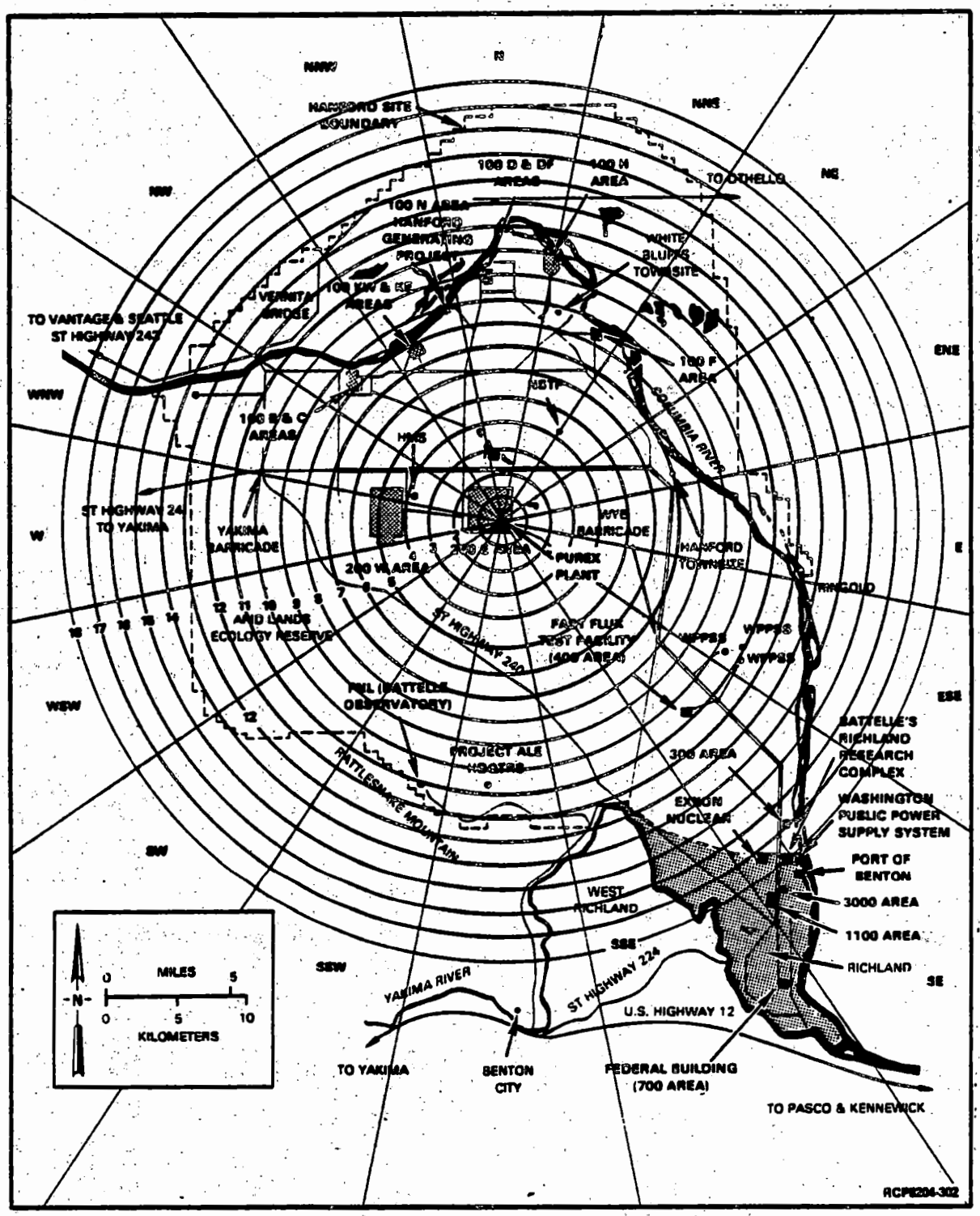


FIGURE 9-2. Hanford Site Population Grid.

TABLE 9-5. Offsite Distribution of Population Within 50-mi Radius of the 200 Area Hanford Meteorological Tower by Population Grid Sector for 1980.

Compass direction	Range (mi)					
	0 to 10	10 to 20	20 to 30	30 to 40	40 to 50	Totals
North	0	174	1,124	772	1,957	4,027
NNE	0	92	656	5,547	14,822	21,117
NE	0	262	5,930	2,963	596	9,751
ENE	0	235	773	2,366	435	3,809
East	0	340	1,329	1,659	588	3,916
ESE	0	283	1,374	230	652	2,539
SE	0	6,757	48,661	50,519	3,474	109,411
SSE	0	1,997	13,161	2,717	5,218	23,093
South	0	1,532	1,489	195	1,799	5,015
SSW	0	905	5,283	652	129	6,969
SW	0	1,190	19,786	2,182	459	23,617
WSW	0	1,840	5,063	15,088	4,573	26,569
West	32	648	949	6,874	78,635	87,138
WNW	73	444	802	833	2,833	4,985
NW	0	555	398	493	1,454	2,900
NNW	0	246	456	864	4,521	6,087
Totals	110	17,500	107,234	93,954	122,145	340,943

9.1.7.2 PABLM. The computer code PABLM is used to determine acute doses from ingestion of food products that contain radionuclides and external exposure to radionuclides deposited on the ground. Standardized consumption rates and other PABLM input parameters were used (PNL-3777), for maximum individual and population dose calculations. The first-year dose was taken as the 1-yr dose committment from 1 yr to intake following the release. The 50-yr dose committment was taken as the 50-yr cumulative dose following 50 yr of continuous farming. The PABLM code assumes the initial soil contamination is only removed by radioactive decay. Foliar deposition is included for first-year doses.

9.1.7.3 SUBDOSA and BIVAR. The computer code SUBDOSA was used to compute external gamma and beta dose from the passing radioactive cloud. The SUBDOSA code used dose rate factors for finite plumes as generated by the program BIVAR code. These external doses are significant for the short-cooled fuel scenario and the postulated criticalities due to the large quantities of inert gases released. Gamma doses at a 5-cm tissue depth were used to determine the external dose contribution to various internal organs.

Sector averaging of population dose was accomplished by computing effective external dose factors from SUBDOSA results and multiplying by the population-weighted X/Q values.

9.2 ANALYSIS OF PUREX ACCIDENT SCENARIOS

This section is concerned with developing scenarios for the accidents identified in the PHA and the potential consequences of the occurrences. Detailed accident scenarios, quantifications, and estimated source terms are presented in this section. Source terms were calculated based on a format of "more likely" and "worst case" assumptions. The "more likely" assumptions reflect the effort by the review team to represent the true severity of a given accident. The "worst case" assumptions reflect the most conservative approach possible. All releases and, therefore, dose calculations, are based on the "worst case" assumptions. Releases were also developed, in some cases, for both 12% ^{240}Pu fuel (2 to 5 yr out of the reactor) and 6% ^{240}Pu fuel (180 d out of the reactor). The 12% fuel resulted in higher dose consequences, however, and was assumed to be the source where appropriate. The 6% releases are provided for information. Mark IV fuel embodies more curies of actinides per metric ton uranium than does Mark IA and the Mark IV fuel charge to a dissolver is more than twice the weight of the Mark IA charge. On the other hand, Mark IA embodies more curies of FP than does Mark IV fuel. The particular fuel or fuel mixture for a worst-case depends on the circumstances of the particular scenario. Accidents whose consequences were investigated and found to be minor are developed as fault trees only. Probabilistic analyses, scenarios, and releases were not developed. It should be noted that the estimated frequency of occurrences provided for these accidents are "order of magnitude" estimates. Appendix D provides additional information on the quantification of fault trees and the expected precision of the results.

9.2.1 Uncontrolled Radiation Release Due to External Phenomena

In preparation for the resumption of PUREX Plant operations, evaluations of the natural forces resistance for PUREX Plant structures, safety systems, and vital equipment were conducted. The facilities which were evaluated are listed below:

- The 202-A building (main PUREX processing building) canyon
- The 202-A building east crane maintenance platform (ECMP), on the east end of the canyon
- The 202-A building R cell for final solvent cleanup and storage, on the north side of the canyon near the west end
- The 202-A building service annex, or noncanyon portion
- The 291-A exhaust ventilation system for the 202-A canyon, consisting of: reinforced concrete filter cells below grade; underground reinforced concrete air tunnels; a partially embedded reinforced concrete plenum; exposed fans, motors and metal ducts; and a 61-m high reinforced concrete stack
- PUREX vital equipment, components, utilities and services which present potential natural forces hazards to the safe confinement of radionuclides.

9.2.1.1 Seismic Resistance. The original PUREX design criteria specified that earthquake resistance be provided in accordance with Zone 2 regulations of the 1952 Uniform Building Code (UBC). These criteria required that facilities have the lateral resistance to withstand a 0.10-g static force.

Recent seismic analyses have evaluated the effect on PUREX facilities of both 0.25-g SSE ground motions and 0.10-g HRHE motions.[8-10] The work included the development of structural upgrades, with cost estimates, that would be required to withstand both earthquakes. A summary of the results of these studies is listed in Table 9-6. In addition, a seismic hazards survey was conducted to identify seismic-induced system and component failures which could affect the prevention or mitigation of the release of radioactive material. Several dispersion mechanisms were postulated to result from seismic-induced system failures:

- Uranium metal fire
- Solvent fire in H-J cell
- Solvent fire in R cell/N cell
- P&O gallery blowback
- Seismically-induced hydrogen explosions.

TABLE 9-6. Seismic Analysis Results.

Facility	HRHE	SSE
202-A building	Survives[a]	Fails[b]
ECMP	Fails	Fails
R cell	Survives	Survives
202-A annex	Survives	Fails
291-A filter cells, tunnels and plenum	Survives	Survives
291-A fans and motors	Survives	Survives
291-A belt drives and metal ducting	Survives	Fails
291-A stack	Survives	Fails
Utilities[c]	Fail	Fail

[a]Structure survives event in "as-in" condition.

[b]Structure may require upgrades to withstand event.

[c]Includes steam, water, and electrical services. May imply loss of power to ventilation exhaust fans.

9.2.1.2 Cost/Benefit Analysis. A risk analysis and cost benefit study evaluated the consequences of the preceding seismic-induced releases for various post-seismic plant conditions:(11)

- Canyon structure, survives/fails
- Ventilation system, survives/fails
- Cell, open/closed
- Mitigating features, success/failure.

Probabilities were established for each configuration based on expert appraisal of the system's seismic resistance. Each configuration, in turn, dictated a unique radionuclide release. Onsite and offsite doses were then

estimated for each configuration and a risk value established. Worst-case doses to the maximum offsite individual were used as a basis for selected plant upgrades based on comparison with the following criteria.

- Maximum Offsite Individual Dose, Annual
  - Low, less than 1 rem, whole body
  - Moderate, 1 to 5 rem, whole body
  - High, greater than 5 rem, whole body
- Maximum Offsite Individual Dose, Lifetime
  - 25 rem, whole body
  - 150 rem, bone
  - 75 rem, other organs.

The only source of uncontrolled seismic-induced radiation release from the PUREX Plant, without structural upgrading, with the potential to produce doses in the "high" category was the uranium metal fire, in conjunction with simultaneous structure failure of the facilities containment capability. All other dispersions resulted in doses in the "low" category.

For the onsite impact evaluation, the dose/risk value (dose consequence times probability) for each seismic-induced dispersion was evaluated against the following guidelines:

- Annual Dose to Individual
  - 5 rem, whole body
  - 15 rem, lung
  - 30 rem, bone.

Although RHO-CD-1550 identified the potential for onsite personnel in close proximity to PUREX to exceed the "LD-50" (sufficient to cause death in 50% of the cases) dose of 450 REM, the dose/risk for all dispersions was determined to be well below these guidelines, hence, no additional seismic upgrades were recommended, based on onsite risk.(12)

Studies indicated that the most cost beneficial method for reducing offsite dose resulting from an HRHE or SSE was to prevent the occurrence of two dispersion mechanisms, the uranium metal fire and the H-J cell solvent fire, rather than to ensure post-seismic containment or confinement.

In the case of the uranium metal fire, dispersion is precluded by ensuring drown tank capability after a seismic event. This in turn ensures that the contents of the dissolver may be kept covered by water, thereby preventing the occurrence of a fire. Several specific upgrades are provided to ensure this capability:

- Manually operated bypass valves which may be operated from outside the confinement wall through extension shafts (a missile guard is provided to protect the outside handles from damage)
- Swing check valves in the drown tank charging lines to prevent loss of fluid through a break in the seismically unprotected portion of the charging lines
- Diagonal bracing of the drown tank cradles and drown tank to cradle attachment brackets
- Missile and debris barrier above each drown tank and associated valve clusters
- Additional straps to firmly locate drown tank piping
- Additional supports for steam lines above the drown tanks
- Valve bonnet support for a nearby valve on the 20-in.-dia. raw water line adjacent to drown tank A-3A.

To preclude a H-J cell fire, the following upgrades have been provided:

- A foam-water sprinkler system capable of covering the entire floor of the H-J cell
- Engineered survivability of the fire suppression system during a seismic event
- A dedicated system supply package including water and pressurized nitrogen
- Automatic actuation and operation of the foam-water sprinkler system.

9.2.1.3 Wind and Tornado Resistance. Tornado design criteria were not specified in the 1952 UBC and so they were not included in the original design criteria.(13) Therefore, recent tornado analyses of the PUREX facilities have been conducted.(8) These studies analyzed the effects of credible tornado conditions for the Hanford Site. A summary of the evaluations is listed in Table 9-7.

TABLE 9-7. Wind and Tornado Resistance Evaluations.

Facility	DBT[a,b]
202-A canyon	Survives[c]
ECMP	Fails[d]
R cell	Survives
202-A annex	Not analyzed[e]
291-A ventilation system	Fails
291 stack	Fails
Utilities	Fail

[a]Design basis tornado.

[b]DBT as specified in Section 4.2.2.

[c]Survives event in "as-is-condition".

[d]Fails to survive event without prior structural upgrades.

[e]Not analyzed; aggregate release estimated to be less than 2.0 g plutonium and less than 5 g mixed FP.

Reference 14 concluded that although the confinement and/or containment capability of the PUREX Plant could not be assured following the pressure transients and missile damage which could result from a DBT, the resulting radiological consequences to the maximum exposed individual were low and well below guideline values. (Note: Due to its elevated release height, approximately 5,000 ft, the maximum exposed individual for a DBT is located offsite. For this reason, and due to the extreme difficulty of defining and modeling the local dispersion patterns of a tornado, no onsite doses were estimated for this event.) In addition, the probability of such a DBT is extremely low; therefore, no upgrades or modifications were provided to enhance the tornadic resistance of the PUREX Plant.

9.2.1.4 Frequency of Occurrence. Reference 14 estimates the recurrence frequency of a DBT at 1×10^{-7}/yr. From Appendix D the estimated probabilities of the seismic-induced dispersions described above are listed in Table 9-8.

9413219.0970

TABLE 9-8. Occurrence Probability of Seismically-Induced Dispersions.

Event	HRHE	SSE
Uranium metal fire	ε*	ε
Solvent fire in H-J cell	2.5×10^{-5}/yr	3.1×10^{-6}/yr
Solvent fire in R cell/N cell	2.2×10^{-6}/yr	1.5×10^{-7}/yr
P&O gallery blowback	2.2×10^{-3}/yr	8.4×10^{-5}/yr
H_2 explosion	1.2×10^{-5}/yr	2.8×10^{-6}/yr

*ε--much smaller than other probabilities involved in evaluation.

9413219.0979

9.2.2 Criticality in the Plutonium Oxide Production Facility

This scenario analyzes an accidental criticality occurring in an unshielded process area of the PUREX Plant. The postulated cause of the accident is an undetected accumulation of plutonium oxalate on the calciner glovebox floor.

The sequence of events resulting in this accident includes:

- Process failure resulting in spillage of plutonium oxalate to the glovebox floor. These failures may involve damage to the doctor blade, bridged chute to the calciner, stoppage of the calciner screw, loss of vacuum drum filter vacuum or filter pluggage

- Failure to detect either the loss of plutonium oxide product or the accumulation of the oxalate mass on the glovebox floor for a period of time sufficient to allow an accumulation of a critical mass. This requires failure of television and periscope monitoring and operator surveillance to note the problem over more than one shift change

- Addition of nominal 1 in. of reflection to the accumulated mass of oxalate.

The criticality is assumed to occur subsequent to the detection of the oxalate buildup, during the recovery activity, when the hands of an individual attempting to recover the spilled oxalate approach the mass, thereby providing the necessary reflection. (It should be noted that the mass postulated to be required for criticality is theoretical. A similar incident resulting in the accumulation of 40 kg of oxalate in two equal piles did not result in a criticality. It is believed that the moderating solution tends to drain out of the mass, leaving it at a lower k_{eff} than that value used for theoretical calculations.)

The initial radiation spike resulting from 1×10^{17} fissions is assumed to occur 2 ft from the operator attempting to recover the oxalate. No shielding is assumed. Two additional personnel are presumed to be 6 ft from the event, shielded by the glovebox shielding. Four other personnel are distributed throughout N cell at various greater distances.

No physical damage to the glovebox or associated equipment is expected or postulated.

The criticality event ends when the oxalate mass is no longer in a critical configuration, either as a result of the moderating soltution being driven off, or more likely, as a result of disruption of the arrangement of the mass during the initial burst.

Table 9-9 summarizes the assumptions made in evaluating this scenario. They are listed as "more likely" and "worst case" situations.

9.2.2.1 Accident-Induced Radionuclide Release from N Cell Criticality. The adverse meteorology is summarized in Table 9-10. The radionuclide release resulting from an N cell criticality is identified in Table 9-11. The release would occur at the 296-A-1 stack, 78 ft above grade.

9413219.0980

TABLE 9-9. Summary of Assumptions for Plutonium Oxide Production Facility Criticality Accident Scenario.

Assumptions	More likely	Worst case
Amount of plutonium oxalate required for criticality	20 kg	11 kg
Time required to accumulate critical mass	24 h	11 h
Criticality Parameters		
Pulses	1	48
Total fissions	10^{17}	1×10^{19}[a]
Duration of pulse	1 s	1 s initial, 0.5 s subsequent
Pulse frequency	-	1 pulse/10 min
Building release fractions:		
Noble gases	1	1
Halogens	<0.1	0.25

[a]Reference 15.

TABLE 9-10. Adverse Meteorology Summary: N Cell Criticality, 8-h Release from Ground Level.

Recipient	Distance/Direction	Atmospheric stability	X/Q[a] (s/m^3)	Wind Speed
		Onsite		
Maximum worker	400 m NE E	Neutral	4.5×10^{-4}	1.0 m/s
Workforce[b]	1,400-7,600 m W	Very Stable	1.2×10^{-1}	1.0 m/s
		Offsite		
Maximum individual	15.5 km	Moderate	1.1×10^{-6}	10 m/s
Population[c]	15-45 mi SE	Moderate	3.4×10^{-3}	10 m/s

[a]Critical organ value, since inert gases and iodine were computed separately.

[b]Workforce totals 2,096 persons, X/Q is population total.

[c]Population totals 109,411 persons, X/Q is population total.

TABLE 9-11. Radionuclide Release at Building Envelope from N Cell Criticality.

Isotope	Ci[a]	Isotope	Ci
^{83m}Kr	1.1×10^{2}	^{135}Xe	4.1×10^{2}
^{85}Kr	8.1×10^{-4}	^{135m}Xe	3.3×10^{3}
^{85m}Kr	7.1×10^{1}	^{137}Xe	4.9×10^{4}
^{87}Kr	4.3×10^{2}	^{138}Xe	1.1×10^{4}
^{88}Kr	2.3×10^{2}	^{131}I	2.8×10^{0}
^{89}Kr	1.3×10^{4}	^{132}I	3.0×10^{2}
^{131m}Xe	1.0×10^{-1}	^{133}I	4.0×10^{1}
^{133}Xe	2.7×10^{1}	^{134}I	1.1×10^{3}
^{133m}Xe	2.2×10^{0}	^{135}I	1.1×10^{2}

[a]There would be other FP generated from the criticality, for example: ^{90}Sr, ^{91}Y, ^{95}Zr, ^{103}Ru, ^{106}Ru, ^{137}Cs, ^{141}Ce, and ^{144}Ce, (these are based on ^{239}Pu fission yields and 1×10^{19} fissions). However, it is believed that most of these would be contained within the oxalate and any released would also be subject to additional filtration. Therefore, only the volatile isotopes are presented as significant releases.

9.2.2.2 Dose Commitment from N Cell Criticality. Dose commitments for onsite and offsite locations follow.

A critical event in the PUREX Oxide Production Facility would result in significant direct neutron and gamma exposure to personnel in the cell, as well as exposure to other onsite and offsite personnel from the radionuclide release. An evaluation of this direct exposure was performed to analyze the adequacy of N cell evacuation routes in the event of such an emergency.(16) The analysis postulated nine employees at various locations within N cell, during two separate critical events. In both cases there were employees within 2 ft of the postulated events. Assuming immediate evacuation from N Cell at a rate of 15 ft/s, all personnel were able to place themselves in dose fields of less than 10 mR/s in 3 s.

Of the nine employees in N Cell at the time of the postulated event, 5 would receive in excess of the "LD-50" dose of 450 R, based on an accident pulse of 1.0×10^{18} fissions in 0.5 s.(15) Onsite and offsite doses resulting from the radionuclide release are identified in Table 9-12.

9.2.2.3 Frequency of Occurrence. From Appendix D, the estimated frequency of an N cell criticality is 7.5×10^{-5} a year.

9.2.3 Dissolving Short-Cooled Fuel

This accident involves charging inadequately aged N Reactor fuel elements to one of the three PUREX fuel dissolvers.

The N Reactor basin is the only credible source of short-cooled ("green", i.e., aged less than 180 d) fuel during future PUREX operations. During N Reactor fuel loadout, fuel elements are discharged into underwater trolleys ("fast carts") which transport the fuel to a storage area where the fuel is placed in baskets. The fuel is held in baskets for a minimum of 18 d and then placed in canisters. Canisters of fuel are then held until the fuel elements have a total post-irradiation age of 180 d. At this time, the canisters are capped and segregated into rows in the N Reactor storage basin.

Initiation of this accident requires an administrative or fuel handling error at the N Reactor storage basin involving the premature capping of canisters containing 18 d aged elements. If this error occurred, it is conceivable that these green fuel canisters are mistaken for aged fuel canisters and loaded into shipping casks for shipment to the plant. Allowing for basin handling, administrative processing, and transport, 25 d aged fuel could be received at the plant.

TABLE 9-12. Doses Resulting from Airborne Releases Due to N Cell Criticality.

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	6.0×10^{-1}	6.0×10^{-1}	8.3×10^{-1}	2.7×10^{0}	6.0×10^{-1}	6.0×10^{-1}	8.3×10^{-1}	2.7×10^{0}
Workforce (man-rem)	5.3×10^{1}	6.0×10^{1}	1.2×10^{2}	2.5×10^{2}	5.3×10^{1}	5.3×10^{1}	1.2×10^{2}	2.5×10^{2}
Offsite								
Maximum individual (rem)	3.0×10^{-3}	3.3×10^{-3}	4.3×10^{-3}	9.0×10^{-2}	3.0×10^{-3}	3.3×10^{-3}	4.3×10^{-3}	9.3×10^{-2}
Population (man-rem)	8.3×10^{0}	1.1×10^{1}	1.3×10^{1}	1.3×10^{2}	8.3×10^{0}	1.1×10^{1}	1.3×10^{1}	1.3×10^{2}

During N Reactor storage basin cask loading operations, casks are loaded with canisters from a given storage row. Normally one fuel cask transport car (three casks, nine canisters) is loaded per shift. Though it is possible under ideal conditions to load two transport cars in a shift, this would be too great a coincidence if simultaneous with a green fuel accident.(16) It is judged unlikely that the chain of events necessary to load green fuel for shipment will persist for more than the operation of one shift. Therefore, this accident addresses the receipt of only nine canisters of green fuel at the PUREX plant.

Each of the three wells in a cask car is equipped with a temperature monitor to indicate temperature of the water surrounding the fuel casks. Elevated cask well temperature could result from a number of causes including green fuel. If a high temperature indication is observed (regardless of cause), the well water level is checked and the cause investigated. This accident scenario assumes a high temperature indication goes unobserved.

The only possible short-cooled fuel available during future PUREX campaigns is 6% ^{240}Pu.(17) It is assumed the green fuel shipped to the PUREX plant is 6% ^{240}Pu Mark IV (about 80% of the fuel to be processed at the plant is Mark IV, a canister of which contains about 40% more fuel than a canister of Mark IA elements). A typical dissolver charge of this fuel consists of 32 canisters (10.2 t of U). Therefore, the dissolver charge analyzed here consists of nine canisters of 25-d aged 6% Mark IV fuel elements and 23 canisters of 180-d aged 6% Mark IV fuel elements.

The green fuel detector is physically located such that fuel canisters cannot be moved into the PUREX Plant canyon without being monitored. This detector is set to alarm when it is exposed to radiation levels greater than those expected from 180-d aged fuel. This monitor must be deenergized, malfunctioning, or misinterpreted to not indicate the charging of green fuel.

During the fuel decladding process, up to 10% of the uranium metal can be attacked by the ammonium fluoride-ammonium nitrate (AFAN) solution releasing fission product gases (specifically I and Xe) and forming insoluble uranium tetrafluoride.(18) Assuming higher iodine concentration at the outer circumference of fuel elements, it is possible to release considerable quantities of iodine to the dissolver solution during fuel decladding. This iodine should remain in solution as ammonium or uranium iodide. If some were released to the DOG system, early detection is doubtful because the single channel gamma scintillation detectors (process monitors) in the DOG stream to monitor ^{131}I are not on line during fuel decladding; however, there is a ^{131}I monitor on the PUREX main exhaust stack (291-A-1). High levels of noble gases released during declad will be detected by new stack monitors. However, for this scenario it is assumed that decladding is completed and fuel dissolution proceeds.

Because process monitors are online during dissolution, elevated levels of ^{131}I released to the DOG system will be detected. During a past accident the process monitors (one in C-DOG system and one in the 293-A facility) alarmed within 5 min of reaching maximum dissolution temperature (110°C) in the dissolver.(19) About 15 min was spent verifying the alarms prior to suspending the dissolution process. Current plans call for immediate suspension of steam to the dissolver and introduction of cooling water to the coils in response to an alarm. The alarm will be verified while the dissolver is being cooled. Upon verification of the alarm, fuel age will be determined and the iodine release potential estimated. The time period between alarm indication and verification is expected to be about 20 to 30 min. Dissolution should be completely terminated in 2 to 3 h. Actions to reduce iodine release will be identified and evaluated based on specific accident conditions. In the past, mercuric nitrate solutions were used to suppress iodine emissions.

About 21% of the fuel is dissolved during the time to reach peak dissolution temperature and the 3 h required to terminate dissolution.(20) Radioisotopes associated with this portion of the charged fuel go into solution where they become subject to entrainment into the DOG. The dissolver tower scrubs most of the less volatile aerosols back into solution. All of the noble gases and 65% of the iodine in solution will be assumed to leave the tower in the DOG.(20) The DOG system is assumed to have a decontamination factor (DF) of 1,000 (700 for worst case) for removal of iodine and a DF of 1 for noble gases. The DF for iodine accounts for a DF of 100 for the silver reactor and a DF of 10 for the backup facility.(20)

Table 9-13 summarizes the assumptions which impact source term calculations for this scenario. They are listed as a "more likely" and "worst" case situations.

9.2.3.1 Accident-Induced Radionuclide Release from Dissolving Short-Cooled Fuel. The adverse meteorology is summarized in Table 9-14. The radionuclide release resulting from this accident is identified in Table 9-15. This release would be at the 291-A-1 stack, 200 ft above grade.

9.2.3.2 Dose Commitment. Onsite and offsite doses resulting from this accident are identified in Table 9-16.

9.2.3.3 Frequency of Occurrence. From Appendix D, the estimated frequency of dissolving short-cooled fuel is 2.5×10^{-7} a year.

TABLE 9-13. Summary of Assumptions for Calculating Source Terms for the "Dissolution of Short-Cooled Fuel" Accident Scenario.

Assumptions	More likely	Worst case
Fuel		
Type	Mark IV	Mark IV
Percent ^{240}Pu	6	6[a]
Total weight of charge	10.2 t of U (32 canisters)	10.2 t of U (32 canisters)
Amount of green fuel charged	1 to 9 canisters)	9 canisters
Age of green fuel charged	25 to 150 d	25 d
Age of remainder of charge	180 d	180 d
Time to detect and terminate dissolution	2 h	3 h
Fraction of radionuclides present in total charge which are released to solution[b]		
Noble gases	1	1
Halogens	<0.23	0.23
Semivolatile solids	<0.21	0.21
Nonvolatile solids	<0.21	0.21
Fraction of radionuclides released from solution to offgas system[c]		
Noble gases	1	1
Halogens	<0.5	0.65
Semivolatile solids[21]	<0.005	0.01
Nonvolatile solids[21]	0.001	0.005
DF of offgas system		
Noble gases	1	1
^{131}I[c]	>1,000	700
^{129}I[d]	>150	100
All other nuclides	>2700	1,000

[a]It is assumed no 12% short cooled fuel is available. If however, 12% is produced, it would be the worst case. (Although Mark IV has fewer curies per t of U than Mark IA, more tons of Mark IV are allowed in a dissolver at one time. This makes Mark IV the worst-case fuel type.)

[b]The worst case is based on the dissolution of 21% of the fuel. The 23% for iodine is based on a theoretical 10% concentration in the outer layers of fuel.

[c]These constitute releases downstream of the reflux tower.

[d]The ^{129}I DF is 1/7 the ^{131}I DF.[22]

TABLE 9-14. Adverse Meteorology Summary: Short-Cooled Fuel, 3-h Release from 200-ft-high Stack.

Recipient	Distance/Direction	Atmospheric stability	X/Q[a] (s/m^3)	Wind speed
		Onsite		
Maximum worker	350 m NE	Unstable	6.5×10^{-5}	1.0 m/s
Workforce[b]	1,400-7,600 mi W	Neutral	1.2×10^{-2}	1.0 m/s
		Offsite		
Maximum individual	15.5 km E	Moderate	1.8×10^{-6}	1.0 m/s
Population[c]	15-45 mi SE	Moderate	6.2×10^{-3}	2.5 m/s

[a]Critical organ value, since inert gases were computed separately.
[b]Workforce totals 2,096 persons, X/Q is population total.
[c]Population totals 109,411 persons, X/Q is population total.

TABLE 9-15. Release at Building Envelope from Dissolution of Short-Cooled Fuel (Sheet 1 of 2).

Isotope	Ci	Isotope	Ci
^{85}Kr	4.2×10^{3}	^{140}Ba	5.6×10^{-1}
^{89}Sr	9.8×10^{-1}	^{140}La	6.4×10^{-1}
^{90}Sr	3.6×10^{-2}	^{143}Pr	6.2×10^{-1}
^{90}Y	3.6×10^{-2}	^{144m}Pr	9.9×10^{-3}
^{91}Y	1.2×10^{0}	^{147}Nd	1.6×10^{-1}
^{95}Zr	1.4×10^{0}	^{148m}Pm	6.6×10^{-3}
^{95}Nb	1.7×10^{0}	^{154}Eu	1.6×10^{-4}
^{103}Ru	1.4×10^{0}	^{155}Eu	3.9×10^{-4}
^{106}Ru	1.7×10^{-1}	^{156}Eu	7.4×10^{-3}
^{125}Sb	2.1×10^{-3}	^{133m}Xe	2.3×10^{2}
^{127}Te	5.8×10^{-3}	^{134}Cs	4.2×10^{-3}
^{127m}Te	5.9×10^{-3}	^{137}Cs	7.5×10^{-2}
^{129}Te	2.1×10^{-2}	^{141}Ce	1.1×10^{0}
^{129m}Te	3.3×10^{-2}	^{144}Ce	8.2×10^{-1}
^{132}Te	7.4×10^{-3}	^{144}Pr	8.2×10^{-1}
^{129}I	1.2×10^{-5}	^{147}Pm	1.3×10^{-1}
^{131}I	2.5×10^{1}	^{234}U	3.2×10^{-6}
^{132}I	1.6×10^{0}	^{235}U	1.7×10^{-7}
^{131m}Xe	3.3×10^{3}	^{236}U	3.9×10^{-7}
^{133}Xe	1.1×10^{5}	^{238}U	3.2×10^{-6}
^{103m}Rh	1.4×10^{0}	^{238}Pu	4.3×10^{-5}
^{106}Rh	1.7×10^{-1}	^{239}Pu	6.0×10^{-4}

TABLE 9-15. Release at Building Envelope from Dissolution of Short-Cooled Fuel (Sheet 2 of 2).

Isotope	Ci	Isotope	Ci
^{137m}Ba	7.1×10^{-2}	^{240}Pu	1.4×10^{-4}
^{95m}Nb	1.2×10^{-2}	^{241}Pu	7.4×10^{-3}
^{125}Sn	1.5×10^{-3}	^{242}Pu	9.3×10^{-9}
^{125m}Te	4.0×10^{-4}	^{241}Am	5.1×10^{-6}

TABLE 9-16. Doses Resulting from Dissolution of Short-Cooled Fuel.

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	4.2×10^{-2}	7.1×10^{-2}	6.1×10^{-2}	6.8×10^{-1}	4.5×10^{-2}	1.3×10^{-1}	6.1×10^{-2}	6.8×10^{-1}
Workforce (man-rem)	9.0×10^{0}	1.4×10^{1}	1.3×10^{1}	1.3×10^{2}	9.5×10^{0}	2.5×10^{1}	1.3×10^{1}	1.3×10^{2}
Offsite								
Maximum individual (rem)	5.5×10^{-3}	7.4×10^{-3}	4.3×10^{-3}	1.0×10^{0}	6.9×10^{-3}	1.3×10^{-2}	4.3×10^{-3}	1.1×10^{0}
Population (man-rem)	1.7×10^{1}	2.2×10^{1}	1.6×10^{1}	1.7×10^{3}	2.0×10^{1}	3.5×10^{1}	1.6×10^{1}	1.8×10^{3}

9.2.4 Uranium Metal Fire in Dissolver

This accident involves extended exposure of irradiated N Reactor fuel elements to air in one of the three PUREX Plant fuel dissolvers.

Initiation of this accident requires declad fuel element exposure to air, adequate time for decay heat buildup, and ignition. Fuel can remain exposed to air in a dissolver long enough for this accident to occur only due to a combination of operator errors and/or equipment failures. During routine process operations there is no period of sufficient duration which permits the undetected occurrence of such an event. This accident is assumed to occur during a nonroutine standby period after decladding before the first dissolution cut on a dissolver charge of Mark IV fuel. It is less likely for this accident to occur later in the Mark IV fuel processing cycle or while processing Mark IA fuel.

In this accident, fuel elements are exposed to air due to a failure to add adequate water to the dissolver at the beginning of an extended standby interval. In addition, liquid level indicator and/or temperature monitor readings are inaccurate, misinterpreted, or overlooked. Heat is generated from radioactive decay of fission products and the (initial) slow oxidation of uranium. Fuel element configuration reduces heat transfer resulting in increased localized temperatures. [Computer thermal analyses predict peak fuel element temperature of about 405°C after 10 h of exposure in air for 180 d aged Mark IV fuel that has been ignited as a source of fire.(23)] Comparison of fuel element thermal and pyrophoricity data to historical dissolver fire data resulted in postulation that ignition occurs about 10 to 12 h after the fuel becomes exposed.(24) [The ignition temperature for uranium fuel elements has been conservatively assumed to be 300°C based on a review of information.(24,25)]

Once ignition occurs, rapid oxidation takes place and considerable quantities of radionuclides enter the DOG stream. It is judged that FP nuclides such as ruthenium and cesium will be detected at the stack monitor within 2 to 4 h. However, it is assumed for the purpose of this evaluation that the burn continues for 10 h without indication and, that upon detection, up to 2 h is required to properly confirm and interpret the alarm. During the 12 h postulated for the unidentified burn it is possible the dissolver shell would be penetrated due to the high temperatures (1,300°C to 1,400°C) generated by the fire. Extinguishing the fire with water may not be possible if penetration occurs low in the dissolver, therefore, it is assumed the entire charge is consumed (10.2 t of U of Mark IV fuel).

The combustion rate of uranium is a function of several complex parameters; this scenario assumes an average combustion rate of 540 kg/h (see note a, Table 9-17).(26) This combustion rate indicates about 19 h are required for the fire to self-extinguish. If an attempt is made to extinguish the fire, water could be added from the drown tank (normally

TABLE 9-17. Summary of Assumptions for Calculating Source Terms for the "Uranium Metal Fire in Dissolver" Accident Scenario.

Assumptions	More likely	Worst case
Fuel		
Type	Mark IV	Mark IV
Percent of ^{240}Pu	6% or 12%	6%
Total weight of charge	10.2 t of U	10.2 t of U
Age	≥180 d	180 d
Fuel exposure time preceeding ignition	>16 h	10 h
U oxidation rate	<540 kg/h	540 kg/h [a]
Duration of undetected burn	2 to 4 h	10 h
Time to respond	0.5 to 2 h	2 h
Time to extinguish	20 to 30 min [b]	[c]
Release fraction for consumed fuel (21)		
Volatiles	100%	100%
Cs and Ru	<10%	10%
All others (except volatiles)	<1%	1%
System DF		
I (cell vent system only)	2	2
^{129}I (DOG)	>150	100
Nonvolatiles (both DOG & cell systems)	>2,700	1,000
^{131}I (DOG)	>1,000	700

[a]It is calculated based on the following: uranium corrosion rate in air at 1,300°C is 16 g/cm^2·h; the outer element surface has an area of 2500 cm^2; 448 elements/charge; only 3% of charge undergoing oxidation at a given time.(25)

[b]This assumes either no penetration of dissolver or penetration at an upper level which would permit effective drowning of the fire with water.

[c]Extinguishing the fire is not possible in the worst case due to penetration of the dissolver wall near the bottom. Fire would self extinguish in about 19 h.

9413219.0993

containing 2,500 gal) and other sources once it is emptied, at the maximum rate of 150 gal/min. Most of the water would drain from the dissolver through penetrations made by the fire spreading contamination onto the cell floor and into the adjacent cell vent tunnel.

About 10% of the semivolatile radionuclides (Cs, Ru) and 1% of all other radioisotopes are released from the dissolver. Based on the time required for dissolver penetration and the relative exhaust rates of the DOG and canyon cell systems, an estimated 90% of the radionuclides are released to the DOG and 10% to the cell vent system. Both systems provide similar levels of filtration, reducing the impacts to negligible levels for all isotopes with the possible exception of ^{129}I. A DF of 2 is assumed for ^{129}I in the cell ventilation system because some iodine is lost to the walls and other system components. For all other nonvolatile isotopes, the DF is conservatively assumed to be 1,000 (corresponding to a 99.90% efficiency).

Table 9-17 summarizes the assumptions which impact source term calculations for this scenario. They are listed as "more likely" and "worst case" situations.

9.2.4.1 Accident-Induced Radionuclide Release from Uranium Metal Fire in Dissolver. The adverse meteorology is summarized in Table 9-18. The radionuclide release resulting from this accident is identified in Table 9-19. The release would be at the 291-A-1 stack, 200 ft above grade.

9.2.4.2 Dose Commitment. Onsite and offsite doses resulting from this accident are identified in Table 9-20.

9.2.4.3 Frequency of Occurrence. From Appendix D, the estimated frequency of a uranium metal fire in a dissolver is 2×10^{-3} a year.

9.2.5 Silver Reactor Explosion

This accident involves an explosion in the DOG silver reactor during fuel dissolution at the PUREX Plant.

The silver reactors consist of silver nitrate coated packing material and are designed for removing iodine (primarily ^{131}I) from the DOG stream. A past reactor explosion has been attributed to a now discontinued practice of washing the reactor with ammonium hydroxide.(27) Even so, small amounts of ammonia reach the silver reactor in the gas stream despite the ammonia scrubbers. An undetected heater failure, inadvertent shutoff, or malfunction of the temperature control/alarm system for the DOG steam and electric heater may allow formation and accumulation of unstable silver-nitrogen-halogen compounds in the silver reactor. Once the deenergized heater is discovered, the reactor condition would be evaluated and possibly be rejuvenated or replaced prior to energizing the electric heater. This scenario assumes the heater is energized without proper consideration causing an explosion in the silver reactor, rupturing it, and ejecting its contents into the cell.

TABLE 9-18. Adverse Meteorology Summary: Uranium Fire, 8-h Release from the 200-ft-high Stack.

Recipient	Distance/Direction	Atmospheric stability	X/Q[a] (s/m^3)
	Onsite		
Maximum worker	350 m NE	Unstable	6.5 x 10-5
Workforce[b]	1,400-7,600 m W	Neutral	1.2 x 10-2
	Offsite		
Maximum individual	15.5 km E	Moderate	3.2 x 10-6
Population[c]	15-45 mi SE	Moderate	2.7 x 10-2

[a]Wind speed is 1 m/s in all cases.
[b]Workforce totals 2,096 persons, X/Q is population total.
[c]Population totals 109,411 persons, X/Q is population total.

TABLE 9-19. Release at Building Envelope from a Uranium Metal Fire in a Dissolver.

Isotope	Ci	Isotope	Ci
^{85}Kr	4.1×10^{3}	^{125m}Te	4.3×10^{-3}
^{89}Sr	3.2×10^{0}	^{144m}Pr	8.3×10^{-2}
^{90}Sr	3.4×10^{-1}	^{155}Eu	3.6×10^{-3}
^{90}Y	3.4×10^{-1}	^{134}Cs	1.9×10^{-1}
^{91}Y	4.8×10^{0}	^{137}Cs	3.5×10^{0}
^{95}Zr	6.2×10^{0}	^{141}Ce	1.2×10^{0}
^{95}Nb	1.1×10^{1}	^{144}Ce	6.9×10^{0}
^{103}Ru	1.3×10^{1}	^{144}Pr	6.9×10^{0}
^{106}Ru	7.5×10^{0}	^{148m}Pm	1.4×10^{-2}
^{125}Sb	2.0×10^{-2}	^{147}Pm	1.2×10^{0}
^{127}Te	3.7×10^{-2}	^{234}U	3.1×10^{-5}
^{127m}Te	3.8×10^{-2}	^{235}U	1.6×10^{-6}
^{129}Te	2.6×10^{-2}	^{236}U	3.7×10^{-6}
^{129m}Te	4.1×10^{-2}	^{238}U	3.1×10^{-5}
^{129}I	4.6×10^{-4}	^{238}Pu	4.1×10^{-5}
^{131}I	3.4×10^{-2}	^{239}Pu	5.7×10^{-3}
^{131m}Xe	2.5×10^{0}	^{240}Pu	1.4×10^{-3}
^{103m}Rh	1.3×10^{1}	^{241}Pu	7.0×10^{-2}
^{106}Rh	7.5×10^{0}	^{242}Pu	8.9×10^{-8}
^{137m}Ba	3.4×10^{0}	^{241}Am	6.2×10^{-5}
^{95m}Nb	5.1×10^{-2}		

9413219.0996

TABLE 9-20. Doses Resulting from a Uranium Metal Fire in a Dissolver.

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	5.9×10^{-3}	6.2×10^{-2}	8.3×10^{-1}	9.2×10^{-4}	2.4×10^{-2}	3.6×10^{-1}	1.1×10^{0}	9.3×10^{-4}
Workforce (man-rem)	1.1×10^{0}	1.2×10^{1}	1.6×10^{2}	1.8×10^{-1}	4.6×10^{0}	7.0×10^{1}	2.2×10^{2}	1.8×10^{-1}
Offsite								
Maximum individual (rem)	2.5×10^{-3}	9.2×10^{-3}	4.1×10^{-3}	3.2×10^{-3}	2.9×10^{-2}	1.0×10^{-1}	6.2×10^{-2}	9.8×10^{-3}
Population (man-rem)	1.5×10^{1}	5.6×10^{1}	3.5×10^{2}	1.5×10^{1}	1.4×10^{2}	5.1×10^{2}	5.2×10^{2}	5.3×10^{1}

An explosion in the silver reactor would be indicated by several means. The historical reactor explosion caused a loud report which was heard along with detectable vibration in the PUREX Plant.(27) Dissolver vacuum and DOG filter DP gauges would show erratic readings and the electric heater temperature control system (if functioning) would indicate temperature fluctuations. Interpretation of these indications to identify an explosion in the silver reactor and formulation of response actions are postulated to take 30 min. Upon identification of the accident, the dissolution process would be stopped within 1 h by cooling the dissolver with drown tank water. The time to terminate the production of offgas is therefore, around 1.5 h in the worst case.

The source terms for this accident include ^{129}I and ^{131}I (for 6% ^{240}Pu, Mark IA fuel) from the DOG. These isotopes are released during the 1.5 h period between the explosion and termination of offgas production. Other radionuclides will be present in the offgas, but are not identified because they pass through filters equivalent to the DOG system (canyon filters) and the release levels are comparable to routine operating levels. The radionuclide inventory in the silver reactor is the major contributor to the abnormal release. This includes up to 45% of the ^{129}I from 2,800 t of fuel (in this case, 12% ^{240}Pu, 80% Mark IV, and 20% Mark IA), 50 kg of uranium, and associated FP.* The iodine inventory is based on an estimated iodine retention DF of 10, 50% release from solution, and six regenerations (recoating with $AgNO_3$) of the silver reactor. The fuel inventory is based on measured levels found on PUREX reactor packing and is in the range of theoretical calculations.(28-30)

Table 9-21 summarizes the assumptions which impact source term calculations for this scenario. They are listed as "more likely" and "worst case" situations.

9.2.5.1 Accident-Induced Radionuclide Release from Silver Reactor Explosion. The adverse meteorology is summarized in Table 9-22. The radionuclide release resulting from this accident is identified in Table 9-23. This release would be at the 291-A-1 stack, 200 ft above grade.

9.2.5.2 Dose Commitment. Onsite and offsite doses resulting from this accident are identified in Table 9-24.

9.2.5.3 Frequency of Occurrence. From Appendix D, the estimated frequency of occurrence of a silver reactor explosion is 4×10^{-5} a year.

*The radionuclides presented are based on average proportions found in fuel. Selective volatilization and preferential retention in the silver reactor can alter these ratios.(28)

TABLE 9-21. Summary of Assumptions for Calculating Source Terms for the "Silver Reactor Explosion" Accident Scenario.

Assumptions	More likely	Worst case
Fuel in dissolver during accident		
Type	Mark IV	Mark IA
Percent ^{240}Pu	6 or 12%	6%
Total weight of charge	10.2 t of U	4.8 t of U
Age of charge	180 d or 2.5 yr	180 d
Fraction of iodine released to solution during dissolution	0.23	0.23
Iodine released from solution to DOG system	0.5	0.65
Source of silver reactor inventory	6% or 12% ^{240}Pu, 80% Mark IV, 20% Mark IA, 180 d to 15 yr old	12% ^{240}Pu, 80% Mark IV, 20% Mark IA all 5 yr old
Amount of fuel in silver reactor (isotope percentage distribution, except iodine, same as for fuel)	2 to 20 kg	50 kg
Amount of iodine in silver reactor	45% of I from 1,800 to 2,400 t of U	45% of I from 2,800 t of U
Fraction of silver reactor iodine inventory volatilized by explosion.	1	1
Fraction of silver reactor fuel[21] inventory aerosolized	<0.01	0.1
Time to identify accident	10 to 20 min	30 min
Time to terminate offgas	1 h	1 h
Canyon Vent System DF		
Noble gases	1	1
Halogens	>2	2
All other nuclides	>2,700	1,000

TABLE 9-22. Adverse Meteorology Summary: Silver Reactor Explosion, 10-min Release from the 200-ft-high Stack.

Recipient	Distance/Direction	Atmospheric stability	X/Q (s/m^3)	Wind speed
		Onsite		
Maximum worker	350 m NE	Unstable	6.5×10^{-5}	1.0 m/s
Workforce[a]	1,400-7,600 m W	Neutral	1.2×10^{-2}	1.0 m/s
		Offsite		
Maximum individual	15.5 km E	Moderate	3.6×10^{-6}	1 m/s
Population[b]	15-45 mi SE	Moderate	6.2×10^{-3}	2.5 m/s

[a]Workforce totals 2,096 persons, X/Q is population total.
[b]Population totals 109,411 persons, X/Q is population total.

TABLE 9-23. Release at Building Envelope from Silver Reactor Explosion.

Isotope	Ci	Isotope	Ci
^{85}Kr	0	^{123}Sn	3.5×10^{-7}
^{89}Sr	3.7×10^{-11}	^{125m}Te	2.6×10^{-4}
^{90}Sr	4.0×10^{-2}	^{134}Cs	1.6×10^{-3}
^{90}Y	4.0×10^{-2}	^{137m}Ba	4.3×10^{-2}
^{91}Y	1.3×10^{-9}	^{137}Cs	4.5×10^{-2}
^{95}Zr	9.8×10^{-9}	^{141}Ce	4.8×10^{-17}
^{95}Nb	2.2×10^{-8}	^{144}Ce	1.5×10^{-2}
^{103}Ru	2.9×10^{-14}	^{144}Pr	1.5×10^{-2}
^{106}Ru	6.8×10^{-3}	^{147}Pm	4.8×10^{-2}
^{125}Sb	1.1×10^{-3}	^{148}Pm	1.5×10^{-16}
^{127}Te	1.4×10^{-7}	^{148m}Pm	3.2×10^{-15}
^{127m}Te	1.4×10^{-7}	^{151}Sm	1.2×10^{-4}
^{129}Te	3.9×10^{-18}	^{154}Eu	3.7×10^{-4}
^{129m}Te	6.2×10^{-18}	^{155}Eu	2.0×10^{-4}
^{129}I	1.4×10^{0}	^{238}Pu	1.7×10^{-4}
^{131}I	2.7×10^{-2}	^{239}Pu	5.9×10^{-4}
^{95m}Nb	8.0×10^{-11}	^{240}Pu	3.3×10^{-4}
^{103m}Ru	2.9×10^{-14}	^{241}Pu	2.7×10^{-2}
^{106}Rh	6.8×10^{-3}	^{242}Pu	1.2×10^{-7}
^{110m}Ag	1.3×10^{-5}	^{241}Am	2.2×10^{-4}

9413219.1001

TABLE 9-24. Doses Resulting from Silver Reactor Explosion.*

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	4.3×10^{-4}	4.3×10^{-3}	1.8×10^{-3}	1.3×10^{-1}	4.6×10^{-4}	8.9×10^{-2}	1.8×10^{-3}	1.4×10^{-1}
Workforce (man-rem)	8.0×10^{-2}	8.0×10^{-1}	3.4×10^{-1}	2.4×10^{1}	8.7×10^{-1}	1.7×10^{1}	3.4×10^{-1}	2.6×10^{1}
Offsite								
Maximum individual (rem)	1.3×10^{-3}	1.0×10^{-3}	2.0×10^{-4}	9.2×10^{-1}	1.2×10^{-2}	2.1×10^{-2}	5.2×10^{-3}	2.9×10^{0}
Population (man-rem)	1.6×10^{0}	1.4×10^{0}	2.9×10^{-1}	1.1×10^{3}	1.2×10^{1}	2.3×10^{1}	6.1×10^{0}	3.1×10^{3}

*More than 90% of the thyroid dose is due to the assumed release from the Silver Reactor. The additional release from the dissolver is a minor contribution.

9.2.6 Hydrogen Explosion in a Dissolver

This accident involves an explosion in a PUREX Plant dissolver during fuel decladding.

This accident assumes AFAN is not available when needed during preparation for Mark IA (spike) fuel charging. A procedure deviation authorization (PDA) is written to minimize process delay. The dissolver liquid level is adjusted with water and spike fuel is charged and covered with water.

To begin declad, dissolver liquid level is reduced by boiling off an appropriate amount of water to make room for the AFAN. After liquid level adjustment, the dissolver should be cooled to between 25°C and 30°C prior to AFAN addition. However, due to operator error or temperature probe failure, the AFAN is added without sufficient cooling of the dissolver. The fluoride begins to react with unoxidized zirconium in the heel before the AFAN addition is complete. Higher than normal hydrogen production results due to the dilute nitrate concentration. Hydrogen builds up in the dissolver and DOG system very rapidly, exceeding the lower explosive limit, and is ignited by a mechanical or static induced spark, or localized high temperature.(31) The ensuing explosion lifts the lid off the dissolver releasing aerosols and volatiles to the canyon vent system.

The aerosol nuclide inventory is estimated by assuming an instantaneous explosion-induced particulate concentration of 100 to 1,000 mg/m^3(32) composed of fuel and dissolver solution in the dissolver cell. The cell volume (about 560 m^3) indicates 56 to 560 g of material are suspended. The calculated volume ratio in the aerosol indicates that approximately 94% is water and AFAN and 6% is fuel (52 wt%). Therefore, 560 g of aerosol is composed of approximately 290 g of fuel, 100% of which is assumed to confront the canyon filter system. The maximum aerosol concentration of 1,000 mg/m^3 is not likely to involve the entire cell volume; however, the 290 g of resulting fuel aerosol is only 0.006% of a dissolver charge, which does not seem excessive for a conservative analysis.

Table 9-25 summarizes the assumptions used to allow calculation of source terms associated with this scenario. They are listed as "more likely" and "worst case" situations.

9.2.6.1 Accident-Induced Radionuclide Release from Hydrogen Explosion in a Dissolver. The adverse meteorology is summarized in Table 9-26. The radionuclide release resulting from this accident is identified in Table 9-27. This release would be from the 291-A-1 stack, at 200 ft above grade.

9.2.6.2 Dose Commitment. Onsite and offsite doses resulting from this accident are identified in Table 9-28.

9.2.6.3 Frequency of Occurrence. From Appendix D, the probability of a hydrogen explosion in a dissolver is 1.2 x 10^{-3} a year.

TABLE 9-25. Summary of Assumptions for Calculating Source Terms for the "Hydrogen Explosion in a Dissolver" Accident Scenario.

Assumptions	More likely	Worst case
Fuel		
Type	Mark IV	Mark IA*
Age	>180 d	180 d
Percent ^{240}Pu	6% or 12%	6%
Amount	10.2 t of U	4.8 t of U
Concentration of aerosols generated by explosion	100 to 1,000 mg/m^3	1,000 mg/m^3
Fraction of aerosol consisting of fuel (vol%)	<6%	6%
Volume associated with aerosol concentration	<560 m^3	560 m^3
Fraction of iodine in suspended matter which is vaporized (versus aerosolized iodides)	<100%	100%
Canyon Vent System DF		
Iodine	>2	2
Nonvolatiles	>2,700	1,000

*Although more tons of Mark IV are present in the dissolver, the aerosol concentration is independent of this fact. Thus Mark IA fuel, with its greater number of FP curies per MTU, is worst case.

TABLE 9-26. Adverse Meteorology Summary: Hydrogen Explosion, 10-min Release from the 200-ft-high Stack.

Recipient	Distance/Direction	Atmospheric stability	X/Q (s/m^3)	Wind speed
		Onsite		
Maximum worker	350 m NE	Unstable	6.5×10^{-5}	1.0 m/s
Workforce[a]	1,400-7,600 m W	Neutral	1.2×10^{-2}	1.0 m/s
		Offsite		
Maximum individual	15.5 km E	Moderate	3.6×10^{-6}	1 m/s
Population[b]	15-45 mi SE	Moderate	6.2×10^{-3}	2.5 m/s

[a]Workforce totals 2,096 persons, X/Q is population total.
[b]Population totals 109,411 persons, X/Q is population total.

TABLE 9-27. Release at Building Envelope from a Hydrogen Explosion in a Dissolver.

Isotope	Ci	Isotope	Ci
^{85}Kr	2.0×10^{3}	^{133m}Xe	4.9×10^{-19}
^{89}Sr	1.1×10^{-2}	^{134}Cs	4.7×10^{-5}
^{90}Sr	1.1×10^{-3}	^{137}Cs	1.1×10^{-3}
^{90}Y	1.1×10^{-3}	^{137m}Ba	1.1×10^{-3}
^{91}Y	1.6×10^{-2}	^{140}Ba	1.6×10^{-5}
^{95}Zr	2.1×10^{-2}	^{140}La	1.8×10^{-5}
^{95m}Nb	1.7×10^{-4}	^{141}Ce	4.2×10^{-3}
^{95}Nb	3.8×10^{-2}	^{144}Ce	2.2×10^{-2}
^{103m}Rh	4.4×10^{-3}	^{144}Pr	2.2×10^{-2}
^{103}Ru	4.4×10^{-3}	^{147}Pm	3.9×10^{-3}
^{106}Rh	2.2×10^{-3}	^{148m}Pm	3.6×10^{-5}
^{106}Ru	2.2×10^{-3}	^{154}Eu	3.9×10^{-6}
^{125}Sb	6.0×10^{-5}	^{155}Eu	1.3×10^{-5}
^{125m}Te	1.3×10^{-5}	^{234}U	9.6×10^{-8}
^{127}Te	1.2×10^{-4}	^{235}U	6.4×10^{-9}
^{127m}Te	1.2×10^{-4}	^{236}U	1.2×10^{-8}
^{129}Te	8.9×10^{-5}	^{238}U	9.6×10^{-8}
^{129m}Te	1.4×10^{-4}	^{238}Pu	8.2×10^{-7}
^{129}I	1.8×10^{-3}	^{239}Pu	1.5×10^{-5}
^{131}I	1.8×10^{-1}	^{240}Pu	3.0×10^{-4}
^{131m}Xe	1.1×10^{0}	^{241}Pu	1.3×10^{-4}
^{133}Xe	3.7×10^{-4}	^{242}Pu	1.3×10^{-10}
		^{241}Am	1.1×10^{-7}

9413219.1006

TABLE 9-28. Doses Resulting from a Hydrogen Explosion in a Dissolver.

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	4.4×10^{-5}	4.9×10^{-4}	4.4×10^{-4}	4.8×10^{-3}	1.2×10^{-4}	1.9×10^{-3}	4.4×10^{-4}	4.8×10^{-3}
Workforce (man-rem)	8.3×10^{-3}	9.2×10^{-2}	8.3×10^{-2}	9.0×10^{-1}	2.3×10^{-2}	3.6×10^{-1}	8.3×10^{-2}	9.0×10^{-1}
Offsite								
Maximum individual (rem)	3.1×10^{-5}	7.7×10^{-5}	2.5×10^{-5}	1.6×10^{-2}	1.2×10^{-4}	4.2×10^{-4}	3.4×10^{-5}	1.9×10^{-2}
Population (man-rem)	2.9×10^{-2}	9.0×10^{-2}	4.3×10^{-2}	1.4×10^{1}	1.1×10^{-1}	4.5×10^{-1}	5.3×10^{-2}	1.7×10^{1}

9.2.7 1BX Column Criticality

This accident involves a criticality in the 1BX partition column during PUREX Plant operations.

The 1BX column is not geometrically favorable for plutonium and therefore requires administrative controls for criticality prevention. These controls include sampling and analysis of the reductant stream makeup, flow control, neutron monitors, and analysis of the feed solution. A number of off-standard conditions and process control failures must occur before the plutonium concentration can reach an unsafe level in the column.

The 1BX column receives an organic solvent feed stream (1BXF) containing plutonium and uranium. A reductant (ferrous sulfamate) is added counterflow to the organic stream which reduces the plutonium from plutonium IV to plutonium III, thereby partitioning it from the organic-favoring uranium. Limits on ferrous ion concentration control the plutonium level in the aqueous phase in the column during normal operation. Both an increased Fe^{+2} ion concentration (possibly due to an error in feed preparation and analysis) and an increase in available plutonium must simultaneously exist to have the potential for a criticality. The additional plutonium could result from a reflux condition where plutonium would accumulate in the aqueous phase of the 1BX column.

Reflux could develop due to the presence of nitrite ions (which oxidize plutonium from III to IV) in the 1BXF stream. Nitrite can be present in a recycle stream from the second plutonium cycle if hydrazine (added to destroy residual NO_2) is absent. The reflux could occur with all column inlet and outlet flows at normal rates.

If reflux did occur, it would take place in the bottom few feet of the column and should be detected by one of three neutron detectors positioned around the outside of the column at 5-ft intervals. Also, the reduced plutonium levels in the product (1BXP) stream should be indicated by sampling and monitoring. Thus, some combination of physical and administrative failures must persist for a period of time sufficient for a minimum critical mass to accumulate.

Calculations indicate that a 20 g/L plutonium concentration, corresponding to a critical mass for the 1BX column geometry of about 1800 g plutonium, could develop with adequate Fe^{+2} ion concentration and 100% reflux in less than 2 h.[33] The resulting criticality, involving less than 100 L of solution, is assumed to involve a total of 1×10^{19} fissions produced by a series of pulses over an 8 h period.[15] A criticality would be detected by cell and offgas monitors and alarms would initiate operator response to alleviate the situation. An 1BX criticality may, in fact, be a self-limiting event; however, the 8-h, 1×10^{19} fissions is a specified criticality for conservative evaluations.[15] The postulated criticality should produce little or no mechanical damage, so all of the released fission products travel via the vessel VOG system to the 291-A-1 stack. The silver reactor in the VOG system has been shown to be largely ineffective in removing iodine due to the required low operating temperature.[34,35]

Table 9-29 summarizes the assumptions which impact source term calculations for this scenario. They are listed as "more likely" and "worst case" situations.

9.2.7.1 Accident-Induced Radionuclide Release from 1BX Column Criticality. The adverse meteorology is summarized in Table 9-30. The radionuclide release resulting from this accident is identified in Table 9-31. This release would be at the 291-A-1 stack, 200 ft above grade.

9.2.7.2 Dose Commitment. Onsite and offsite doses resulting from this accident are identified in Table 9-32.

9.2.7.3 Frequency of Occurrence. From Appendix D, the estimated frequency of a 1BX column criticality is 3.8×10^{-8} a year. As a result of the fault tree analysis, another criticality event, criticality in a dissolver due to overcharge of spike fuel, was determined to have a significantly higher probability, 1.5×10^{-2} a year. Therefore, this probability value will be used in evaluations of overall facility risk.

9413219.1009

9.2.8 Solvent Fire in H Cell

This accident involves the contents of the organic continuous HA column in a solvent fire within the PUREX Plant canyon H cell.

The HA column contains a high inventory of mixed FP, uranium and plutonium. The organic used to separate uranium and plutonium from the FP is 30 vol% tributyl phosphate (TBP) in a normal paraffin hydrocarbon (NPH) diluent.

The postulated cause of this accident is a failure of the interface float jumper on the HA column resulting in a partial loss of column contents to the SH sump. Subsequent ignition of the solvent mist results in burning the solvent in the sump. The postulated leak is located such that the column does not completely drain, with most of the aqueous phase remaining in the lower disengaging section where it is assumed to be evaporated by the fire. Leakage from the HA column would be detected by instruments monitoring column differential and static pressures, and a rapid increase in the SH sump WF would also result in an alarm. Column influents are postulated to be terminated within 5 min of leak initiation. During this 5 min interval some of the column contents are being drawn off at the normal rate and the influents contribute negligibly to the accident.

TABLE 9-29. Summary of Assumptions for Calculating Source Terms for the "1BX Column Criticality" Accident Scenario.

Assumptions	More likely	Worst case
Amount of Pu required for 1BX column criticality	>1.8 kg	1.8 kg
Time to accumulate critical mass in column	>2 h	2 h
Criticality parameters		
Pulses	1	48
Total fissions	10^{17}	1×10^{19} (15)
Duration of pulse	1 s	1 s initial, 0.5 s subsequent
Pulse frequency	-	1 pulse/10 min
Building release fractions: (15,36)		
Noble gases	1	1
Halogens	< 0.1	0.25
Time from gas generation to exit from stack	1 to 5 min	0 min

TABLE 9-30. Adverse Meteorology Summary: 1BX Column Criticality, 8-h Release from the 200-ft-high Stack.

Recipient	Distance/Direction	Atmospheric stability	X/Q[a] (s/m^3)	Wind speed
		Onsite		
Maximum worker	350 m NE	Unstable	6.5×10^{-5}	1.0 m/s
Workforce[b]	1,400-7,600 m W	Neutral	2.6×10^{-3}	1.0 m/s
		Offsite		
Maximum individual	15.5 km E	Moderate	1.3×10^{-6}	1.0 m/s
Population[c]	15-45 mi E	Moderate	6.1×10^{-3}	5 m/s

[a]Critical organ value; inert gases were computed separately.
[b]Workforce totals 2,096 persons, X/Q is population total.
[c]Population totals 109,411 persons, X/Q is population total.

TABLE 9-31. Radionuclide Release at Building Envelope from 1BX Column Criticality.

Isotope	Ci[a]	Isotope	Ci[a]
^{83m}Kr	1.1×10^{2}	^{135}Xe	4.1×10^{2}
^{85}Kr	8.1×10^{-4}	^{135m}Xe	3.3×10^{3}
^{85m}Kr	7.1×10^{1}	^{137}Xe	4.9×10^{4}
^{87}Kr	4.3×10^{2}	^{138}Xe	1.1×10^{4}
^{88}Kr	2.3×10^{2}	^{131}I	2.8×10^{0}
^{89}Kr	1.3×10^{4}	^{132}I	3.0×10^{2}
^{131m}Xe	1.0×10^{-1}	^{133}I	4.0×10^{1}
^{133}Xe	2.7×10^{1}	^{134}I	1.1×10^{3}
^{133m}Xe	2.2×10^{0}	^{135}I	1.1×10^{2}

[a]There would be other FP generated from the criticality, for example: ^{90}Sr, ^{91}Y, ^{95}Zr, ^{103}Ru, ^{106}Ru, ^{137}Cs, ^{141}Ce, ^{144}Ce, (these based on ^{239}Pu fission yields and 1×10^{19} fissions). However, it is believed that most of these would be contained within the column and any release would also be subject to additional filtration. Therefore, only the volatile isotopes are presented as significant releases.

TABLE 9-32. Doses Resulting from 1BX Column Criticality.

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	2.2×10^{-1}	2.2×10^{-1}	2.6×10^{-1}	6.3×10^{-1}	2.2×10^{-1}	2.2×10^{-1}	2.6×10^{-1}	6.3×10^{-1}
Workforce (man-rem)	2.2×10^{1}	2.2×10^{1}	3.3×10^{1}	7.3×10^{1}	2.2×10^{1}	2.2×10^{1}	3.3×10^{1}	7.3×10^{1}
Offsite								
Maximum individual (rem)	6.3×10^{-4}	1.7×10^{-3}	5.7×10^{-4}	9.7×10^{-2}	6.7×10^{-4}	1.9×10^{-3}	5.7×10^{-4}	1.0×10^{-1}
Population (man-rem)	7.0×10^{0}	9.0×10^{0}	8.7×10^{0}	2.2×10^{2}	7.0×10^{0}	9.3×10^{0}	8.7×10^{0}	2.2×10^{2}

A light water foam system in H cell and the adjacent ventilation tunnel is activated automatically if a fire is sensed by dual heat detectors in H cell.(37,38) Energized heat detectors would also actuate an alarm in the PUREX Plant control room and the 200 Area fire stations. The installed fire protection equipment would have to malfunction or not perform satisfactorily for an appreciable fire to develop and progress. The ventilation tunnel is equipped with an automatic water spray to protect the final exhaust filters from excessive heat and soot.

The fire could possibly burn at a rate of 6 in/h over the surface available in the sump (about 710 L/h), and would last 8 h, assuming no manual backup extinguishing systems were activated or other mitigating steps taken such as reducing cell ventilation flow or reducing the level of sump contents.(31,39)

Table 9-33 summarizes the assumptions which impact source term calculations for this scenario. They are listed as a "more likely" and "worst case" situations.

9.2.8.1 Accident-Induced Radionuclide Release Resulting from Solvent Fire in H Cell. Table 9-34 summarizes the meteorological conditions. The radionuclide release resulting from this accident in identified in Table 9-35. The release would be at the 291-A-1 stack, 200 ft above grade.

9.2.8.2 Dose Commitment. Onsite and offsite worst-case doses resulting from this accident are identified in Table 9-36. Two possible worst-case mixtures were tested.

9.2.8.3 Frequency of Occurrence. From Appendix D, the estimated frequency of a solvent fire in H cell is 8.6×10^{-6} a year.

9.2.9 Solvent Fire in L Cell

This accident involves the contents of the 3A column within L cell at the PUREX Plant.

The 3A column resembles the HA column in purpose, i.e., it is a dual purpose organic-continuous column for extracting the plutonium into an organic stream, and then scrubbing the organic to accomplish additional FP and metallic impurity decontamination. The organic solvent used for separation is a mixture of 30 vol% TBP in NPH diluent.

TABLE 9-33. Summary of Assumptions for the "Solvent Fire in H Cell" Accident Scenario.

Assumptions	More likely	Worst case
Fuel		
Type	Mark IV	Mark IA
Age	≥180 d	180 d
Percent ^{240}Pu	6% or 12%	6%
Duration of fire	<10 min	8 h
Rate of Organic Combustion	<2 in/h	6 in/h
Inventory involved in fire		
U	10 kg	400 kg
MFP* and Pu	Corresponding with the amount of U in fuel minus volatiles released during dissolution, etc.	
Release Factors(21,40)		
Volatiles (I)	<1.00	1.00
Semivolatiles (Cs, Ru)	0.20	0.80
Nonvolatiles	0.01	0.10
Actinides	<0.01	0.01
Ventilation DF	2,700(41)	1,000

*Mixed FP.

TABLE 9-34. Adverse Meteorology Summary: Solvent Fire H Cell, 8-h Release from the 200-ft-high Stack.

Recipient	Distance/Direction	Atmospheric stability	X/Q[a] (s/m^3)
	Onsite		
Maximum worker	350 m NE	Unstable	6.5×10^{-5}
Workforce[b]	1,400-7,600 m W	Neutral	1.3×10^{-2}
	Offsite		
Maximum individual	15.5 km E	Moderate	3.2×10^{-6}
Population[c]	15-45 mi SE	Moderate	2.7×10^{-2}

[a]Wind speed is 1 m/s in each case.

[b]Workforce totals 2,096 persons, X/Q is population total.

[c]Population totals 109,411 persons, X/Q is population total.

TABLE 9-35. Release at Building Envelope from Solvent Fire in H Cell. (Sheet 1 of 2)

Isotope	Ci[a]	Ci[b]
^{85}Kr	0	0
^{89}Sr	1.5×10^{0}	1.4×10^{-3}
^{90}Sr	1.5×10^{-1}	4.0×10^{-1}
^{90}Y	1.5×10^{-1}	4.0×10^{-1}
^{91}Y	2.2×10^{0}	4.8×10^{-3}
^{95}Zr	2.8×10^{0}	1.4×10^{-2}
^{95m}Nb	2.3×10^{-2}	1.6×10^{-6}
^{95}Nb	5.2×10^{0}	3.5×10^{-2}
^{103m}Rh	4.8×10^{0}	5.4×10^{-4}
^{103}Ru	4.8×10^{0}	5.4×10^{-4}
^{106}Rh	2.5×10^{0}	3.9×10^{0}
^{106}Ru	2.5×10^{0}	3.9×10^{0}
^{125}Sb	8.3×10^{-3}	2.1×10^{-2}
^{125m}Te	1.8×10^{-3}	4.8×10^{-3}
^{127}Te	1.6×10^{-2}	1.4×10^{-3}
^{127m}Te	1.7×10^{-2}	1.4×10^{-3}
^{129}Te	1.2×10^{-2}	2.1×10^{-7}
^{129m}Te	1.9×10^{-2}	3.4×10^{-7}
^{129}I	1.7×10^{-4}	5.5×10^{-4}
^{131}I	1.7×10^{-2}	0
^{131m}Xe	0	0
^{134}Cs	5.2×10^{-2}	3.2×10^{-1}

TABLE 9-35. Release at Building Envelope from Solvent Fire in H Cell. (Sheet 2 of 2)

Isotope	Ci[a]	Ci[b]
^{137m}Ba	1.2×10^{0}	3.4×10^{0}
^{137}Cs	1.2×10^{0}	3.6×10^{0}
^{141}Ce	5.8×10^{-1}	6.1×10^{-6}
^{144}Ce	3.1×10^{0}	2.0×10^{0}
^{144}Pr	3.1×10^{0}	2.0×10^{0}
^{147}Pm	5.4×10^{-1}	9.8×10^{-1}
^{148m}Pm	4.9×10^{-3}	3.7×10^{-6}
^{154}Eu	5.4×10^{-4}	4.2×10^{-3}
^{155}Eu	1.8×10^{-3}	4.2×10^{-3}
^{234}U	1.3×10^{-6}	1.3×10^{-6}
^{235}U	8.8×10^{-8}	6.9×10^{-8}
^{236}U	1.6×10^{-7}	2.6×10^{-7}
^{238}U	1.3×10^{-6}	1.3×10^{-6}
^{238}Pu	1.1×10^{-5}	1.5×10^{-4}
^{239}Pu	2.1×10^{-4}	4.7×10^{-4}
^{240}Pu	4.2×10^{-5}	2.7×10^{-4}
^{241}Pu	1.8×10^{-3}	2.6×10^{-2}
^{242}Pu	1.9×10^{-9}	9.7×10^{-8}
^{241}Am	1.6×10^{-6}	9.8×10^{-5}

[a]Releases based on Mark IA fuel, 6% ^{240}Pu, 180 d cooled. Based on the computed doses, this was worst-case.

[b]Releases based on Mark IA fuel, 12% ^{240}Pu, 2 yr cooled.

TABLE 9-36. Doses Resulting from Solvent Fire in H Cell.*

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	2.6×10^{-3}	2.9×10^{-2}	3.0×10^{-1}	4.6×10^{-4}	8.4×10^{-3}	1.2×10^{-1}	4.0×10^{-1}	4.6×10^{-4}
Workforce (man-rem)	5.1×10^{-1}	5.7×10^{0}	5.9×10^{1}	8.9×10^{-2}	1.6×10^{0}	2.2×10^{1}	7.8×10^{1}	8.9×10^{-2}
Offsite								
Maximum individual (rem)	9.2×10^{-4}	4.1×10^{-3}	1.5×10^{-2}	1.5×10^{-3}	1.2×10^{-2}	4.1×10^{-2}	2.2×10^{-2}	3.8×10^{-3}
Population (man-rem)	5.6×10^{0}	2.5×10^{1}	1.3×10^{2}	7.7×10^{0}	5.5×10^{1}	2.0×10^{2}	1.8×10^{2}	2.0×10^{1}

*Worst-case doses were found to come from MK IA at 6% ^{240}Pu aged 180 d.

In this accident, a rupture (possibly due to corrosion or a pressure surge) in the lower disengaging section of the column, during otherwise normal operation, empties the column contents to the cell floor and sump. Pressure fluctuations would indicate a major leak. Column influents then continue to flow for 5 min before being terminated. Based on this input, column size, leak location and other factors, the total amount of lost liquid is assumed to be composed of 115 L of solvent and 40 L of aqueous solution. This fills the sump and rises to a height of about 1/2 in. above the sump lip covering a surface area of about 250 ft^2 on the cell floor. The denser aqueous layer fills the sump and constitutes about 1/5 in. of the liquid on the floor with an interface area of about 110 ft^2 with the solvent. The agitation of the spilling organic may produce a solvent mist.

Ignition of the solvent mist results in burning over a decreasing surface area from 250 to 110 ft^2. Cell ventilation is assumed to sustain a combustion rate of 380 L/h (less than 1 in/h for the average available solvent surface area). At this burn rate the 115 L of solvent will be consumed in less than 20 min assuming the fire detection or supression system fails to perform satisfactorily.

Table 9-37 summarizes the assumptions which impact source term calculations for this scenario. They are listed as "more likely" and "worst case" situations.

9.2.9.1 Accident-Induced Radionuclide Release Resulting from a Solvent Fire in L Cell. The meteorology is summarized in Table 9-38. The radionuclide release resulting from this accident is identified in Table 9-39. The release would be at the 291-A-1 stack, 200 ft above grade.

9.2.9.2 Dose Commitment. Onsite and offsite worst-case doses resulting from this accident are identified in Table 9-40. Two possible worst cases are listed.

9.2.9.3 Frequency of Occurrence. From Appendix D, the estimated frequency of a solvent fire in L cell is 5.9×10^{-5} a year.

9.2.10 Plutonium Release via the P&O Gallery

This accident involves the release of plutonium and some FP through the P&O gallery and white room ventilation systems.

A pipe leak, similar to the one postulated in this scenario, resulted in the creation of the white room (westernmost 160 ft of P&O gallery) and the installation of HEPA filters on the exhaust from that zone.(42,43) Any new leaks involving solutions from cells adjacent to the white room would be subject to HEPA filtration. Leaks outside of this zone (much of the remainder of the P&O gallery) may be subject to temporary unfiltered release until detected and ventilation is diverted to the white room exhaust.

TABLE 9-37. Summary of Assumptions for the "Solvent Fire in L Cell" Accident Scenario.

Assumptions	More likely	Worst case
Fuel		
Type	Mark IV	Mark IA
Age	180 d or 2 yr	2 yr
Percent ^{240}Pu	6% or 12%	12%
Duration of fire	<10 min	20 min
Inventory involved in fire	<800 g Pu	1,800 g Pu
Release factors for burning solvent	<0.005	0.01
Canyon vent system DF (nonvolatiles)	≥2,700	1,000

TABLE 9-38. Adverse Meteorology Summary: Solvent Fire L Cell, 20-min Release from the 200-ft-high Stack.

Recipient	Distance/Direction	Atmospheric stability	X/Qa (s/m3)
		Onsite	
Maximum worker	350 m NE	Unstable	6.5×10^{-5}
Workforce[b]	1,400-7,600 m W	Neutral	1.3×10^{-2}
		Offsite	
Maximum individual	8.5 km SW	Moderate	1.5×10^{-5}
Population[c]	15-45 mi SE	Moderate	2.7×10^{-2}

[a]Wind speed is 1 m/s in each case.
[b]Workforce totals 2,096 persons, X/Q is population total.
[c]Population totals 109,411 persons, X/Q is population total.

TABLE 9-39. Release at Building Envelope from Solvent Fire in L Cell.

Isotope	Ci[a]	Ci[b]
^{238}Pu	5.6×10^{-5}	3.0×10^{-4}
^{239}Pu	1.1×10^{-3}	9.4×10^{-4}
^{240}Pu	2.1×10^{-4}	5.4×10^{-4}
^{241}Pu	8.8×10^{-3}	5.1×10^{-2}
^{242}Pu	9.2×10^{-9}	1.9×10^{-7}

[a]Releases based on Mark IA fuel, 6% ^{240}Pu, 180 d cooled.
[b]Releases based on Mark IA fuel, 12% ^{240}Pu, 2 yr cooled. Dose computations revealed this to be the worst case.

Airflow diversion is accomplished by a system of continuous air monitors (CAM) distributed throughout the P&O gallery, and provisions for automatic or manual power shut-off to exhausters, closing of dampers, and reduction of supply air.(44) An automatic response requires detection of elevated radiation levels [two maximum permissible concentration (MPC) hours] by more than one CAM. The manual action can be taken at any time based on an alarm from one CAM, visual detection of a leak, or any of a number of other indications. The CAM system reliability is backed by sensors which alarm if power to CAM is interrupted or an instrument fails.(44)

A leak of solution into the gallery must involve two simultaneous events: a breached line (corroded, sheared, or disconnected for some reason), and a force behind the solution sufficient to overcome both the elevation head and air pressure differences between canyon cells and the gallery. A line leak becomes more probable with time as copper tubing becomes brittle and solutions have longer to corrosively attack piping. Operational line opening is carefully controlled to prevent releases.(45)

The most likely "force" can result from steam condensation (in the event steam is shut off and the line is not purged) and the subsequent vacuum created.(42) These "suckbacks" are short lived and would only result in a small liquid leak. A more serious leak could occur if the radionuclide solution were under pressure. Pressurization of cell vessels could develop due to explosions (ammonia, hydrogen, red oil), vigorous chemical reactions, air pulser valve leaks, or attempts to unplug lines

TABLE 9-40. Doses Resulting from Solvent Fire in L Cell.*

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	8.5×10^{-6}	2.2×10^{-4}	8.3×10^{-3}	0	1.6×10^{-3}	3.6×10^{-2}	2.1×10^{-2}	0
Workforce (man-rem)	1.7×10^{-3}	4.2×10^{-2}	1.6×10^{0}	0	3.1×10^{-1}	7.0×10^{0}	4.0×10^{0}	0
Offsite								
Maximum individual (rem)	1.9×10^{-6}	5.0×10^{-5}	1.9×10^{-3}	0	3.6×10^{-4}	8.2×10^{-3}	4.7×10^{-3}	0
Population (man-rem)	3.6×10^{-3}	9.1×10^{-2}	3.5×10^{0}	0	6.6×10^{-1}	1.5×10^{1}	8.7×10^{0}	0

*Worst-case doses were found to come from the Mark IA 12% ^{240}Pu aged 2 yr.

with compressed gases (similar to the white room incident).(45) Such overpressurization should be detected by vessel pressure monitors, and relieved by venting through the vessel vent offgas system or through any penetrations caused by the event itself.

One plutonium leak which could occur outside the white room is associated with the product stream from the 1BS column. Some residual FP (primarily zirconium, niobium, and ruthenium) would also be associated with this leak. Of the 20 CAM located in the nonwhite room portion of the P&O gallery, only 2 are dedicated alpha monitors. Due to the "ventilation areas" created in the gallery by multiple exhausters,(46) it is possible only one of the alpha CAM would detect the plutonium. The lower levels of FP would require up to 12 min to activate one of the beta/gamma CAM once the postulated aerosol concentration reached the sampling port. Due to this delay, along with possible exhauster shut-down problems or calibration errors on CAM, an unfiltered release could continue for up to 20 min. Once the airflow is diverted through HEPA filtered exhaust, releases will be significantly reduced.

The potential also exists for liquid releases to the environment from the P&O gallery floor drains via a chemical sewer to a surface pond.(43) However, there are two different diversion systems that should contain both the initial leak and any clean-up solutions. First, once the leak is detected, the dispatcher can activate an air operated valve to send the P&O gallery drainage to the canyon.(47) This becomes automatic if power or air fails. Second, failing this or perhaps even before drains are diverted, monitors on the combined chemical sewer flow could detect elevated radiation levels and automatically divert the stream to a retention basin.

Table 9-41 summarizes the assumptions which impact source term calculations for this scenario. They are listed as "more likely" and "worst case" situations.

9.2.10.1 Accident-Induced Radionuclide Release Resulting from Plutonium Release via the P&O Gallery. The meteorological conditions are summarized in Table 9-42. The radionuclide release resulting from this accident is identified in Table 9-43. This release would be in two parts: the unfiltered portion, if any, would be released from vents near the top of the canyon building at 20 ft above grade; and the filtered portion would be released at the 296-A-8 stack, 78 ft above grade.

9.2.10.2 Dose Commitment. The onsite and offsite worst-case doses resulting from this accident are identified in Table 9-44. Two possible worst-case isotopic mixtures were tried.

9.2.10.3 Frequency of Occurrence. From Appendix D, the estimated frequency of a blowback into the P&O gallery, (the accident producing the most significant source terms) is 1×10^{-3} a year. The estimated frequency that the release will be totally unfiltered due to HEPA failure is 1.8×10^{-6} a year.

TABLE 9-41. Summary of Assumptions for the "Plutonium Release via the P&O Gallery" Accident Scenario.

Assumptions	More likely	Worst case
Fuel		
Type	Mark IV	Mark IA
Age	180 d	2 yr
Percent ^{240}Pu	6	12
Type of incident	Suckback	Blowback
Duration of leak	<10 min	2 h
Detection of leak	<2 min	12 min
Duration of unfiltered release to atmosphere	<2 min	20 min
Aerosol concentration resulting from leak	<1 mg/m^3	10 mg/m^3 (48)
Ventilation associated with aerosol concentration:		
Unfiltered portion	6,400 ft^3/min	16,000 ft^3/min
Filtered portion	22,500 ft^3/min	22,500 ft^3/min
Radionuclide content of aerosol	Pu in less concentrated vessel	Pu in 1BP
White room HEPA DF	3,300	2,000

TABLE 9-42. Adverse Meteorology Summary: Plutonium Release via P&O Gallery, 20-min Release from Ground Level.

Recipient	Distance and Direction	Atmospheric stability	X/Qa (s/m3)
	Onsite		
Maximum worker	350 m NE	Very stable	2.3×10^{-3}
Workforce[b]	1,400-7,600 m W	Very stable	1.0×10^{-1}
	Offsite		
Maximum individual	8.5 km E	Very stable	5.3×10^{-5}
Population[c]	15-45 mi SE	Moderate	2.5×10^{-2}

[a]Wind speed is 1 m/s in each case.
[b]Workforce totals 2,096 persons, X/Q is population total.
[c]Population totals 109,411 persons, X/Q is population total.

TABLE 9-43. Release at Building Envelope from P&O Gallery Blowback Involving Plutonium. (Sheet 1 of 2)

Isotope	Ci[a]	Ci[b]
^{85}Kr	0	0
^{89}Sr	5.4×10^{-4}	5.2×10^{-7}
^{90}Sr	5.5×10^{-5}	1.5×10^{-4}
^{90}Y	5.5×10^{-5}	1.5×10^{-4}
^{91}Y	8.2×10^{-4}	1.9×10^{-6}
^{95}Zr	1.0×10^{-3}	5.4×10^{-6}
^{95}Nb	1.9×10^{-3}	1.3×10^{-5}
^{95m}Nb	8.3×10^{-6}	4.5×10^{-8}
^{103m}Rh	2.2×10^{-4}	2.5×10^{-8}
^{103}Ru	2.2×10^{-4}	2.5×10^{-8}
^{106}Rh	1.1×10^{-4}	1.8×10^{-4}
^{106}Ru	1.1×10^{-4}	1.8×10^{-4}
^{125}Sb	3.1×10^{-6}	7.8×10^{-6}
^{125m}Te	7.6×10^{-7}	1.9×10^{-6}
^{127}Te	6.1×10^{-6}	5.1×10^{-7}
^{127m}Te	6.2×10^{-6}	6.3×10^{-7}
^{129}Te	7.1×10^{-6}	7.9×10^{-11}
^{129m}Te	7.2×10^{-6}	1.1×10^{-10}
^{129}I	7.8×10^{-11}	2.6×10^{-10}
^{131}I	7.9×10^{-9}	0
^{131m}Xe	0	0

TABLE 9-43. Release at Building Envelope from P&O Gallery Blowback Involving Plutonium. (Sheet 2 of 2)

Isotope	Ci[a]	Ci[b]
^{134}Cs	2.4×10^{-6}	1.5×10^{-6}
^{137m}Ba	5.4×10^{-5}	1.5×10^{-4}
^{137}Cs	5.7×10^{-5}	1.6×10^{-4}
^{141}Ce	2.1×10^{-4}	2.3×10^{-9}
^{144}Ce	1.1×10^{-3}	7.4×10^{-4}
^{144}Pr	1.1×10^{-3}	7.4×10^{-4}
^{147}Pm	2.0×10^{-4}	3.6×10^{-4}
^{234}U	6.5×10^{-10}	6.5×10^{-10}
^{235}U	4.4×10^{-11}	3.4×10^{-11}
^{236}U	8.0×10^{-11}	1.3×10^{-10}
^{238}U	6.5×10^{-10}	6.5×10^{-10}
^{238}Pu	2.2×10^{-4}	1.1×10^{-3}
^{239}Pu	4.1×10^{-3}	3.7×10^{-3}
^{240}Pu	8.2×10^{-4}	2.1×10^{-3}
^{241}Pu	3.4×10^{-2}	1.9×10^{-1}
^{242}Pu	3.6×10^{-8}	7.5×10^{-7}
^{241}Am	6.3×10^{-9}	3.8×10^{-7}

[a]Releases based on Mark IA fuel, 6% ^{240}Pu, 180 d cooled.

[b]Releases based on Mark IA fuel, 12% ^{240}Pu, 2 yr cooled. Dose calculations revealed this to be the worst case.

TABLE 9-44. Doses Resulting from Plutonium Release from Pipe and Operating Gallery.*

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	1.8×10^{-2}	4.5×10^{-1}	2.8×10^{-1}	1.8×10^{-9}	5.9×10^{-1}	1.4×10^{1}	2.8×10^{-1}	1.8×10^{-9}
Workforce (man-rem)	7.7×10^{-1}	2.0×10^{1}	1.2×10^{1}	7.9×10^{-8}	2.6×10^{1}	6.0×10^{2}	1.2×10^{1}	7.9×10^{-8}
Offsite								
Maximum individual (rem)	4.1×10^{-4}	1.0×10^{-2}	6.4×10^{-3}	4.2×10^{11}	1.4×10^{-2}	3.2×10^{-1}	6.5×10^{-3}	4.2×10^{11}
Population (man-rem)	1.9×10^{-1}	4.8×10^{0}	3.0×10^{0}	7.6×10^{-5}	6.4×10^{0}	1.5×10^{2}	3.0×10^{0}	3.9×10^{-3}

*Worst-case doses were found to come from the 12% ^{240}Pu Mark IA fuel 2 yr out of N Reactor.

9.2.11 Fission Product Release via the P&O Gallery

This accident involves the release of FP through the P&O gallery and main canyon ventilation systems.

This accident is comparable in many respects to the plutonium P&O gallery release scenario. The requirements for obtaining a significant leak and associated mitigating mechanisms are essentially the same. Only those aspects which differ from the plutonium release scenario are presented here.

The source of radionuclides is assumed to be vessel TK-F15 which contains concentrated high level waste (1WW). Sugar is added to reduce the nitric acid concentration, and if added too rapidly and/or to a cool 1WW solution which is subsequently heated, the tank could pressurize.(18) This procedure is normally carefully controlled and offgases are routinely exhausted through scrubbers, mist eliminators, an absorber, condenser, and the process vent system. If a leak pathway were simultaneously available during overpressurization, 1WW solution could spray into the eastern portion of the P&O gallery.

There are adequate beta/gamma CAM in this portion of the gallery(49) to expect a dual alarm in a matter of seconds after ventilation currents carry the aerosol to CAM sample ports.(49) This would result in automatic P&O gallery exhauster shutdown, damper closures, reduced supply air, and airflow diversion toward the white room and through the east change room to the canyon and main exhaust filter system.(50) Should the CAM be significantly out of calibration (unlikely because they are source checked monthly) or the automatic response systems fails in some respect, unfiltered releases could continue for up to 15 min.

Liquid releases resulting from the FP leak are subject to the same diversions identified for the plutonium P&O gallery release. The only differences are that eastern P&O gallery drains divert via an alternate pathway to a common canyon tank,(47) and the higher activity contamination would be more readily detected by the chemical sewer monitor.

Table 9-45 summarizes the assumptions which impact source term calculations for this scenario. They are listed as a "more likely" and "worst case" situations.

9.2.11.1 Accident-Induced Radionuclide Release Resulting from Fission Product Release via P&O Gallery. The meteorological conditions are summarized in Table 9-46. The radionuclide release resulting from this accident is identified in Table 9-47. The release would be in two parts: the unfiltered portion, if any, would be released from vents near the top of the canyon building, 20 ft above grade; and the filtered portions would be released at the 296-A-8 stack, 34 ft above grade, and the 291-A-1 stack, 200 ft above grade.

TABLE 9-45. Summary of Assumptions for the "Fission Product Release via the P&O Gallery" Accident Scenario.

Assumptions	More likely	Worst case
Fuel		
Type	Mark IA	Mark IA
Age	180 d or 2 yr	180d
Percent 240Pu	6 or 12	6
Type of incident	Suckback	Blowback
Duration of leak	<10 min	2 h
Detection of leak	<2 min	5 min
Duration of unfiltered release to atmosphere	<2 min	15 min
Aerosol concentration resulting from leak	<1 mg/m^3	10 mg/m^3
Ventilation associated with aerosol concentration		
Unfiltered portion	6,400 ft^3/min	16,000 ft^3/min
Filtered portion	11,000	11,000
Radionuclide content of aerosol	MFP in less concentrated tank	MFP in TK-F15
Canyon filter DF	2,700	1,000

TABLE 9-46. Adverse Meteorology Summary: Fission Product Release P&O Gallery, 15-min Release from Ground Level.

Recipient	Distance and Direction	Atmospheric stability	X/Qa (s/m3)
Onsite			
Maximum worker	350 m NE	Very stable	2.4×10^{-3}
Workforce[b]	1,400-7,600 m W	Very stable	1.0×10^{-1}
Offsite			
Maximum individual	8.5 km SW	Very stable	5.6×10^{-5}
Population[c]	15-45 mi SE	Moderate	2.5×10^{-2}

[a]Wind speed is 1 m/s in each case.
[b]Workforce totals 2,096 persons, X/Q is population total.
[c]Population totals 109,411 persons, X/Q is population total.

TABLE 9-47. Fission Product Release at Building Envelope from P&O Gallery Blowback. (Sheet 1 of 3)

Isotope	Ci[a]	Ci[b]
^{85}Kr	0	0
^{89}Sr	2.8×10^{0}	2.7×10^{-3}
^{90}Sr	2.9×10^{-1}	7.6×10^{-1}
^{90}Y	2.9×10^{-1}	7.6×10^{-1}
^{91}Y	4.3×10^{0}	1.0×10^{-2}
^{95m}Nb	4.6×10^{-2}	2.3×10^{-4}
^{95}Zr	5.5×10^{0}	2.8×10^{-2}
^{95}Nb	1.0×10^{1}	6.7×10^{-2}
^{103m}Rh	1.2×10^{0}	1.3×10^{-4}
^{103}Ru	1.2×10^{0}	1.3×10^{-4}
^{106}Rh	5.9×10^{-1}	9.5×10^{-1}
^{106}Ru	5.9×10^{-1}	9.5×10^{-1}
^{125}Sb	1.6×10^{-2}	4.0×10^{-2}
^{125m}Te	3.8×10^{-3}	9.6×10^{-3}
^{127}Te	3.2×10^{-2}	2.7×10^{-3}
^{127m}Te	3.2×10^{-2}	3.3×10^{-3}
^{129}Te	3.7×10^{-2}	4.1×10^{-7}
^{129m}Te	3.7×10^{-2}	5.9×10^{-7}
^{129}I	5.1×10^{-8}	1.7×10^{-7}
^{131}I	5.2×10^{-6}	0

9413219.1032

TABLE 9-47. Fission Product Release at Building Envelope from P&O Gallery Blowback. (Sheet 2 of 3)

Isotope	Ci[a]	Ci[b]
^{133m}Xe	0	0
^{134}Cs	1.2×10^{-2}	7.7×10^{-3}
^{137m}Ba	2.9×10^{-1}	8.1×10^{-1}
^{137}Cs	3.0×10^{-1}	8.5×10^{-1}
^{141}Ce	1.1×10^{0}	1.2×10^{-5}
^{144}Ce	5.9×10^{0}	3.8×10^{0}
^{144}Pr	5.9×10^{0}	3.8×10^{0}
^{147}Pm	1.0×10^{0}	1.9×10^{0}
^{148m}Pm	9.6×10^{-3}	0
^{154}Eu	1.0×10^{-3}	7.2×10^{-3}
^{155}Eu	3.4×10^{-3}	4.8×10^{-3}
^{234}U	2.3×10^{-12}	2.3×10^{-12}
^{235}U	1.5×10^{-13}	1.2×10^{-13}
^{236}U	2.8×10^{-13}	4.5×10^{-13}
^{238}U	2.3×10^{-12}	2.3×10^{-12}
^{238}Pu	3.3×10^{-10}	1.7×10^{-9}
^{239}Pu	6.2×10^{-9}	5.5×10^{-9}
^{240}Pu	1.2×10^{-9}	3.2×10^{-9}
^{241}Pu	5.1×10^{-8}	2.9×10^{-7}

9413219.1033

TABLE 9-47. Fission Product Release at Building Envelope from P&O Gallery Blowback. (Sheet 3 of 3)

Isotope	Ci[a]	Ci[b]
^{242}Pu	5.4×10^{-14}	1.1×10^{-12}
^{241}Am	3.3×10^{-5}	2.0×10^{-3}

[a]Releases based on Mark IA fuel, 6% ^{240}Pu, 180-d cooled. Dose calculations revealed this column to give higher doses.

[b]Releases based on Mark IA fuel, 12% ^{240}Pu, 2-yr cooled.

9.2.11.2 Dose Commitment. The onsite and offsite doses resulting from this accident are identified in Table 9-48.

9.2.11.3 Frequency of Occurrence. From Appendix D, the estimated frequency of occurrence of a blowback into the P&O gallery is 1×10^{-3} a year. This estimated frequency that the release will be totally unfiltered due to HEPA failure is 1.8×10^{-6} a year.

9.2.12 Oxide Line Fire

This accident involves a fire in the PUREX Plant N cell that results in a release of plutonium to the environment.

The PUREX plutonium nitrate product solution is converted to plutonium oxide in the recently modified N cell. This cell is well protected because it is located in the lowest elevation within the PUREX Plant canyon building. Both the N cell room exhaust and glovebox exhausts are filtered by multiple HEPA systems.(51) The only combustibles in N cell (possible accumulations of solid wastes), if ignited, would not result in any significant consequences. Some gloveboxes are protected with automatic Halon fire suppression systems and the cell is covered by an automatic wet pipe sprinkler system. A solvent fire, on the other hand, may not be readily extinguished within N cell. Several sources of solvent for such an accident have been considered.(52-54) The most credible solvent fire, yet still highly unlikely, is assumed to originate from a major solvent leak in R cell.

The R cell leak could be caused by the dropping of a cover block or heavy piece of equipment which breaks lines connected to a large storage tank. A water line must also be ruptured in order to float the solvent out of the cell, down the PR corridor, and into N cell. Such an incident should be readily apparent to the respective crane operator and personnel in N cell, who would be expected to take measures to alleviate the situation and prevent ignition. However, an electrical spark or hot machinery will be assumed to ignite the solvent, after personnel have evacuated.

A complex series of events is then required to provide a pathway to the environment for plutonium in the gloveboxes. The fire on the N cell floor melts or burns the gloves and/or window gaskets, and Benelex shielding, thereby allowing for possible entrainment of plutonium (about 1,500 g total of oxalate and oxide exposed) into cell air. Glovebox Halon systems would probably not be effective in suppressing a large cell fire, especially since ventilation would be maintained for filtered containment of any contamination. Also, water sprinkler systems have been shown to be unreliable in extinguishing solvent fires.(55)

TABLE 9-48. Doses Resulting from Fission Product Release from Pipe and Operating Gallery.*

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	3.4×10^{-1}	4.4×10^{0}	4.0×10^{0}	3.4×10^{-4}	7.6×10^{-1}	1.1×10^{1}	1.1×10^{1}	3.4×10^{-4}
Workforce (man-rem)	1.4×10^{1}	1.9×10^{2}	1.7×10^{2}	1.5×10^{-2}	3.2×10^{1}	4.7×10^{2}	1.7×10^{2}	1.5×10^{-2}
Offsite								
Maximum individual (rem)	8.0×10^{-3}	1.1×10^{-1}	9.6×10^{-2}	8.1×10^{-6}	1.8×10^{-2}	2.7×10^{-1}	9.6×10^{-2}	8.1×10^{-6}
Population (man-rem)	5.7×10^{0}	6.3×10^{1}	4.3×10^{1}	7.7×10^{-1}	7.1×10^{1}	3.5×10^{2}	4.6×10^{1}	3.6×10^{0}

*Worst-case doses were found to come from Mark IA fuel at 6% ^{240}Pu aged 180 d.

The first HEPA filters on both air systems (glovebox and cell) can be assumed to fail due to hot combustion gases. However, the next HEPA filter sets also receive cooler air from other areas and can be anticipated to plug from soot and fragments of breached filters rather than fail due to heat.(55-57) Once the exhaust is blocked by plugged HEPA filters, the supply air dampers automatically close to avoid pressurizing the cell. Although this may restrict the solvent burn rate, the release pathway now becomes more or less independent of the fire.

A plutonium release fraction of 5×10^{-2} (54,58) is assumed to be evenly entrained into 20 min of ventilation air plus the stagnant air volume of N cell and the PR corridor after ventilation ceases (Ref. 54 and 58 through 61). Air is assumed to exchange between glovebox and cell due to failed gloves/windows and plugging of the glovebox exhaust HEPA filters slightly before those in the cell. Releasing plutonium at the 5% rate into the "post-ventilation" air is very conservative since entrainment should decrease once the air flow and fire intensity are reduced. The stagnant air is assumed to migrate (diffusion, convection, pressure fluctuations) to the canyon lobby via stairwell and elevator shafts. Since the lobby is also without ventilation (part of the same system) some of this air may be released outside. A worst-case estimate will be 5% although a greater release could occur if the supply air did not shut down immediately (a nonrelated and therefore, multiple accident).(60)

Table 9-49 summarizes the assumptions which impact source term calculations for this scenario. They are listed as "more likely" and "worst-case" situations.

9.2.12.1 Accident-Induced Radionuclide Release Resulting from Oxide Line Fire. The meteorological conditions are summarized in Table 9-50. The radionuclide release resulting from this accident is identified in Table 9-51. This release would be in two parts: the unfiltered portion would be released at the canyon lobby door at ground level; and the filtered portions at stacks 296-A-1 and 296-A-7 at 74 and 78 ft above grade, respectively.

9.2.12.2 Dose Commitment. The onsite and offsite doses resulting from this accident are identified in Table 9-52.

9.2.12.3 Frequency of Occurrence. From Appendix D, the estimated frequency of an oxide line fire in N cell is 3.7×10^{-5} a year.

9.2.13 Cooling Coil Leak with Release to 216-A-25 Pond

This accident involves a leak in the PUREX metal solution TK-D5 cooling coil with a subsequent release of radionuclides to the surface disposal site, 216-A-25 pond.

TABLE 9-49. Summary of Assumptions for the "Oxide Line Fire" Accident Scenario.

Assumptions	More likely[a]	Worst case
Fuel		
Type	Mark IA	Mark IA
Age	180 d or 2 yr	2 yr
Percent ^{240}Pu	6 or 12	12
Amount of Pu		
Exposed		
PuO_2	0	1,000 g
$Pu(C_2O_4)_2 \cdot 6H_2O$	500 g	500 g
Total Pu	240 g	1,110 g
Release fraction of Pu into air	$< 5 \times 10^{-2}$	5×10^{-2}
Time to		
Fail 1st set of HEPA in N cell and glovebox exhaust systems	>10 min	10 min
Plug 2nd set of HEPA in each system after first set fails	Not plugged	10 min
Extinguish fire - from ignition	<20 min	3 h[b]
Volume of air in which Pu is entrained		
Exhausted from HEPA filters	>1,000 m^3	1,000 m^3
Not exhausted to HEPA	None	N cell and PR corridor volume (1,300 m^3)
DF for HEPA filtered release	>3,300	2,000
Percent of stagnant air that leaks to environment	None	5% (of 1,300 m^3)

[a]This "more likely" column is based in part on a scenario in which several hundred gallons (versus 6,000 gal involved in worst case) of solvent (which could be on the floor in R cell) is floated out of cell by a water line break.

[b]Any burn time in excess of the 20 min needed to produce plutonium-contaminated stagnant air does not significantly impact the source terms. It does, however, represent a possible hindrance to resolving the plutonium release in a timely fashion.

TABLE 9-50. Adverse Meteorology Summary: Oxide Line Fire, 3-h Release from Ground Level.

Recipient	Distance and Direction	Atmospheric stability	X/Q[a] (s/m^3)
	Onsite		
Maximum worker	350 m NE	Very stable	1.0×10^{-3}
Workforce[b]	1,400-7,600 m W	Very stable	1.0×10^{-1}
	Offsite		
Maximum individual	8.5 km E	Very stable	2.5×10^{-5}
Population[c]	15-45 mi SE	Moderate	2.5×10^{-2}

[a]Wind speed is 1 m/s in each case.
[b]Workforce totals 2,096 persons, X/Q is population total.
[c]Population totals 109,411 persons, X/Q is population total.

TABLE 9-51. Release at Building Envelope from Oxide Line Fire.

Isotope	Ci
^{238}Pu	2.7×10^{-2}
^{239}Pu	8.4×10^{-2}
^{240}Pu	4.8×10^{-2}
^{241}Pu	4.6×10^{0}
^{242}Pu	1.7×10^{-5}

TABLE 9-52. Doses Resulting from Oxide Line Fire.

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	1.2×10^{-2}	3.0×10^{-1}	1.1×10^{1}	0	2.2×10^{0}	5.0×10^{1}	2.9×10^{1}	0
Workforce (man-rem)	1.2×10^{0}	3.0×10^{1}	1.2×10^{3}	0	2.2×10^{2}	5.0×10^{3}	2.9×10^{3}	0
Offsite								
Maximum individual (rem)	3.0×10^{-4}	7.5×10^{-3}	2.9×10^{-1}	0	5.4×10^{-2}	1.2×10^{0}	7.1×10^{-1}	0
Population (man-rem)	2.9×10^{-1}	7.5×10^{0}	2.8×10^{2}	0	5.4×10^{1}	1.2×10^{3}	7.1×10^{2}	0

There are several PUREX radioactive service tanks with cooling coils which discharge to surface disposal sites.[61] The cooling water discharge stream is continuously monitored for alpha, beta, and gamma radiation and is equipped for automatic diversion to retention basin 216-A-42.[62,63] Coil leaks have occurred and can result in a release to a pond if diversion fails.[64,65]

Solutions in TK-D3, -D4, and -D5 have one of the highest radionuclide concentrations of tanks with cooling water discharges to ponds. The coil in TK-D5 is the largest of the three tank coils and has been selected for a postulated leak scenario. The tank coil is periodically leak checked; however, a leak could develop at any time.[66] The TK-D5 functions as a storage/sampling tank for metal solutions from the fuel dissolvers and is batch transferred an average of 1.5 times a day.

The cooling water supply and discharge headers are located such that the pressure head within the coil is always greater than the tank contents. A shut-off valve bypass with a flow rate of 3 gal/min (Drawing H-2-63219) has been installed to prevent drainage of the coil in the simultaneous event that the valve is closed, the coil is breached, and the tank is empty. Such drainage could occur only if the leak exceeds 3 gal/min or the bypass is plugged, because if the main supply valve were shut off (no bypass around it) procedure calls for hydrostatic leak tests of coils before reopening this valve. Refilling of a drained coil with tank solution can then only occur, obviously, if the tank is refilled prior to introducing cooling water to the coil.

Even if this sequence occurs, the cooling water monitor would detect the elevated radiation and alarm if above a safe discharge level. There are also alarms, located in the control room, for sample flow deviation and monitor failure. These alarms would have to be deactivated (coupled with a failure) or the monitor incorrectly calibrated for the discharge to go undetected. This is considered a double accident (occurring coincidentally with a leak) although both are necessary for a release to the environment. Alternately, a coil leak could be indicated by changing tank solution specific gravity (especially for relatively empty tanks) or volume (more noticeable if the tank is near full and therefore closely watched) due to in-leaking cooling water. Another indicator would be the weekly radionuclide analyses performed on proportional effluent samples. It is this last method which is considered worst case for duration of an unrecognized leak condition.

Once discharged to the pond, the radionuclides are still isolated from the public. When the leak is discovered, special procedures (maintain pond level, discourage animal access, add chelating or precipitating agents, etc.) will be undertaken to minimize releases from the pond. However, during the assumed 8 d of undetected leakage, some evaporational releases to the atmosphere could occur. Releases might also result from nuclides deposited along the shore by wave action or left behind if the pond level drops, if they dry and are suspended by the wind.

Table 9-53 summarizes the assumptions which impact source term calculations for this scenario. They are listed as "more likely" and "worst case" situations.

9.2.13.1 Accident-Induced Radionuclide Release Resulting from Cooling Coil Leak. The meteorological conditions are summarized in Table 9-54. The radionuclide release resulting from this accident is identified in Table 9-55. This release would be at the 216-A-25 pond surface at ground level.

9.2.13.2 Dose Commitment. The onsite and offsite doses resulting from this accident are identified in Table 9-56.

9.2.13.3 Frequency of Occurrence. From Appendix D, the estimated frequency of occurrence of a coil leak is 1.3×10^{-6} a year.

9.2.14 Personnel Overexposure During Fuel Charging

This accident involves exposure of an individual to unshielded fuel canisters within the PUREX Plant railroad tunnel during a dissolver charging operation.

There are extensive procedural and physical safeguards which should prevent such an occurrence (Ref. 18 and 68 through 71). In addition, the high noise level generated while removing a cask lid would alert anyone within the tunnel to the impending operations. The crane operator will also sound a warning claxon to signal evacuation.

Tunnel access doors are clearly marked as leading to a high radiation area and are controlled by the dispatcher through electric door locks and door position indicators. As a designated radiation area, workers must wear protective clothing and contamination and dose rate monitoring is required. Qualified personnel are authorized to self-survey and to self-monitor up to 100 mrad/h. Above this limit, workers must be accompanied at all times by a Radiation Protection Technologist. The dispatcher logs the name and entry and exit times for all personnel involved with any tunnel work. Prior to a dissolver charge, generally two operators and a Radiation Protection Technologist enter the tunnel to prepare the cask cars (engaging car puller and opening metal well doors). Procedures explicitly state that all personnel must be evacuated from the tunnel prior to any canister unloading.

An accidental exposure revolves around the departure of all but one individual followed by improper actions from his partners and/or the dispatcher. Due to the presence of the dispatcher's TV monitor and the crane operator's periscope, the closest he would be to a canister without being detected is on the floor beside the car to be unloaded. The worker's white SWP could easily blend against the white floor and he may be partially hidden by an open door on the cask car.

TABLE 9-53. Summary of Assumptions for the Calculating Source Terms for the "Cooling Coil Leak with Release to 216-A-25 Pond" Accident Scenario.

Assumptions	More likely	Worst case
Tank Inventory		
Volume[a]	5,000 gal	5,000 gal
Fuel		
Type	Mark IV	Mark IA
Percent ^{240}Pu	6% or 12%	12%
Age	≥180 d	2 yr
Time to detect and terminate leak	<1 d[b]	8 d
Volume of contaminated solution released during each tank batch cycle	<1 coil vol	1 coil vol (86 gal)
Number of tank cycles involved with a leak	1	12[c]
Isotopic inventory of solution which enters coil (U)[d]	410 g/L	380 g/L
Contaminated pond	216-B-3 and/or 216-A-25	216-A-25
Pond resuspension factors from evaporation (40)[e]		
I	<1.0	1.0
Semivolatiles (Cs, Ru)	$<1 \times 10^{-3}$	1×10^{-2}[f]
Nonvolatiles	5.7×10^{-7} to 1×10^{-10} (40)	1×10^{-5}

[a]This has little bearing on release estimates since the only mechanism which could drain the entire tank contents (siphoning through a large coil leak) is not possible.

[b]This is based on closer timing between sampling and the leak or on tank specific gravity readings. Obviously, it is most likely that the diversion monitor system will work and there will be little or no release to ponds. (No release might be possible since monitor is located well upstream of point of diversion.)

[c]Based on 8 d x 1.5 cycles a day.

[d]This uranium concentration is tied to the original fuel properties of various isotopes. The more likely is based on Mark IV while the worst case is for MK IA.(67) Although the uranium concentration is smaller, the MK IA still contains a greater isotope inventory than Mark IV. Volatile isotopes in fuel were included although their prior release in the DOG or VOG systems is likely.

[e]Based on a drying experiment for plutonium nitrate [$Pu(NO_3)_4$] at 25°C to 90°C and 10 to 100 cm/s air velocity.

[f]This accounts for the potential for release mechanisms other than evaporation.

TABLE 9-54. Adverse Meteorology Summary: Cooling Coil Leak, 8-d Release from Ground Level.

Recipient	Distance and Direction	X/Q[a] (s/m^3)
	Onsite	
Maximum worker workforce[b]	3 km S 3-9 km S-SW	1.7×10^{-6} 2.7×10^{-3}
	Offsite	
Maximum individual Population[c]	15.5 km E 10-45 mi SE-S	1.4×10^{-6} 4.3×10^{-3}

[a]Model used obtained from NRC Regulatory Guide 1.3, in which Pasquill-Gifford classes, C, D, and F are equally weighted and sector averaged. The wind speed for classes C and D is 3 m/s, while the wind speed for class F is 2 m/s. In addition, the wind only heads in the worst case direction 1/3 of the time. Three sectors were thus used for population X/Q determination.

[b]Workforce totals 3,025 persons, X/Q is population total.

[c]Population totals 137,519 persons, X/Q is population total.

TABLE 9-55. Release to Atmosphere from Cooling Coil Leak to 216-A-25 Pond. (Sheet 1 of 2)

Isotope	Ci[a]	Ci[b]
^{85}Kr	0	0
^{89}Sr	5.5×10^{-1}	5.1×10^{-4}
^{90}Sr	5.5×10^{-2}	1.5×10^{-1}
^{90}Y	5.5×10^{-2}	1.5×10^{-1}
^{91}Y	8.2×10^{-1}	1.8×10^{-3}
^{95m}Nb	8.7×10^{-3}	4.9×10^{-5}
^{95}Zr	1.0×10^{0}	5.2×10^{-3}
^{95}Nb	1.9×10^{0}	1.3×10^{-2}
^{103m}Rh	2.2×10^{2}	2.5×10^{-2}
^{103}Ru	2.2×10^{2}	2.5×10^{-2}
^{106}Rh	1.1×10^{2}	1.8×10^{2}
^{106}Ru	1.1×10^{2}	1.8×10^{2}
^{125}Sb	3.1×10^{-3}	7.7×10^{-3}
^{125m}Te	6.7×10^{-4}	1.8×10^{-3}
^{127}Te	6.0×10^{-3}	5.0×10^{-4}
^{127m}Te	6.1×10^{-3}	5.1×10^{-4}
^{129}Te	4.5×10^{-3}	7.9×10^{-8}
^{129m}Te	7.1×10^{-3}	1.3×10^{-7}
^{129}I	1.2×10^{-3}	4.1×10^{-3}
^{131}I	1.2×10^{-1}	0
^{131m}Xe	0	0
^{134}Cs	2.4×10^{0}	1.5×10^{1}
^{137m}Ba	5.4×10^{1}	1.6×10^{2}

TABLE 9-55. Release to Atmosphere from Cooling Coil Leak to 216-A-25 Pond. (Sheet 2 of 2)

Isotope	Ci[a]	Ci[b]
^{137}Cs	5.7×10^{1}	1.6×10^{2}
^{141}Ce	2.1×10^{-1}	2.3×10^{-6}
^{144}Ce	1.1×10^{0}	7.4×10^{-1}
^{144}Pr	1.1×10^{0}	7.4×10^{-1}
^{147}Pm	2.0×10^{-1}	3.6×10^{-1}
^{148m}Pm	1.8×10^{-3}	1.4×10^{-6}
^{154}Eu	2.0×10^{-3}	1.6×10^{-3}
^{155}Eu	6.6×10^{-3}	1.6×10^{-3}
^{234}U	4.9×10^{-4}	4.9×10^{-6}
^{235}U	3.2×10^{-7}	2.5×10^{-7}
^{236}U	6.0×10^{-7}	9.5×10^{-7}
^{238}U	4.9×10^{-6}	4.9×10^{-6}
^{238}Pu	4.2×10^{-5}	5.6×10^{-4}
^{239}Pu	7.8×10^{-4}	1.7×10^{-3}
^{240}Pu	1.5×10^{-4}	1.0×10^{-3}
^{241}Pu	6.5×10^{-3}	9.5×10^{-2}
^{242}Pu	6.9×10^{-9}	3.5×10^{-7}
^{241}Am	5.7×10^{-6}	3.6×10^{-4}

[a]Releases based on Mark IA fuel, 6% ^{240}Pu, 180-d cooled.

[b]Releases based on Mark IA fuel, 12% ^{240}Pu, 2-yr cooled. Dose calculations revealed this column to give the higher doses.

9101.6124/14

TABLE 9-56. Doses Resulting from Cooling Coil Leak to 216-A-25 Pond.*

Recipient	1 yr				50 yr			
	Total body	Bone	Lung	Thyroid	Total body	Bone	Lung	Thyroid
Onsite								
Maximum worker (rem)	1.7×10^{-3}	3.3×10^{-3}	9.7×10^{-4}	3.5×10^{-6}	2.0×10^{-3}	6.1×10^{-3}	1.0×10^{-3}	3.8×10^{-6}
Workforce (man-rem)	2.8×10^{0}	5.2×10^{0}	1.6×10^{0}	5.6×10^{-3}	3.3×10^{0}	9.8×10^{0}	1.6×10^{0}	6.1×10^{-3}
Offsite								
Maximum individual (rem)	4.3×10^{-3}	4.4×10^{-3}	8.1×10^{-4}	7.1×10^{-4}	2.0×10^{-2}	2.2×10^{-2}	1.3×10^{-2}	1.4×10^{-2}
Population (man-rem)	8.9×10^{1}	9.4×10^{1}	1.8×10^{1}	1.5×10^{1}	4.0×10^{2}	4.4×10^{2}	2.7×10^{2}	2.7×10^{2}

*Worst-case doses were found to come from the Mark IA fuel, 12% ^{240}Pu, 2 yr out of N Reactor.

It is conceivable that an individual's exposure could accrue from the unloading of 27 canisters (three railcars), whereupon he would be discovered when the tunnel is re-entered. However, his absence should be detected within 2 h, when he would be missed at a break time or shift change, which corresponds to the unloading of nine canisters (one railcar). The canisters are assumed to contain 180-d cooled, 6% ^{240}Pu Mark IV fuel. Based on the unloading cycle of crane operation, the individual would be exposed to each canister at varying distances (18 to 30 ft) for up to 3 min. Depending on the exact location, residual contaminated water draining from the canisters passing overhead could contact the individual. Protective clothing and respirator should reduce the chances for any internal deposition or skin contact.

Table 9-57 summarizes the assumptions necessary to calculate exposure dosage. The "more likely" column reflects alternative locations for the individual, shorter period until missed or detected, faster crane operation, and longer cooled fuel. However, the worst case assumptions are all combined for a very conservative dose estimate.

No radionuclide releases are associated with this accident; therefore, there are no offsite dose results from this accident. Onsite dose is confined to the individual located in the railroad tunnel, who would receive an estimated 74 rem.

No frequency of occurrence was developed for this accident as it was not associated with a radionuclide release.

TABLE 9-57. Summary of Assumptions for the "Personnel Overexposure During Fuel Charging" Accident Scenario.

Assumptions	More likely	Worst case
Fuel		
Type	Mark IV	Mark IV[a]
Age	≥180 d	180 d
Percent ^{240}Pu	6% or 12%	6%[b]
Exposure source (canister number)[b]	≤9	27
Distance to source		
Closest	>18 ft	18 ft
Furthest	>30 ft	30 ft
Length of exposure (per canister)	3 min	3 min

[a]Mark IA would contain a higher radionuclide content; however, Mark IV canisters typically contain 40% more fuel than Mark IA.

[b]6% fuel aged 180 d is worse than 12% fuel aged 2 yr.

[c]The exposure per canister at 18 ft would be around 100 R/h, based on 1.295×10^3 R/h at 5 ft for 180-d cooled, 6% ^{240}Pu, Mark IV fuel.(72)

9.3 EVALUATION OF RESULTS

The dose consequences of several postulated accidents have been determined in this analysis. Dose consequences of postulated external phenomena are summarized in Section 9.2.1. Direct radiation dose consequences of a postulated N cell criticality event and fuel charging accident are summarized in Sections 9.2.2 and 9.2.14, respectively. All other accidents investigated resulted in airborne releases, the consequences of which are summarized in Table 9-58. To evaluate their acceptability, these consequences were compared to the criteria of Reference 73.

9.3.1 Hazard Level

Reference 73 specifies criteria for the determination of hazard level based in terms of potential for annual radiation exposure. These criteria are listed in Table 9-59.

With the exception of three accidents, the seismic event, the N cell criticality event, and overexposure during fuel charging, all accidents analyzed were found to be in the low-hazard-level category when compared with criteria listed in Table 9-59.

The seismic event was determined to have the potential for significant inhalation and immersion exposure, including a potential for lethal exposure to individuals in close proximity.(11,12) The tornado event was evaluated as having minimal radiological consequences.

The N cell criticality was evaluated as having the potential for significant direct radiation exposure (greater than 25 rem) to nearby personnel, including a potential for lethal exposure to individuals in close proximity (a few feet) to the event. Due to the recognized severity of this type of accident, a triple contingency safety factor has been built into the operation of the oxide production facility, requiring the simultaneous failure of three engineered or administrative safety features before a criticality is possible.

The overexposure during fuel charging was evaluated using highly conservative assumptions as having the potential for direct radiation exposure of 74 rem to a nearby individual.

Due to the extremely low probability of these events, all accidents evaluated meet the criteria for acceptable risk as described in Section 9.3.2.

9.3.2 Risk Acceptability

Risk acceptance guidelines have been implemented by Rockwell to provide assurance that risks have been limited to a level which is acceptable. Risk criteria are discussed in Section 2.5.2.

TABLE 9-58. Accident Analysis Summary. (Sheet 1 of 2)

Accident	Accident frequency (yr^{-1})	Organ of concern	Dose in rem to maximum onsite individual		Dose in rem to maximum offsite individual		Dose risk (rem/yr) to maximum onsite individual	Dose risk (rem/yr) to maximum offsite individual
			1 yr	50 yr*	1 yr	50 yr*	1 yr	1 yr
Criticality in the PuO_2 production facility	7.5×10^{-5}	Total Body	6.0×10^{-1}	6.0×10^{-1}	3.0×10^{-3}	3.0×10^{-3}	4.5×10^{-5}	2.3×10^{-7}
		Bone	6.0×10^{-1}	6.0×10^{-1}	3.3×10^{-3}	3.3×10^{-3}	4.5×10^{-5}	2.5×10^{-7}
		Lung	8.3×10^{-1}	8.3×10^{-1}	4.3×10^{-3}	4.3×10^{-3}	6.3×10^{-5}	3.2×10^{-7}
		Thyroid	2.7×10^{0}	2.7×10^{0}	9.0×10^{-2}	9.3×10^{-2}	2.0×10^{-4}	6.8×10^{-6}
Dissolving short-cooled fuel	2.5×10^{-7}	Total Body	4.2×10^{-2}	4.5×10^{-2}	5.5×10^{-3}	6.9×10^{-3}	1.1×10^{-8}	1.4×10^{-9}
		Bone	7.1×10^{-2}	1.3×10^{-1}	7.4×10^{-3}	1.3×10^{-2}	1.8×10^{-8}	1.9×10^{-9}
		Lung	6.1×10^{-2}	6.1×10^{-2}	4.3×10^{-3}	4.3×10^{-3}	1.5×10^{-8}	1.1×10^{-9}
		Thyroid	6.8×10^{-1}	6.8×10^{-1}	1.0×10^{0}	1.1×10^{0}	1.7×10^{-7}	2.5×10^{-7}
U Metal fire in a dissolver	2.0×10^{-3}	Total Body	5.9×10^{-3}	2.4×10^{-2}	2.5×10^{-3}	2.9×10^{-2}	1.2×10^{-5}	5.0×10^{-6}
		Bone	6.2×10^{-2}	3.6×10^{-1}	9.2×10^{-3}	1.0×10^{-1}	1.2×10^{-4}	1.8×10^{-5}
		Lung	8.3×10^{-1}	1.1×10^{0}	4.1×10^{-3}	6.2×10^{-2}	1.7×10^{-3}	8.2×10^{-6}
		Thyroid	9.2×10^{-4}	9.3×10^{-4}	3.2×10^{-3}	9.8×10^{-3}	1.8×10^{-6}	6.4×10^{-6}
Silver reactor explosion	1.0×10^{-3}	Whole Body	4.3×10^{-4}	4.6×10^{-4}	1.3×10^{-3}	1.2×10^{-2}	4.3×10^{-7}	1.3×10^{-6}
		Bone	4.3×10^{-3}	8.9×10^{-2}	1.0×10^{-3}	2.1×10^{-2}	4.3×10^{-6}	1.0×10^{-6}
		Lung	1.8×10^{-3}	1.8×10^{-3}	2.0×10^{-4}	5.2×10^{-3}	1.8×10^{-6}	2.0×10^{-7}
		Thyroid	1.3×10^{-1}	1.4×10^{-1}	9.2×10^{-1}	2.9×10^{0}	1.3×10^{-4}	9.2×10^{-4}
H_2 explosion in a dissolver	1.2×10^{-3}	Whole Body	4.4×10^{-5}	1.2×10^{-4}	3.1×10^{-5}	1.2×10^{-4}	5.3×10^{-8}	3.7×10^{-8}
		Bone	4.9×10^{-4}	1.9×10^{-3}	7.7×10^{-5}	4.2×10^{-4}	5.9×10^{-7}	4.1×10^{-8}
		Lung	4.4×10^{-4}	4.4×10^{-4}	2.5×10^{-5}	3.4×10^{-5}	5.3×10^{-7}	3.7×10^{-7}
		Thyroid	4.8×10^{-3}	4.8×10^{-3}	1.6×10^{-2}	1.9×10^{-2}	5.8×10^{-6}	1.9×10^{-5}
In-canyon criticality	1.5×10^{-2}	Whole Body	2.2×10^{-1}	2.2×10^{-1}	6.3×10^{-4}	6.7×10^{-4}	3.3×10^{-3}	9.5×10^{-6}
		Bone	2.2×10^{-1}	2.2×10^{-1}	1.7×10^{-3}	1.9×10^{-3}	3.3×10^{-3}	2.6×10^{-5}
		Lung	2.6×10^{-1}	2.6×10^{-1}	5.7×10^{-4}	5.7×10^{-4}	3.9×10^{-3}	8.6×10^{-6}
		Thyroid	6.3×10^{-1}	6.3×10^{-1}	9.7×10^{-2}	1.0×10^{-1}	9.5×10^{-3}	1.5×10^{-3}

TABLE 9-58. Accident Analysis Summary. (Sheet 2 of 2)

Accident	Accident frequency (yr^{-1})	Organ of concern	Dose in rem to maximum onsite individual		Dose in rem to maximum offsite individual		Dose risk (rem/yr) to maximum onsite individual	Dose risk (rem/yr) to maximum offsite individual
			1 yr	50 yr*	1 yr	50 yr*	1 yr	1 yr
Solvent fire in H Cell	8.6×10^{-6}	Whole Body	2.6×10^{-3}	8.4×10^{-3}	9.2×10^{-4}	1.2×10^{-2}	2.2×10^{-8}	7.9×10^{-9}
		Bone	2.9×10^{-2}	1.2×10^{-1}	4.1×10^{-3}	4.1×10^{-2}	2.5×10^{-7}	3.5×10^{-8}
		Lung	3.0×10^{-1}	4.0×10^{-1}	1.5×10^{-2}	2.2×10^{-2}	2.6×10^{-6}	1.3×10^{-7}
		Thyroid	4.6×10^{-4}	4.6×10^{-4}	1.5×10^{-3}	3.8×10^{-3}	4.0×10^{-9}	1.3×10^{-8}
Solvent fire in L Cell	5.9×10^{-5}	Whole Body	8.5×10^{-6}	1.6×10^{-3}	1.9×10^{-6}	3.6×10^{-2}	5.0×10^{-10}	1.1×10^{-10}
		Bone	2.2×10^{-4}	3.6×10^{-2}	5.0×10^{-5}	8.2×10^{-3}	1.3×10^{-8}	3.0×10^{-9}
		Lung	8.3×10^{-3}	2.1×10^{-2}	1.9×10^{-3}	4.7×10^{-3}	4.9×10^{-7}	1.1×10^{-7}
		Thyroid	0	0	0	0	0	0
Pu release via P&O gallery	1.8×10^{-6}	Whole Body	1.8×10^{-2}	5.9×10^{-1}	4.1×10^{-4}	1.4×10^{-2}	3.2×10^{-8}	7.4×10^{-10}
		Bone	4.5×10^{-1}	1.4×10^{1}	1.0×10^{-2}	3.2×10^{-1}	8.1×10^{-7}	1.8×10^{-8}
		Lung	2.8×10^{-1}	2.8×10^{-1}	6.4×10^{-3}	6.5×10^{-3}	5.0×10^{-7}	1.2×10^{-8}
		Thyroid	1.8×10^{-9}	1.8×10^{-9}	4.2×10^{-11}	4.2×10^{-11}	3.2×10^{-15}	7.6×10^{-17}
FP release via P&O gallery	1.8×10^{-6}	Whole Body	3.4×10^{-1}	7.6×10^{-1}	8.0×10^{-3}	1.8×10^{-2}	6.1×10^{-7}	1.4×10^{-8}
		Bone	4.4×10^{0}	1.1×10^{1}	1.1×10^{-1}	2.7×10^{-1}	7.9×10^{-6}	2.0×10^{-7}
		Lung	4.0×10^{0}	4.0×10^{0}	9.6×10^{-2}	9.6×10^{-2}	7.2×10^{-6}	1.7×10^{-7}
		Thyroid	3.4×10^{-4}	3.4×10^{-4}	8.1×10^{-6}	8.1×10^{-6}	6.1×10^{-10}	1.5×10^{-11}
Oxide line fire	3.7×10^{-5}	Whole Body	1.2×10^{-2}	2.2×10^{0}	3.0×10^{-4}	5.4×10^{-2}	4.4×10^{-7}	1.1×10^{-8}
		Bone	3.0×10^{-1}	5.0×10^{1}	7.5×10^{-3}	1.2×10^{0}	1.1×10^{-5}	2.8×10^{-7}
		Lung	1.1×10^{1}	2.9×10^{1}	2.9×10^{-1}	7.1×10^{-1}	4.1×10^{-4}	1.1×10^{-5}
		Thyroid	0	0	0	0	0	0
Cooling coil leak to 216-A-25 pond	1.3×10^{-7}	Whole Body	1.7×10^{-3}	2.0×10^{-3}	4.3×10^{-3}	2.0×10^{-2}	2.2×10^{-10}	5.6×10^{-10}
		Bone	3.3×10^{-3}	6.1×10^{-3}	4.4×10^{-3}	2.2×10^{-2}	4.3×10^{-10}	5.7×10^{-10}
		Lung	9.7×10^{-4}	1.0×10^{-3}	8.1×10^{-4}	1.3×10^{-2}	1.3×10^{-10}	1.1×10^{-10}
		Thyroid	3.5×10^{-6}	3.8×10^{-6}	7.1×10^{-4}	1.4×10^{-2}	4.6×10^{-13}	9.2×10^{-11}

*Lifetime.

TABLE 9-59. Hazard Level Criteria.

Hazard level	Offsite			Onsite		
	Whole body	Bone surface thyroid	Lung other organs	Whole body	Bone surface thyroid	Lung other organs
Low	≤ 0.5	≤ 3	≤ 1.5	≤ 5	≤ 30	≤ 15
Moderate	> 0.5 to < 5	>3 to <30	> 1.5 to <15	> 5 to <25	>30 to <150	>15 to < 75
High	≥ 5	≥ 30	≥ 15	≥ 25	≥ 150	≥ 75

Based upon first-year dose risks to the maximum onsite and offsite individual listed in Table 9-58, PUREX presents an acceptable level of risk.

9.3.3 Consequence Guidelines

Reference 73 provides guidelines for evaluation of accident consequences to offsite personnel. The guidelines are shown in Table 2-11.

Based upon the lifetime dose commitment to organs of concern with respect to offsite personnel, all of the postulated accidents are well below the consequence guidelines.

9.4 SUMMARY

In summary, a PHA was developed for PUREX Plant operations. The hazards identified in the PHA were reviewed by personnel experienced in the operations of the PUREX Plant. Preliminary risk assessment was used to screen for primary risk contributors as well as to provide approximate frequency of occurrence of the accidents. The accidents were developed as scenarios with onsite and offsite doses to maximum exposed individuals and on and offsite population doses calculated.

Comparison of the predicted accident consequences with established criteria result in the judgement that PUREX Plant operations may be conducted with an acceptable level of risk.

9.5 REFERENCES

1. Letter to G. S. McElroy, "Preliminary Hazards Analysis for Operation of the Hanford PUREX Plant" Pacific Northwest Laboratory, Richland, Washington (November 17, 1981).

2. ANSI, Criteria for the Design of Plants for the Manufacture of Mixed Oxide (Uranium-Plutonium) Fuels, ANSI N 287 American National Standards Institute (December 30, 1976).

3. DOE, Safety Analysis and Review System, DOE Order 5481.1A, U.S. Dpeartment of Energy, Washington, D.C. (August 13, 1981).

4. PNL, Population Estimates for the Areas Within a 50-Mile Radius of Four Reference Points on the Hanford Site, PNL-4010, Pacific Northwest Laboratory, Richland, Washington (November 1981).

5. BNWL, DACRIN, A Computer Program for Calculating Organ Dose from Acute or Chronic Radionuclide Inhalation, BNWL-B-389, Rev. 1.2, Pacific Northwest Laboratory, Richland, Washington (1980).

6. PNL, PABLM - A Computer Program for Calculating Accumulated Radiation Doses from Radionuclides in the Environment, PNL-3209, Rev. 2.2, Pacific Northwest Laboratory, Richland, Washington (1980).

7. BNWL, SUBDOSA, A Computer Program for Calculating External Doses from Accidental Atmosphere Releases of Radionuclides, BNWL-8-351, Pacific Northwest Laboratory, Richland, Washington (1975).

8. JABE, Seismic and Tornado Evaluation of 202-A PUREX Buildings, RHO-CD-1560, URS/John A. Blume and Associates, Engineers, San Francisco, California for Rockwell Hanford Operations, Richland, Washington (November 1981).

9. JABE, PUREX Plant Structural Fragility Analysis, RHO-CD-1561, URS/John A. Blume and Associates, Engineers, San Francisco, California (November 1981).

10. JABE, PUREX Plant Equipment Fragility Analysis, RHO-CD-1526, URS/John A. Blume and Associates, Engineers, San Francisco, California (August 1981).

11. Rockwell, PUREX Plant Seismic Hazards Risk Analysis and Cost Benefit Studies Based on Offsite Dose Commitments, RHO-CD-1483, Rockwell Hanford Operations, Richland, Washington (November 1981).

12. Rockwell, PUREX Plant Seismic Hazards Onsite Risk Analysis, RHO-CD-1550, Rockwell Hanford Operations, Richland, Washington (December 1981).

13. Uniform Building Code, Pacific Coast Building Official's Conference (1952).

14. Rockwell, Tornado Induced Damages to the PUREX Plant and the Resulting Offsite Radiological Consequences, RHO-MA-1551, Rockwell Hanford Operations, Richland, Washington (October 1981).

15. NRC, Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Plutonium Processing Plant or Fuel Fabrication Plant, Regulatory Guide 3.35, U.S. Nuclear Regulatory Commission, Washington, D.C. (July 1979).

16. D. A. Marsh to W. Prosk, "Postulated N Cell Criticality Events and Resulting Evacuation Dose," Internal Letter, Rockwell Hanford Operations, Richland, Washington (March 19, 1982).

17. R. T. Stordeur, Chairman, "Receipt of Short Cooled Fuel from N-Reactor," Meeting Minutes with A. P. Larrick of United Nuclear Corporation (January 7, 1982).

9413219.1054

18. Resarch and Engineering, PUREX Technical Manual, RHO-MA-116, Rockwell Hanford Operations, Richland, Washington (March 1980).

19. R. W. Harvey, Radioiodine Release to Atmosphere - PUREX Plant - September 2, 1963, Unusual Incident Report, Rockwell Hanford Operations, Richland, Washington (September 1963).

20. Letter to R. T. Stordeur, "Source Terms for I-131 in a Green Fuel Accident," Internal Letter 65480-82-112, Rockwell Hanford Operations, Richland, Washington (May 11, 1982).

21. Bechtel, A Summary of Parameters Affecting the Release and Transport of Radioactive Material from an Unplanned Incident, Bechtel Company, Richland, Washington (September 1978).

22. Rockwell, PUREX Plant Gaseous Iodine-129 Control Capability and Process Development Requirements, RHO-CD-1400, Rockwell Hanford Operations, Richland, Washington (January 1981).

23. Rockwell, Safety Analysis Report for Packaging: Irradiated Fuel Cask Cars, HCS-028-001-00, Rockwell Hanford Operations, Richland, Washington (June 1981).

24. Redox Dissolver Incident Comprehensive Report, HW-66850-Del, Hanford Atomic Products Operation, Richland, Washington (September 13, 1960).

25. Pyrophoricity of Uranium in Reactor Environments, HW-62442, Hanford Atomic Products Operation, Richland, Washington (January 22, 1960).

26. Oxidation of Uranium in Air at High Temperatures, HW-58022, Hanford Atomic Products Operation, Richland, Washington (December 10, 1958).

27. ARHCO, Failure Mechanisms in Separations Plant of Significance to Radiological Safety, ARH-1008, Atlantic Richfield Hanford Operations, Richland, Washington (March 31, 1969).

28. Rockwell, Analysis of the Dissolver Silver Reactors from Hanford's PUREX Plant, RHO-ST-2, Rockwell Hanford Operations, Richland, Washington (June 1978).

29. Letter to M. L. Bell, "Safety Analysis Report (SAR) for the Disposal of PUREX Silver Reactors," Internal Letter 65411-81-061, Rockwell Hanford Operations, Richland, Washington (March 3, 1981).

30. SI, Probabilistic Risk Analysis of the Operations of the PUREX Plant, RHO-C-4, Science Applications, Inc. for Rockwell Hanford Operations, Richland, Washington (October 1977).

31. DOI, Flammability Characteristics of Combustible Gases and Vapors, Bureau of Mines Bulletin 627, U.S. Department of the Interior (1965 revised 1976).

32. ANL, Characterization of Particulate Plutonium Released in Fuel Cycle Operations, ANL-75-78, Argonne National Laboratory, p. 54 (May 1976).

33. Rockwell, PUREX Criticality Safety Analysis Report, CSAR-80-018, Rockwell Hanford Operations, Richland, Washington (1983).

34. ARHCO, Operating Experience Using Silver Reactors for Radioiodine Removal in the Hanford PUREX Plant, ARH-SA-67, Atlantic Richfield Hanford Operations, Richland, Washington (June 17, 1970).

35. GE, Treatment of Gaseous Effluents, HW-49549, General Electric Company, Richland, Washington (April 1957).

36. NRC, Assumptions Used for Evaluating the Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors, Regulatory Guide 1.3, U.S. Nuclear Regulatory Commission, Washington, D. C.

37. Rockwell, PUREX Preliminary Safety Analysis Report, RHO-CD-400, Rockwell Hanford Operations, Richland, Washington (June 30, 1978).

38. Rockwell, Plant Operating Procedure, "Emergency Response to a Canyon Fire," PO-001-050, Rev. C-0, Rockwell Hanford Operations, Richland, Washington (April 13, 1982).

39. NFPA, Fire Protection Handbook, National Fire Protection Association 15:4-33

40. BNWL, Interim Report: The Fractional Airborne Release of Dissolved Radioactive Materials During the Combustion of 30 Percent Normal Tributyl Phosphate in a Kerosine-Type Diluent, BNWL-B-274, Pacific Northwest Laboratory, Richland, Washington (June 1973).

41. Letter to E. C. Vogt, "PUREX Exhaust Filters Decontamination Factors (DF)," Internal Letter 65940-82-112, Rockwell Hanford Operations, Richland, Washington (March 25, 1982).

42. Letter to J. H. Roecker, "Project B-205," Internal Letter 65400-81-009, Rockwell Hanford Operations, Richland, Washington (February 2, 1981).

43. GE, Status Report of 202-A Building Areas Contaminated with Plutonium as a Result of L-6 Blowback, HW-43073, General Electric Company, Richland, Washington (May 15, 1956).

44. Rockwell, Conceptual Design Report, PUREX White Room Filtration System, Project B-379, RHO-CD-1254-CDR, Rockwell Hanford Operations, Richland, Washington (August 1981).

45. Rockwell, Radiation Work Procedures, "PUREX Pipe and Operating Gallery," RHO-MA-172-A, RWP No. A-10, Rockwell Hanford Operations, Richland, Washington (April 1, 1981).

46. Letter to G. J. Lebaron, "Assessed Reduction in Risk Attributable to Project B-205 - PUREX P&O Gallery Ventilation and Filter System," (October 27, 1980).

47. Rockwell, Standard Operating Procedure, "Pipe and Operating Gallery Floor Drain System," PO-600-080, Rockwell Hanford Operations, Richland, Washington (August 18, 1978).

48. ORNL, Oak Ridge National Laboratory Siting of Fuel Reprocessing Plants and Waste Management Facilities, ORNL-4451, Oak Ridge National Laboratory, Oak Ridge, Tennessee (1970).

49. ARHCO, Part I, 221-B Building Strontium 90 Release, March 22, 1970, ARH-1648, PTI Revised, Atlantic Richfield Hanford Operations, (April 23, 1970).

50. "Action Items for Proposed Modifications to the PUREX East Change Room Ventilation System," Letter CPSC-82-027 (March 30, 1982).

51. VITRO, Title 1 Design Report - PUREX Oxide Conversion Facility, VITRO-R-578, Vitro Engineering, Richland, Washington (November 1978).

52. Rockwell, PUREX Oxide Conversion Facility Preliminary Safety Analysis Report, RHO-CD-704, Rockwell Hanford Operations, Richland, Washington (July 1979).

53. M. W. Gibson, to R. E. Van der Cook, "An Investigation of an Alternate Solvent Fire in N Cell," DSI, Rockwell Hanford Operations, Richland, Washington, April 19, 1982).

54. Rockwell, Prediction of Releases of Airborne Particulates from PUREX Gloveboxes, Process Cells, and Canyon, RHO-CD-1113, Rockwell Hanford Operations, Richland, Washington (December 1981).

55. ARHCO, Final Report - Program for Fire Protection, Caves, Canyons and Hot Cells, ARH-ST-104, Atlantic Richfield Hanford Operations, Richland, Washington (August 1974).

56. LLNL, Fire Protection Countermeasures for Containment Ventilation Systems, UCID-18781, Lawrence Livermore National Laboratory (August 25, 1980).

57. ARHCO, Guide to Fire Protection in Caves, Canyons, and Hot Cells, ARH-3020, Atlantic Richfield Hanford Operations, Richland, Washington (July 1974).

58. PNL, Potential Releases from Soil Working Operations in a Contaminated Area, PNL-3498, Pacific Northwest Laboratory, Richland, Washington (August 1980).

9413299.1057

59. BNWL, Some Experimental Measurements of Airborne Uranium (Representing Plutonium) in Transportation Accidents, BNWL-1732, Pacific Northwest Laboratory, Richland, Washington (August 1973).

60. R. E. Van der Cook to J. B. Fecht, "Estimated Consequences of Simultaneous Waste Liner Rupture and U Plant Exhaust System Failure" Internal Letter, Atlantic Richfield Hanford Operations, Richland, Washington (March 26, 1974).

61. Rockwell, Containment and Monitoring of Transuranics from PUREX Tank Coil Failures, RHO-CD-1234, Rockwell Hanford Operations, Richland, Washington (January 1981).

62. Rockwell, Analysis and Assessment of 200 Areas Liquid Effluent Sampling and Monitoring Systems, RHO-CD-1503, Rockwell Hanford Operations, Richland, Washington (October 27, 1981).

63. Rockwell, Standard Operating Procedure, "216-A-42 Diverted Solution Disposal," PO-500-120, Rockwell Hanford Operations, Richland, Washington (February 23, 1982).

64. Radiation Occurrence, "PUREX Cooling Water Swamp - Leak from TK-F15 Cooling Coil" (June 12, 1964).

65. ARHCO, Hazards Review - Waste Management Effluent Control - PUREX Cooling Water and Steam Condensate Diversion (Project B-112a), ARH-CD-471, Atlantic Richfield Hanford Operations, Richland, Washington (October 14, 1975).

66. Rockwell, Standard Operating Procedure, "TK-D5 Isolation and Restoration" PO-160-224, Rockwell Hanford Operations, Richland, Washington (April 2, 1982).

67. ARHCO, PUREX Flowsheet Processing N-Reactor Fuels, ARH-F-103, Atlantic Richfield Hanford Operations, Richland, Washington (November 24, 1976).

68. Rockwell, Standard Operating Procedure, "Railroad Tunnel - Car Puller Operation," PO-530-020, Rockwell Hanford Operations, Richland, Washington (March 31, 1981).

69. Rockwell, Standard Operating Procedure, "Dispatcher Operation," PO-020-040, Rockwell Hanford Operations, Richland, Washington (June 4, 1981).

70. Rockwell, Radiation Work Procedures, "PUREX Railroad Facilities," RHO-MA-172, RWP No. A-6, Rockwell Hanford Operations, Richland, Washington (April 1, 1981).

71. Rockwell, Radiological Controls, RHO-MA-220, Rockwell Hanford Operations, Richland, Washington (November 1, 1978).

72. Letter to S. M. O'Toole, "Dose Rate Calculation of Mark-II Canister and Type IV Annular Dissolver," Internal Letter 72320-82-067, Rockwell Hanford Operations, Richland, Washington (February 24, 1982).

73. G. M. Christensen and R. Jacobs, Safety Analysis and Report Preparation Guide, RHO-HS-MA-1, REV 1, Rockwell Hanford Operations, Richland, Washington (February 1984).

THIS PAGE INTENTIONALLY
LEFT BLANK

PUREX FSAR

CHAPTER 10.0 CONDUCT OF OPERATIONS

THIS PAGE INTENTIONALLY
LEFT BLANK

This page intentionally left blank.

CONTENTS

10.0 Conduct of Operations 10-1

10.1 Organizational Structure and Management Control System . . . 10-1
10.1.1 Corporate Organization 10-1
10.1.2 Rockwell Hanford Operations 10-1
10.1.3 Rockwell Programs and Activities 10-3
10.1.4 Program Management 10-3
10.1.5 Functional Management 10-6
10.1.6 Qualifications of Management Personnel 10-33
10.1.7 Staffing and Manpower Requirements 10-33

10.2 Training Programs 10-33
10.2.1 General Training Program Description 10-37
10.2.2 Administration and Records 10-40
10.2.3 Adequacy of Testing 10-40

10.3 Normal Operations 10-41
10.3.1 Plant Procedures 10-41
10.3.2 Records. 10-44

10.4 Emergency Plans. 10-44
10.4.1 Fire Protection. 10-45
10.4.2 Emergency Shutdown of Process Systems. 10-46

10.5 Decommissioning 10-47

10.6 References 10-51

FIGURES:
10-1. Matrix Management Relationship 10-4
10-2. Plant Operations 10-8
10-3. Production Processing Function 10-9
10-4. Production Support Function 10-10
10-5. PUREX Maintenance Organization 10-11
10-6. PUREX Operations Organization 10-13
10-7. PUREX Operating Organization 10-16
10-8. Analytical Laboratories 10-26
10-9. Research and Engineering Function 10-28
10-10. Process Engineering Department 10-29
10-11. Process Design Department 10-30
10-12. Design Engineering Department 10-31
10-13. Development Engineering Organization 10-32

9413219.1062

TABLES:

10-1. Quality Assurance Verification System 10-22
10-2. Qualifications of Management Personnel Associated with Other Support Organizations 10-34
10-3. Qualifications of Key PUREX Operating and Resident Support Personnel 10-35
10-4. Manpower Requirements for the PUREX Plant 10-36

9413219.1063

10.0 CONDUCT OF OPERATIONS

This chapter describes the Rockwell Hanford Operations (Rockwell) organizational structure and management control system as it applies to the Plutonium-Uranium Reduction Extraction (PUREX) program. Also discussed are the provisions for preoperational testing, training, normal operation, and emergency planning to ensure safe operation of the plant.

10.1 ORGANIZATIONAL STRUCTURE AND MANAGEMENT CONTROL SYSTEM

The organization described herein was established by Rockwell International Corporation to conduct operations at the Hanford Site under contract to the U.S. Department of Energy (DOE).

Salient features of this organization are discussed, including the matrix management system and the responsibilities, authorities, and activities of the program and functional components.

10.1.1 Corporate Organization

Rockwell is a subsidiary of North American Space Operations Division, Rockwell International Corporation.

10.1.2 Rockwell Hanford Operations

Rockwell conducts operations at the DOE Hanford Site facilities near Richland, Washington in fulfillment of contract DE-AC06-77RL01030. This contract, administered by the Richland Operations Office of DOE (DOE-RL), specifies that Rockwell will provide materials and services for certain identified operations at the Hanford Site. Rockwell International corporate personnel are available for technical assistance when it is requested.

The top officer for Rockwell is the Vice President and General Manager. This resident officer reports to the President, North American Space Operations.

The Chief Counsel, Rockwell, is an assistant general counsel of the Corporation assigned to Rockwell. He reports administratively to the Vice President and General Manager, Rockwell, and professionally to the Staff Vice President and Associate General Counsel, corporate headquarters, El Segundo, California.

The Rockwell organization was established to plan, manage, and control effectively the various programs and activities assigned to Rockwell by DOE. The Program Directors and Functional Directors report to the Vice President the General Manager. The Rockwell Policy Manual describes the Rockwell organization.[1]

10.1.2.1 Organizational Levels. Rockwell organization levels are identified by their reporting relationship to the Vice President and General Manager's office as follows.

- Program Office

  An organization reporting directly to the Vice President and General Manager's office, assigned responsibilities for the overall planning and conduct of specific programs.

- Function

  An organization reporting to the Vice President and General Manager's office, assigned specific functional responsibilities for performing the technical tasks in support of various programs. The exceptions are Production Support and Production Processing Functions, which report to the Director, Plant Operations and functional organizations whose directors report through the Director, Site Services.

- Department

  An organization reporting directly to a function.

- Group

  An organization reporting directly to a department.

- Unit

  An organization reporting directly to a group.

10.1.2.2 General Manager's Staff. The Vice President and General Manager appoints a staff to formulate policy and conduct day-to-day business activities. The Vice President and General Manager's staff is comprised of the following:

- Assistant General Manager
- Program Director, Chemical Processing
- Program Director, Waste Management
- Program Director, Landlord
- Director, Basalt Waste Isolation Project (BWIP)
- Director, Personnel and Public Relations
- Director, Safety and Quality Assurance (S&QA)

- Director, Research and Engineering (R&E)
- Director, Plant Operations
- Director, Site Services
- Chief Counsel
- Chief Auditor.

10.1.2.3 Executive Control Meeting. An essential part of the Rockwell management control system is the weekly executive control meeting (ECM). On a rotating basis, the program offices and functions present briefings on the status of their programs and operations to Rockwell management for internal visibility, evaluation, and control. Emphasis is placed on program assessment in areas of performance, schedule, technical, financial, and customer interfaces. The ECM provides management with information to resolve problems or redirect resources in problem areas.

10.1.2.4 Rockwell Policies Manual. The Rockwell policies manual(1) serves as the formal channel for communication of policies and procedures throughout the total Rockwell organization. These policies implement company, contractual, legal, and governmental requirements to achieve coordinated and efficient operations.

Rockwell management is provided with the procedures required to effectively plan, organize, direct, and control operational activities in a formal management control system.(2)

10.1.3 Rockwell Programs and Activities

Rockwell activities at the Hanford Site include four major program areas: Chemical Processing, Waste Management, Landlord, and the BWIP. Included in the Chemical Processing Program is the management and operation of the PUREX Plant.

A requirement of Rockwell's contract is to prepare and publish for DOE a program plan for the startup, operation, and maintenance of the PUREX and to update the plan at appropriate intervals. A program plan for the PUREX Plant responds to that requirement and is periodically revised. Approval of the plan by DOE is required.

10.1.4 Program Management

Effectively managing programs requires planning, integrating, organizing, staffing, directing, and controlling the combined efforts of program and functional organizations to achieve program objectives. Position descriptions defining the responsibilities and authorities for the Program and Functional Directors are contained in the Rockwell policies manual;(1) a summary follows.

9413219.1066

10.1.4.1 Matrix Management. Each program is managed by utilizing the integrated matrix management system (Fig. 10-1). This involves assignment of staff from various functional disciplines to meet the needs of particular programs.

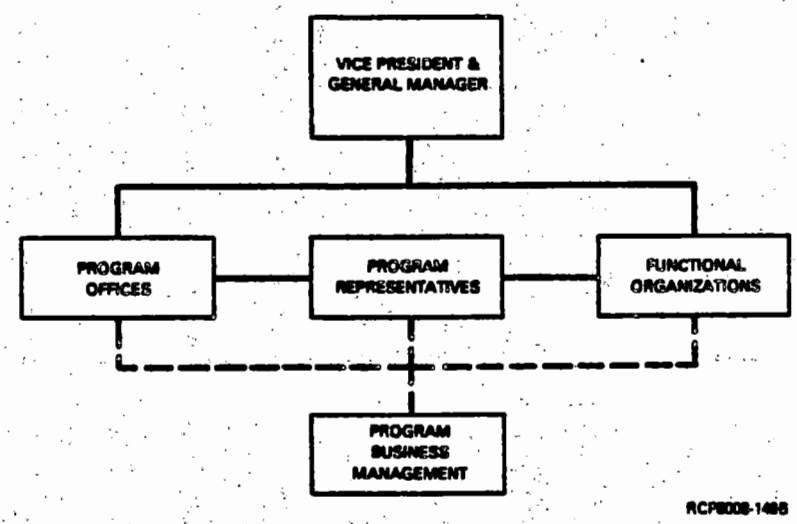


Figure 10-1. Matrix Management Relationship.

The program offices establish program requirements and objectives by adapting program plans, work packages, budgets, and program schedules. In addition, program offices provide performance evaluations and reports and specify corrective action when appropriate. The actual work on any program is accomplished by the functional organizations, which are responsible for detailed scheduling, provision of manpower, and performance of identified tasks.

The matrix management system assures that the program requirements are accomplished. The program management team consists of representatives from each of the functions who serve as the interface between the program office and the functional organizations.

Program Business Management (PBM) provides support to both functional and program organizations.

10.1.4.2 Program Director. A Program Director is appointed by and is responsible to the Vice President and General Manager and is delegated authority to plan, coordinate, and control the performance of the program. One Program Director is designated for each program area. The Program Director acts as the primary point of contact with DOE on program activities and is responsible for receiving and approving all correspondence pertaining

to the program activities. Responsibility for the management of the PUREX Plant is assigned to the Associate Program Director, PUREX, within the Chemical Processing Program as indicated in the Rockwell Policy Manual.(1)

10.1.4.3 PUREX Program Team. The organization established for the PUREX Program is headed by the Associate Program Director, PUREX, who is accountable to the Program Director for the technical objectives, schedule requirements, and cost management of PUREX.(1)

The Associate Program Director is appointed by the Program Director with the concurrence of the Vice President and General Manager and acts as the primary point of contact with DOE-RL or other contractors on program reporting and communication.

Program Managers and Project Managers are appointed for portions of the program, with the following responsibilities:

- Prepare functional planning documents (FPD) to assist in developing work packages that define work to be accomplished, establishing schedules, and allocating appropriate resources.

- Obtain functional organization commitments to accomplish program objectives as defined in the FPD and assure that appropriate functional skills, resources, and capabilities are applied to the programs.

The Construction Projects Manager for the Chemical Processing Program Office represents Rockwell concerning construction activity at the PUREX Plant performed by subcontractors. The full responsibilities for construction activities and the procedures for performing, monitoring, and accepting work are described in detail by formal written procedure.

10.1.4.4 Program Team Responsibilities. A program charter developed for the PUREX Plant startup and operation designates the members of the Program Team and defines their responsibilities as follows:

- Represent Rockwell in all program matters with the customer

- Prepare and maintain the program plan that defines the activities; subactivities; technical, scheduling, and financial requirements; and constraints of the PUREX program

- Issue work packages and budget targets of all functional organizations and other participants in the program

- Evaluate performance against the program plan and direct alternative actions as required

- Present status of program to Rockwell management and others as required.

10.1.4.5 Programmatic Work Breakdown Structure, Work Packages, and Functional Planning Document. The work breakdown structure in the program plan breaks down the program into elements of work that can be scheduled, estimated, and tracked. The schedules are made to the work breakdown structure (WBS) so that elements of work can be authorized in an orderly and logical pattern.

A work package is a subdivision of the WBS and is consistent with the WBS as contained in the approved program plan.

The work package is of manageable size from the standpoint of schedule, cost, and technical objectives and control, and it is an element for which the collection and reporting of complete dollar costs and variances are warranted. Work packages define the work to be accomplished, establish milestone schedules, and identify the resources to be utilized. Work packages assign specific management responsibilities and are an integral part of the management control system.

The performing functional organization prepares an FPD, which at the lowest level of the WBS provides time phasing and reestimating of the functional schedules. These are submitted to the program offices and, when different from the target budget or the work package, are negotiated with the functions. Once this negotiation is completed, the work package is signed by the Functional and Program Directors, or their delegates, and becomes the fiscal year budget, subject to periodic amendment and semiannual revision.

10.1.5 Functional Management

Functional Directors are responsible for the accomplishment and technical excellence of all assigned work and the effective cost management of assigned tasks as reflected in the program plans and work packages. Functional Directors are responsible for providing necessary staff, facilities, and services in support of program requirements. If resource allocation conflicts occur, the Functional Directors are obligated to negotiate changes with the Program Manager. They provide for the development and effective utilization of management and technical talent and skills to assure competence and capability.

In a matrix management organization, functional organizations are responsible for performing the services and providing the material and other resources required to meet the objectives of the program. Functional organizations are staffed for maximum utilization of specialists who can be reassigned as required.

Functional manuals are issued under the authority of Functional Directors for the purpose of administering operations within the director's organization and for implementing management policies. The Rockwell policies manual lists the manuals maintained by Rockwell.(1)

10.1.5.1 Plant Operations. The Plant Operations organization is responsible for the site-wide operations of Rockwell. The organizational structure is shown in Figure 10-2. The organization consists of two Functions, Production Processing and Production Support, and three department-level organizations, which all report to the Director, Plant Operations. An Associate Director of Plant Operations is provided to assure management overview of this rather large organization.

10.1.5.1.1 Production Processing Function. The operation of the PUREX Plant is the responsibility of the Production Processing Function. The organizational structure of the Production Processing Function is shown in Figure 10-3. The significant responsibilities that are assigned to Production Processing within the PUREX program are implemented by the PUREX Operations Department (discussed in Section 10.1.5.2).

10.1.5.1.2 Production Support Function. The Production Support functional organization is shown in Figure 10-4. The PUREX Maintenance organization (Fig. 10-5) reports to the Plant and Facilities Maintenance, 200 Areas Department, which is integrated with the other Production Support departments.

The PUREX Maintenance organization is responsible for maintaining the PUREX Facility and equipment in a safe operating condition. This includes providing general maintenance services, calibration of instruments, checkout of control equipment, preventive maintenance, installation of process related wiring and piping, maintenance of all process equipment, and support services required for installation and removal of equipment.

An instrument technician, a pipe fitter, and an electrician are assigned to each of the four shifts. These craftsmen are allocated 8 h of work by their respective managers. On shift, they report to the operating shift manager who identifies assignments and establishes work priorities. The shift craftsmen respond to situations to keep the plant operating in a safe manner. Day-shift craftsmen do the preventive maintenance overhaul of equipment and all major repairs.

The Maintenance Engineering Department develops and issues preventive maintenance procedures and computerized maintenance schedules.

The Utilities and Services Department provides water and steam to the PUREX Plant and maintains the transport lines. This organization also balances the ventilation system and performs the integrity testing of the PUREX Plant high-efficiency particulate air (HEPA) filters.

10.1.5.1.3 Plant Operations Control. The prime objective of this organization is to provide a central coordinating, monitoring, controlling, and auditing role for the operating departments within the Plant Operations organization. A list of principal responsibilities follows.

- Provide planning for all Plant Operations Processing maintenance and fabrication activities and ensure that this planning includes and is consistent with other support organization requirements e.g., S&QA.

9413219.1070

9413219.1071

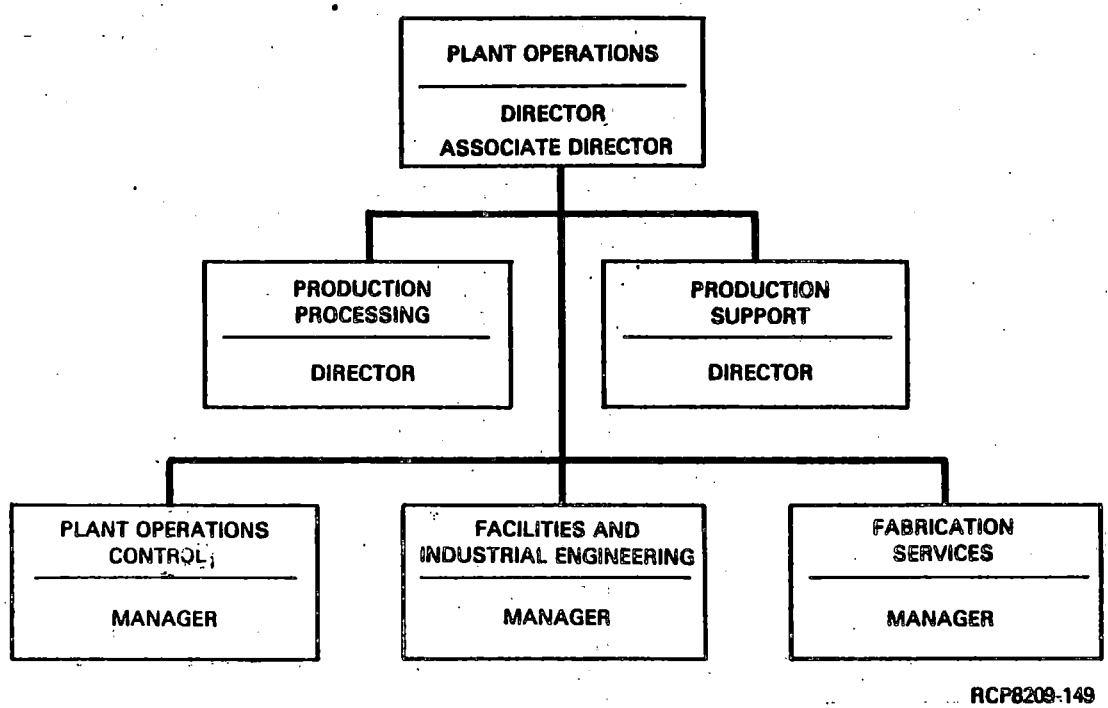


FIGURE 10-2. Plant Operations.

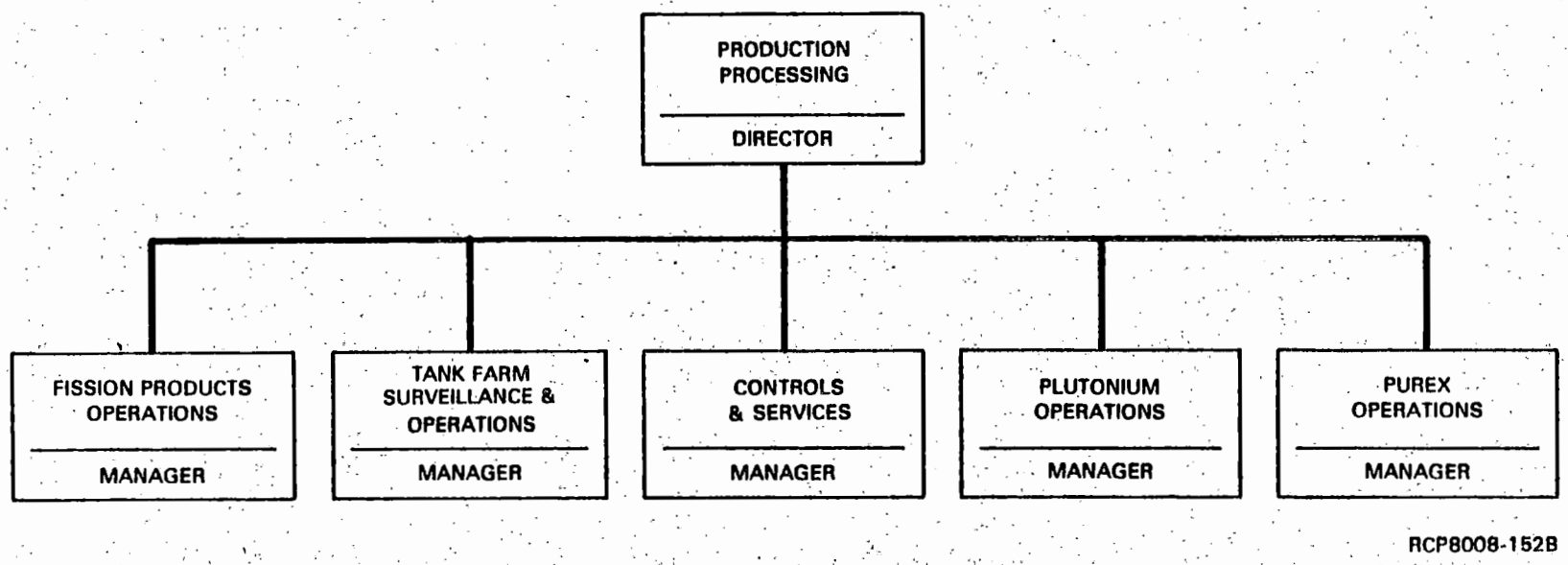


FIGURE 10-3. Production Processing Function.

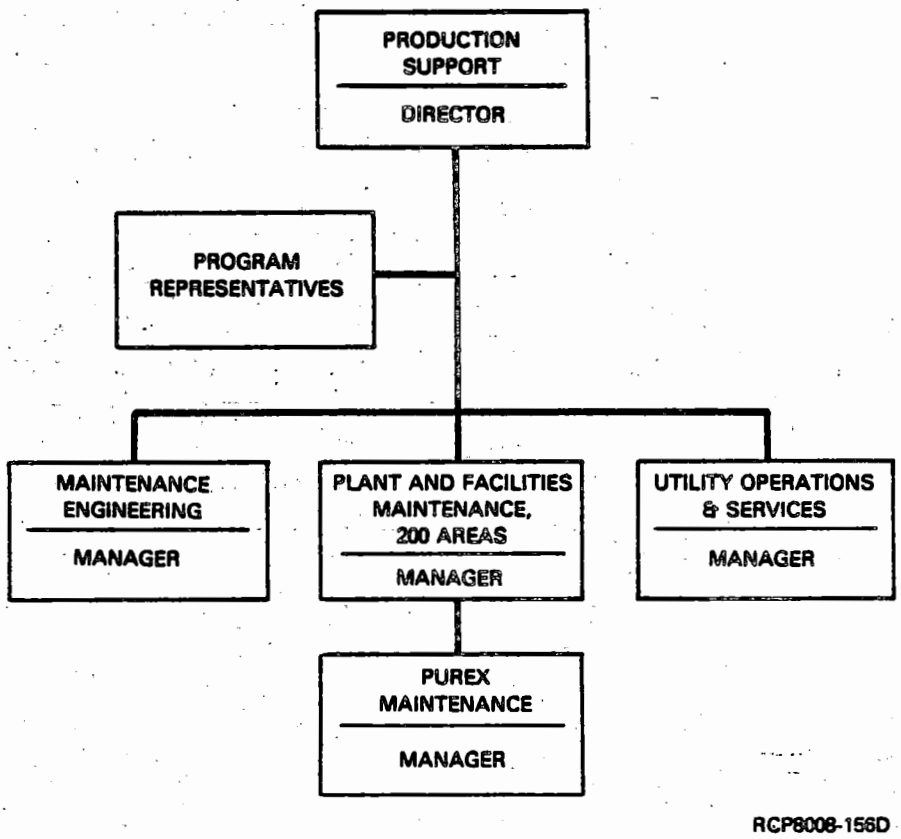


FIGURE 10-4. Production Support Function.

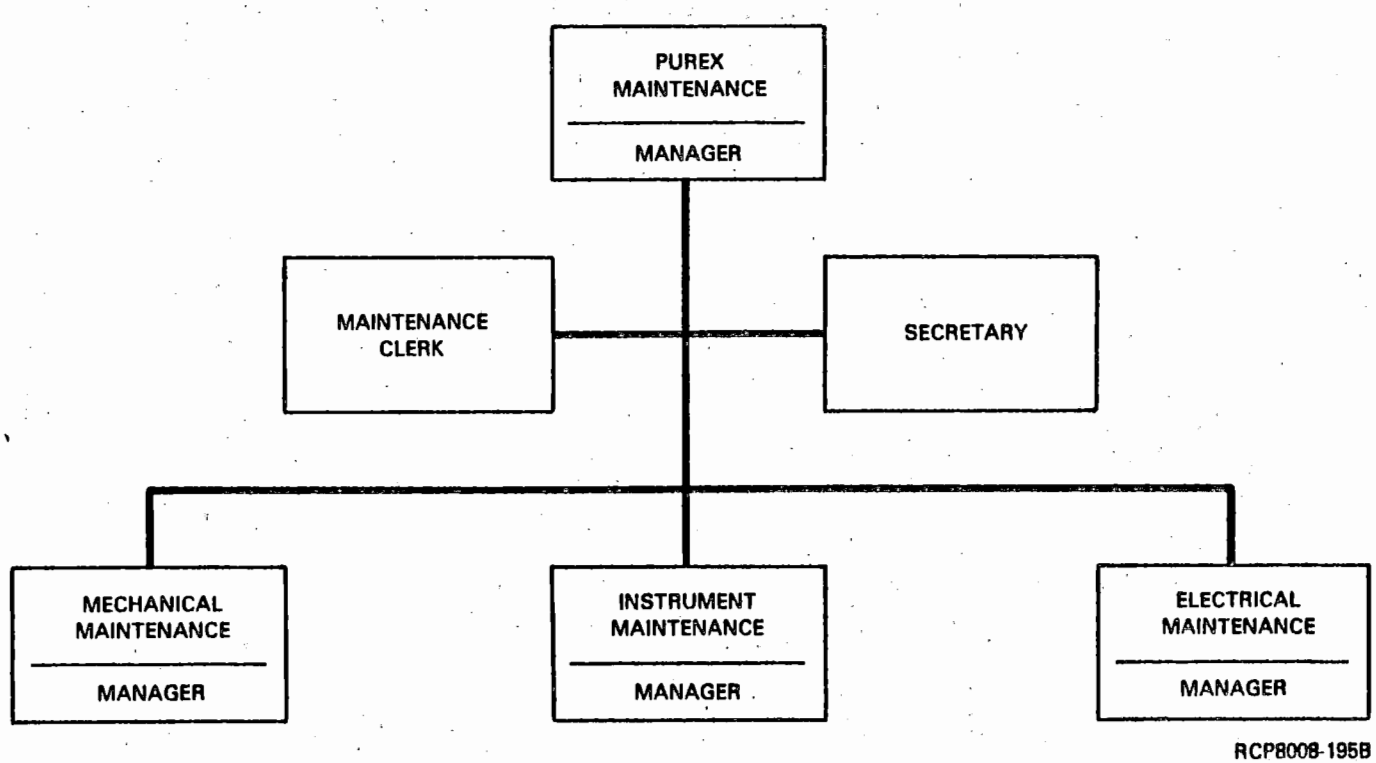


FIGURE 10-5. PUREX Maintenance Organization.

- Provide management information, systems, and procedures within Plant Operations.

- Maintain and track performance against logic networks and schedules that, at a minimum, consider function, manpower, equipment, material and document availability.

- Direct implementation and use of the Plant Operations work order system and supervise operation of the computerized work order authorization system.

- Develop work-around and/or catch-back plans as required to achieve Plant Operations' overall objectives.

- Provide inventory, management, and control for spare parts and equipment used by Plant Operations.

- Provide cost estimating capabilities and expertise for "make or buy" analysis and proposals.

- Provide liaison with other functions, other Hanford contractors, or outside subcontractors for work that cannot be performed within Plant Operations.

10.1.5.1.4 Facilities and Industrial Engineering. The Facilities and Industrial Engineering organization provides the following: (1) field construction coordination for new or modified Rockwell-managed facilities, (2) master planning and management for the assignment and utilization of general purpose facilities as landlord and controller of space, (3) industrial engineering studies, and (4) management overview of nonconstruction capital equipment.

10.1.5.1.5 Fabrication Services. Fabrication Services provides shops and specialists to fabricate equipment and material required by all Rockwell components and by other Hanford contractors, as required. The facilities located in the 200 East and 200 West Areas have a variety of equipment and knowledgeable personnel that results in an unusually wide range of fabrication capabilities.

10.1.5.2 PUREX Plant Management.

10.1.5.2.1 PUREX Operations. Responsibility for the operation of the PUREX Plant is assigned to the Manager, PUREX Operations Department. The PUREX Operations organization is shown in Figure 10-6. The significant responsibilities assigned to the PUREX Operations Department follow.

- Perform process activities required to meet program goals and objectives, including N Reactor fuel processing, PWR Core 2 fuel processing, and conversion of plutonium nitrate solutions to oxide.

- Operate the PUREX Plant in a safe manner.

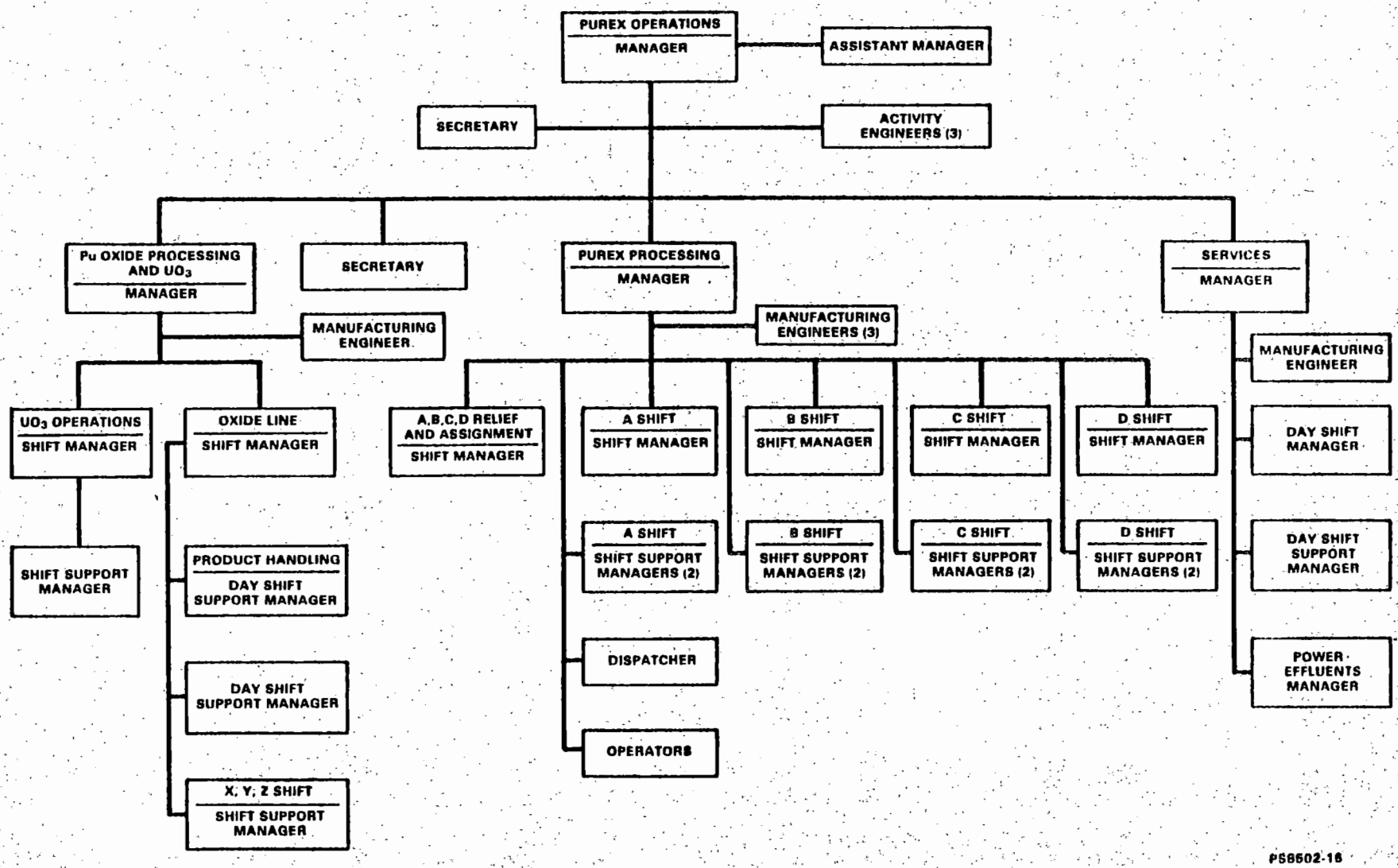


FIGURE 10-6. PUREX Operations Organization.

- Coordinate facilities maintenance.
- Plan, schedule, execute, and monitor Production Processing activities.
- Coordinate material and equipment required to support production activities.
- Maintain records for controlling shipping containers.
- Initiate, coordinate, and participate in readiness reviews.
- Provide building service to tenants not directly associated with Production Operations.
- Coordinate access for construction activities.

The Manager, PUREX Operations, is also the PUREX Building Superintendent; this position has the following responsibilities.

- Establish, approve, and disseminate specific building rules covering these areas:
  - Safety conditions and practices for the building, including periodic safety inspections
  - Building evacuation and emergency procedures
  - Contaminated waste generation and handling
  - Special nuclear materials and building security
  - Other contractor housekeeping and safety requirements.
- Establish and provide for the enforcement of good housekeeping practices in the building and on the building grounds.
- Prepare and issue plans for the general maintenance of the building appurtenances, ventilation, air filtration, and other building systems and request correction of maintenance deficiencies, via work orders, to Production Support or Maintenance and Transportation.
- Recommend changes in space allocations and equipment installations to Industrial Engineering.
- Implement security regulations and controls as directed by Safeguards and Security.
- Sample/monitor effluent releases by building tenants to assure compliance with applicable standards.

9413219.1077

The PUREX Processing Group has primary responsibility for day-to-day operation of the process system, including all aspects of radioactive fuel receipt and processing waste disposal in conformance with established written procedures. Total facility and process responsibility is assigned for each shift to a shift manager, who is assisted by two shift-support managers in directing the work force.

The PUREX Oxide Processing and Uranium Oxide (UO_3) Operations Group is responsible for the plutonium nitrate sampling area (product removal room), and the plutonium conversion facility. Activities include: an around-the-clock operation, 5 d/wk (X, Y, Z shift) in the plutonium conversion from a liquid nitrate into a powder oxide; the canning of the plutonium oxide as well as the shipment of the oxide; and the rework of any product that is not within the prescribed specifications. It is also responsible for conversion of uranium nitrate to uranium oxide in the UO_3 facility.

The PUREX Services Group is responsible for all utilities systems within the facilities and various ancillary operational functions required in support of process, maintenance, and project activities. This includes direct responsibility for building ventilation, water purification and distribution systems, steam distribution, and process and instrument air supply systems. All effluent waste sampling, chemical receipt and storage, and project and maintenance support work unique to the day shift are included in the scope of this function.

Manufacturing Engineering in support of PUREX operations is provided by engineers who address the current processing and long-range facility and process improvements and additions from an operational standpoint. Their responsibilities include design review, facilities and equipment analyses for replacement and improvement, and planning and operational coordination of project and major maintenance activities.

10.1.5.2.2 PUREX Operations Functional Support. The PUREX organization receives technical or administrative support from all Rockwell functions. This operating organization is shown in Figure 10-7. The net result is an integrated work force that is composed of the necessary professional and craft skills required to operate the PUREX Plant. The operational components are under the administrative and operational control of the PUREX Operations Department Manager. The direct functional support components are under the administrative control of their respective functional management but under the operational control of the PUREX Operations Department Manager. The QA laboratories, general building, operations, construction and maintenance activities, etc., all function together as an integrated work force. The Manager, PUREX Operations, has direct control of the operational component of the organization and has the authority and responsibility for coordinating the represented functional groups. All functions that provide essential daily input needed for effective operation of PUREX are occupants in the plant. The organization configuration and number of personnel provided in these groups are

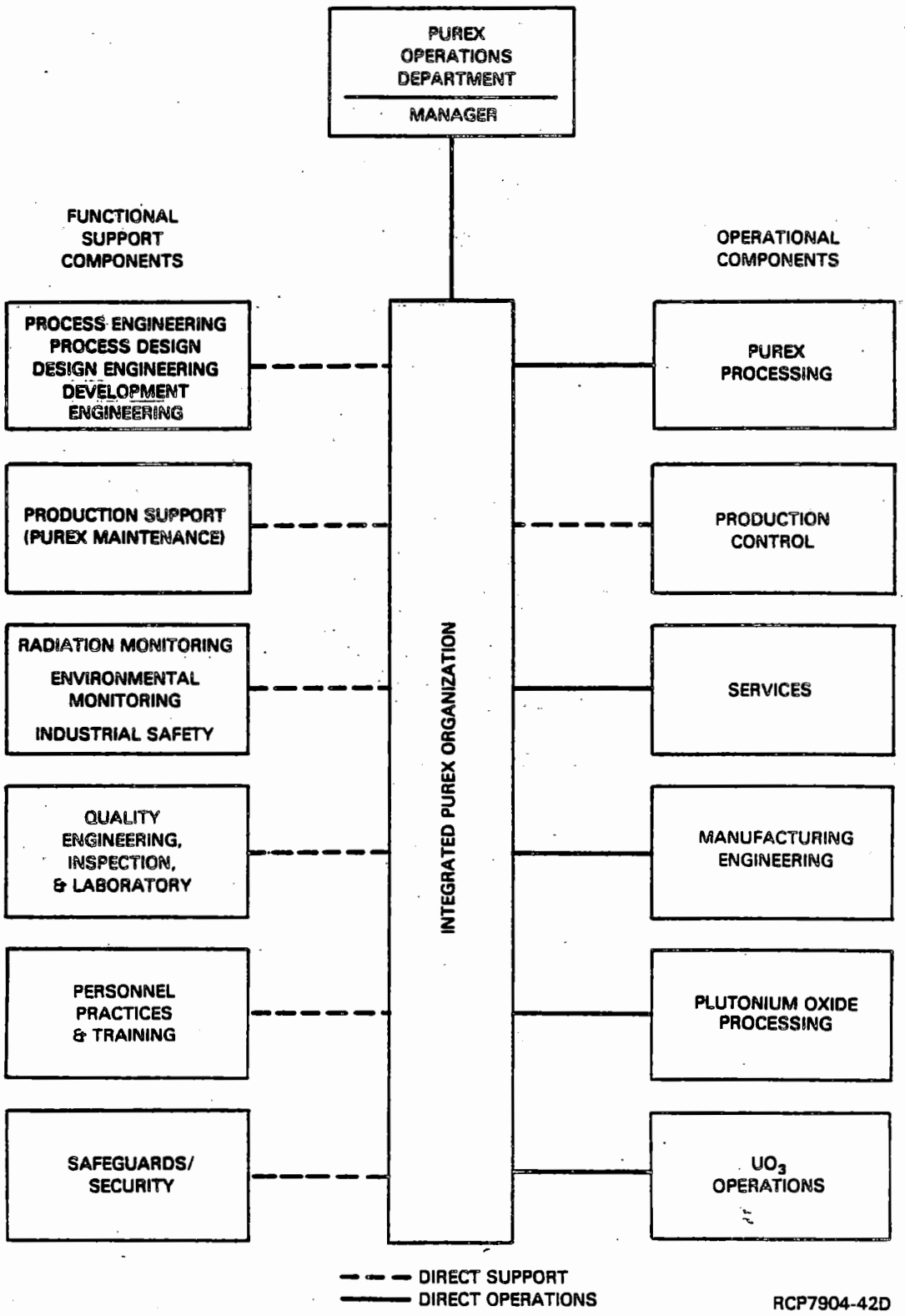


FIGURE 10-7. PUREX Operating Organization.

determined by the Functional Director, with the concurrence of the Manager, PUREX Operations. The result is an integrated organization in which the expertise required for the daily decisions involved in plant operation is immediately available. These decisions include personnel and environmental safety matters, process control problem resolution, quality conformance of maintenance and design changes, schedule modifications, and changes in fabrication or purchasing priorities. The Manager, PUREX Operations is accountable for the effective integration of all functional disciplines required to assure safety, quality, schedule effectiveness, and economy of plant operations.

During the offshifts (i.e., weekends, evenings, and nights) the responsibility for overall functional integration is delegated to the senior shift manager. All functional support services, whether resident or on call, respond to the needs of the shift manager relative to safety, process support, quality, and schedule effectiveness during that particular shift of operations.

10.1.5.3 Safety and Quality Assurance. This function includes various aspects of safety, environmental protection, radiological protection, and quality assurance and control. A discussion of the various activities assigned to this function follows.

- Provide radiation monitoring, approve radiation work permits and special work permits, and approve Production Operations' procedures to assure that personnel radiation exposures are maintained within safe levels.

- Provide radiation-shielding calculations and decontamination procedures; recommend tools to be used.

- Conduct design reviews to assure that radiation safety, industrial safety, fire protection, industrial hygiene, and environmental protection have been adequately considered in the design of facilities and equipment. Provide technical support to assure that environmental discharges from operation of facilities are reduced as low as is reasonably achievable (ALARA) in compliance with applicable county, state, and federal directives.

- Perform operational safety reviews. Provide criticality safety calculations and prepare criticality prevention specifications. Prepare safety assessment documentation and/or safety analysis reports (SAR) as required by Rockwell policies.

- Assure quality of design and operation.

Rockwell has developed and implemented a comprehensive program responding to the requirements of Reference 2. In keeping with the Rockwell policy of conducting its activities in a manner to protect employees, the public, the environment, and property, specific safety responsibilities are assigned to members of management organizations and employees. The S&QA Function supports this Rockwell policy through the following activities.

- Radiation Protection

  Rockwell's radiation protection program and methods for assuring that personnel radiation exposures are ALARA is described in Chapter 8.0.

- Nuclear Criticality Safety

  The Rockwell program for criticality safety ensures that fissile materials are processed, stored, transferred, and handled on the site and within facilities in such a manner that the probability of a criticality accident is remote. Clear lines of responsibility for nuclear safety, management review, and audit systems are established and maintained, and training programs for employee indoctrination and instruction are developed. The criticality prevention program and responsibilities for its implementation are documented by formal written policy. The Criticality Engineering and Analysis Group in the Safety Integration Department provides the technical support for the Rockwell nuclear safety program.

- Environmental Analysis and Monitoring Program

  This program is operated by the Environmental Protection Department that measures radionuclide release to the environs, maintain surveillance of operating facilities and storage sites to permit prompt detection and correction of unplanned releases, determine the impact of operations on the environs, and develop programs to reduce discharges to the environment. The frequency of inspection, corrective actions taken, compliance reports, unplanned release reports, surveillance standards, and other specific means by which the program is implemented are documented in a formal policy manual.

- Industrial Safety Program

  Although the basic responsibility for safety rests with line management having immediate authority over workers, a separate S&QA department (Industrial Hygiene and Safety) has been established to assist in the conduct of safe operations. This group reviews and approves designs, specifications, and procedures

with respect to compliance with industrial safety and hygiene standards. The program is monitored by the Industrial Hygiene and Safety Department through surveys and routine facility inspections. The Quality Assurance departments assist by monitoring the implementation of mandatory standards.

- Environmental Safety and Health Codes, Standards, and Guides

It is Rockwell's intent that all operations be performed in compliance with applicable DOE environmental protection safety and health protection requirements and that they adhere to generally recognized and accepted high standards of performance in the areas of occupational health; nuclear, radiological, industrial, and fire safety; and environmental protection. To this end, Rockwell's program describes compliance to many government and industry codes, standards, and guides. In-house criteria are also extensively developed and applied.

- Review of New Design

As specified by References 2 and 3, Rockwell requires that third-party engineering design reviews be conducted. These reviews assure that applicable requirements are correctly incorporated into engineering documents. All engineering documentation is subject to this policy. The Director, R&E establishes design review methodology and hold points for specific documentation and approves documents prior to release and implementation.

- Independent Safety Reviews

Rockwell policy establishes an independent safety review program, consisting of four safety review committees: Chemical Processing, Nuclear Criticality, Packaging and Shipping, and Waste Management Committees. These committees assure that all operations and functions are performed in accordance with established safety guidelines. Reviews of specific areas are performed by standing committees chaired by a senior staff person with acknowledged expertise in important, related disciplines. Approval authority is vested in senior management.

The overall functioning of the safety review committees is the responsibility of the Safety Review Panel, which consists of the Director, S&QA, the individual chairman of the four safety review committees, an executive secretary, and an administrative secretary.

- Safety Documentation

It is Rockwell's policy to provide safety analyses and reviews that meet the policy, objectives, and requirements of DOE Order 5481.1A.[4] The Safety Analysis Group of S&QA prepares

documentation that reports on the capability of a facility to operate without undue risk to the public, work force, or environment. The preparation of SAR is Rockwell's formal system for identifying and assessing risks.

- Compliance with Approved Procedures

Rockwell personnel involved in operations that have health, environmental, or safety implications are guided by written procedures. These procedures are approved by designated personnel involved with safety of the operation for which the procedure is prepared. In turn, the procedures are based on Rockwell-approved standards and specifications.

- Unusual Occurrences

The Rockwell program provides a policy for the notification, investigation, and reporting of unusual occurrences. This policy requires immediate investigation to identify causes and corrective actions to prevent similar occurrences. Analyses and documentation of the occurrences are required so that informed decisions can be made to improve operations.

Rockwell policy also provides that in the event of an emergency situation, local response teams under the leadership of an Emergency Director will act to bring the situation under control. Emergency procedures for operations and support buildings are formulated. General guidelines and assignment of duties during plant-wide emergency response are compiled in a widely distributed manual.(5) Emergency tests and drills are conducted as part of a continuing training program. An Emergency Planning Review Committee is established to coordinate and evaluate these tests and drills to review emergency plans, procedures, equipment, and facilities, and to prepare and update the Rockwell emergency plan.

- Continuing Program Overview

The Rockwell program establishes that DOE environmental protection, safety, and health protection objectives are a prime responsibility of Rockwell management. Therefore, an Executive Safety Committee with responsibility for creating safety policy, approving goals and objectives, and evaluating safety performances was established. This safety committee consists of the Vice President and General Manager, the Program Directors and the directors of the following functions: R&E, S&QA, Plant Operations, Production Processing, Production Support, Personnel and Public Relations, Safeguards and Security, and Maintenance and Transportation.

Rockwell policy provides for an Accident Prevention Council whose function is to encourage employee review of safety matters and to serve as a communications link between employees and management. The Rockwell internal audit program provides an independent audit to assess the extent of compliance with DOE and Rockwell policy, as well as to measure functional performance against established guidelines.

Special emphasis is given to activities involving fissile materials. Authority is given to S&QA to suspend any operation considered to constitute an immediate and significant danger to safety, health, property, or the environment. A nonconformance reporting and control system was established to document and correct conditions not in compliance with specified requirements.

- Quality Assurance

The requirements applied by the Quality Assurance activities to promote the safety of design and operation of the PUREX Plant and the responsibilities with respect to quality assurance policy are in accordance with References 1, 3, and 6. Table 10-1 presents a generic summary of the quality assurance verification system.

Quality assurance program plans (QAPP) are proposed and issued under the authority of the Director, S&QA. The QAPP identified the planned actions necessary to provide adequate confidence that facility and administrative controls are in place for designated programs and special support activities.

- Level of Quality

Rockwell quality and safety objectives are included in program plans to comply with DOE orders. Each Rockwell function identifies its organizational responsibilities to the program objectives using functional manuals, plan, procedures, instructions, and reports. Rockwell applies the QA level necessary to assure each objective is maintained within the limits prescribed.

- Personnel Training

Personnel orientation, training, and qualification programs are conducted in accordance with Rockwell policy.(1) Records of the employee's current training status are maintained by Personnel and Public Relations. As a minimum, Operations personnel designated to perform processing operations and those observing these operations in close proximity are required to attend courses in radiological controls for radiation exposure protection.

9413219.1085

TABLE 10-1. Quality Assurance Verification System.

Elements	A&E facility design phase	Construction	Operations	Maintenance	Modified design phase	Material/part procurements
QA verification during element development (1)	Conceptual study (7) Pre-design review (7) Project design review (7) Pre-SAR (7) Project QA Plan (8) QA Program Plan (7) QA Systems audit (6)	Construction management (1) Project engineer (7) Construction QE (6) Plant safety (1)	Facility management (1) Process engineer (7) Personnel training (1) Radiological controls (16) Test plans (7) Inspection planning (6) QC verification (6) QE&C surveillance (6) QA laboratory system (9) Glove control system (1) Plant safety (1) Nuclear criticality safety (10) Essential material control (7)	Maintenance management (1) Fabrication specialist (1) Inspection planning (6) QE verification (6) QE&C surveillance (6) Calibration control (1) Plant safety (1)	Codes and standards (7) QA levels (6) QA drawing review (6) Document checking (7) Design review (7) QE&C document review (7) Packaging committee (PSSC) (1) Nuclear criticality safety (1)	Purchase requisition review (14) PQA review (6) Supplier control (6) PQA surveillance (6) Receiving controls (6, 14) Material certification (6) Essential material control (7) Hypalon glove control (11)
Quality Assurance evaluation of completed element activity (1)	Readiness Review Board (1) Startup Team (6) Final Safety Analysis Report (7) Annual Safety and Quality Assurance Audits (6) Periodic Safety and Quality Assurance Audits (6, 17) Material Audit of Rad-Material Shipping Practices Onsite (1)					

*All references from Section 10.6.

- Design Control

The Engineering Procedures Manual[7] identifies specific procedures for an effective design control system. These procedures establish organizational interface with the design to assure the safety and quality aspects are properly applied.

Design verification is conducted as the design of the facility and the associated components and support equipment are established or modified. Design verification is made by consensus agreement.

The S&QA function participates in all formal design verification reviews as do invited representatives from other functions. Additional waste container designs and related administrative controls are reviewed by a PSSC. Committee recommendations are reviewed prior to the design's release.

The R&E organization maintains a design change system to assure that the same level of review is provided design changes as was afforded the original document.

- Procurement Document Control

Procurement documents are generated and controlled by Rockwell policy and function manual procedures. Procedures provide guidance in the preparation of procurement orders to ensure that all the technical elements associated with the item for purchase are transmitted to the selected supplier.

An active supplier survey program is conducted by Rockwell Procurement Quality Assurance to assess the credentials of potential suppliers and to maintain a list of viable candidates for procurement placements. Procurement Quality Assurance assures the same degree of quality is applied throughout the purchasing cycle as is afforded to activities conducted onsite.

- Drawings, Procedures, and Instructions

Drawings, procedures, and instructions are prepared and issued in support of activities affecting quality and safety.

Functional procedures (test plans) and instructions generated by engineering to accomplish work tasks are reviewed by the cognizant Quality Engineer (QE) and approved when the acceptance criteria associated with the work task adequately represents quality assurance.[12]

- Inspection and Test Control

Quality Control (QC) observes operations/fabrications in progress and verifies that characteristics are in compliance with engineering documentations.

9413219.1086

The level of quality assurance effort in procurement areas and for onsite operations is in concert with the requirements determined by the design media. Design criteria is translated by the QE into specific inspectable characteristics for QC inspection and verification. Instructions follow.

- The verification of a configuration characteristic shall occur at the earliest practical point in the sequence.

- Choice of inspection method shall yield results consistent with the relative significance of the characteristic.

- Instructions shall be presented in a manner that will minimize the potential for error during the inspection process.

- Safety measures shall be applied commensurate with the potential impact on personnel and environmental safety.

In addition to planned inspections, QC representatives conduct ongoing surveillance of operations to assure that quality and safety requirements are being maintained in compliance with procedures designated for the area.

Infractions noted during the surveillance activity are reported to the appropriate management authority for action. Repetitive infractions are also reported on the Corrective Action Request.

Quality Engineering and Control (QE&C) conducts surveillance during the operation of lifting equipment to ensure its proper use, proof-loading, and identification. When in use, this equipment shall bear an unexpired proof-load date, and a code or tag indicating safety working-load limits. Equipment not in accordance with approved practices are tagged to preclude use.

- <u>Calibration and Control</u>

Instruments (including radiation protection instrumentation), test equipment, and working standards used are calibrated to standards traceable to the National Bureau of Standards or other recognized equivalent.

Instruments in use, other than those identified for "indication only," and test equipment shall have valid evidence of a current calibration. The QE&C personnel detecting the use of items not showing this evidence shall attach a "hold" tag to the item and issue a nonconformance report for immediate corrective action.

9413219.1087

Items removed from service for maintenance or calibration servicing are labeled to prevent inadvertent use. Item maintenance is conducted in accordance with the Production Support Maintenance Manual[13] and calibrations are in accordance with the Instrumentation Calibration Control Manual.[14]

- Incidents, Nonconformance, and Corrective Action Reports

  Nonconforming conditions noted by S&QA during equipment or material acceptance are reported as specified in the QA manual.[6]

  Incidents in the facility process system (e.g., system/component anomaly or procedure infraction) noted by QE&C, S&QA, or Process Engineering that could affect operational safety are cause for system shutdown. In addition, incidents may be subject to processing as "Unusual Occurrences" in accordance with formally established accident prevention standards.

  Deviations in which corrective action is required to prevent recurrence beyond the scope of disposition of the nonconformance are documented as "corrective action requests," generated by the individual recognizing the need, coordinated by QE, and responded to by responsible management.

- Quality Assurance Laboratory

  Analytical Laboratory personnel perform measurements of samples extracted from the process flow to characterize the contents per laboratory procedures[11] and process requirements. Results are utilized for balancing the system to the prescribed limits.

10.1.5.4 PUREX Analytical Laboratory. The Manager, PUREX Laboratory reports to the Manager, analytical laboratories (Fig. 10-8), who in turn reports to the Director, S&QA (see Section 10.7).

The Manager, PUREX Analytical Laboratory, is responsible for laboratory startup and operation and for ensuring safety, quality, cost effectiveness, and schedule effectiveness in the support of PUREX operations. The manager controls the QA analytical laboratory program for PUREX and is responsible for the organization of its components.

Personnel and facilities for all process control analyses are part of the PUREX Analytical Laboratory though support, particularly for product and environmental assays will be provided by the Plutonium Finishing and 222-S analytical laboratories. The number of personnel and their organizational assignments are determined by the Manager, PUREX Laboratory.

9413219.1088

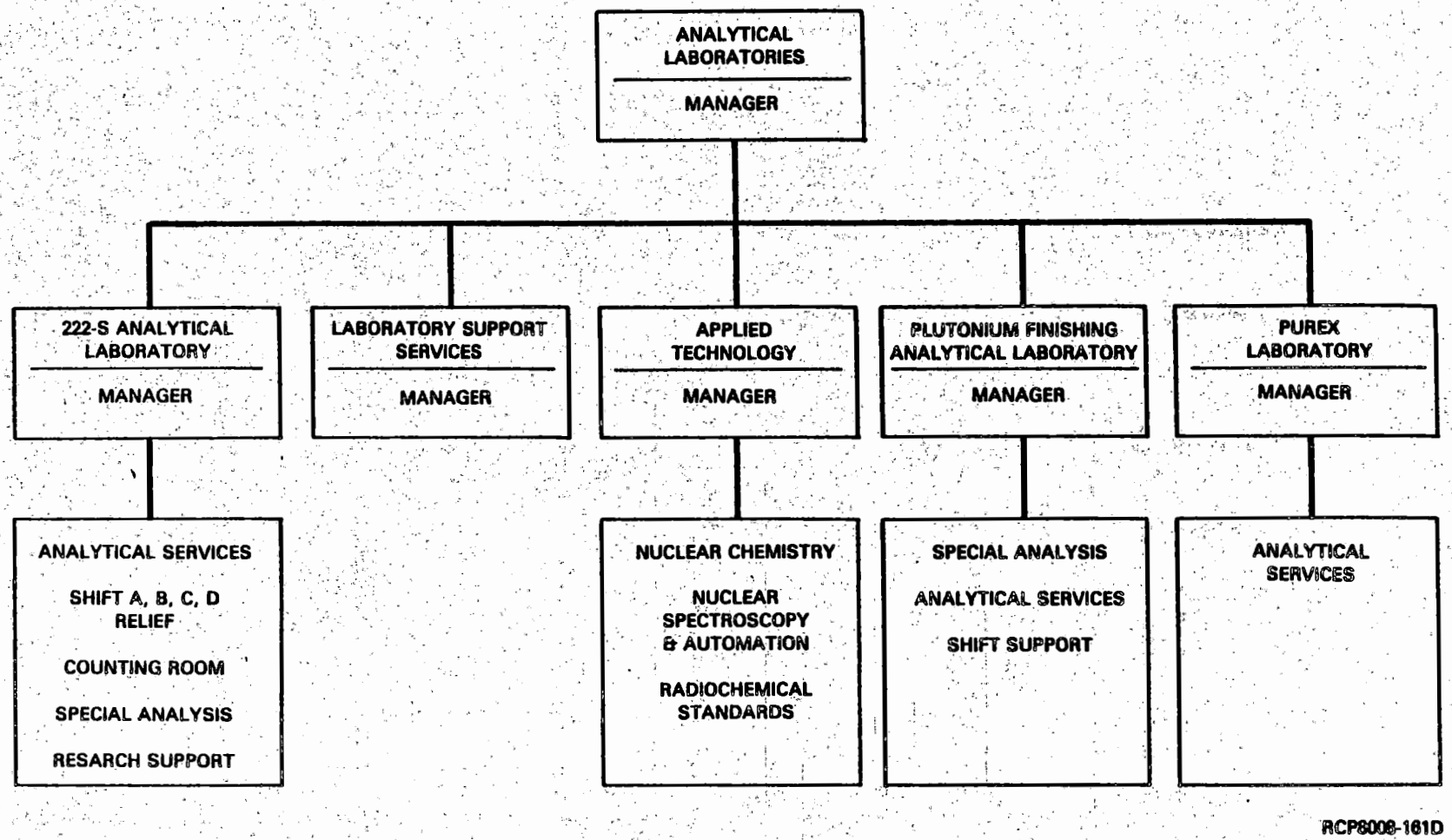


FIGURE 10-8. Analytical Laboratories.

Personnel with authority to make decisions for laboratory support of the PUREX process are always available. Personnel safety, laboratory procedural problem resolution, laboratory standard and reference difficulties, and schedule adherence are several areas in which decisions are made.

During the offshifts (i.e., swing, graveyard, and weekends), responsibility for the laboratory operation is delegated to the shift supervisor, with the day-shift staff "on call." All support services respond to the needs of the laboratory shift supervisor in matters concerning safety, quality assurance, and schedule effectiveness during that period of operations.

10.1.5.5 Research and Engineering Function. One of the missions of the R&E Function (Fig. 10-9) is to provide technical direction and leadership in support of plant operations. At PUREX, this is done through the PUREX Process Control (PPC) Group of the Process Engineering Department (Fig. 10-10). The PUREX separations process technology is provided by the Process Design Department (Fig. 10-11). Also in this function and in support of PUREX is the Design Engineering Department (Fig. 10-12) and the Development Engineering Department (Fig. 10-13).

The PPC Group provides the required process engineering support for the PUREX Plant. The types of support given depends on the operating status of the plant. Before startup, the main responsibilities of PPC Group personnel have been in the areas of technical materials, project liaison, and training. After the PUREX Plant is operational, the PPC Group will supply PUREX Operations with the required technical support for plant operations.

In the area of technical materials, PPC Group responsibilities include the preparation, evaluation, and/or review of the necessary procedures and documents for plant startup and operation. Examples include operability test procedures (OTP), operational safety requirements, and the process flowsheets.

Staff from PPC monitors the plant operations for adherence to safety, adherence to flow sheet requirements, and efficient operations. Based on this activity, PPC provides daily technical direction to operations (including shifts) and initiates design requests for systems upgrades. The PPC Group has been organized into a Head-end and Support Systems Team, a Plutonium Product Systems Team, a Solvent Extraction Systems Team, and a Special Projects Team to carry out the various responsibilities.

The Manager, PUREX/UO_3 Engineering Design Unit, reporting to the Manager, Engineering Design Group, is responsible for the PUREX Plant design activities. This unit is responsible for the engineering and design nonproject work, consisting primarily of maintenance, repair, minor facility improvements, and engineered equipment design. The drafting support for these engineers is provided by the Design Drafting Group. The draftsmen providing the majority of the support to the PUREX Plant are located near the PUREX Plant.

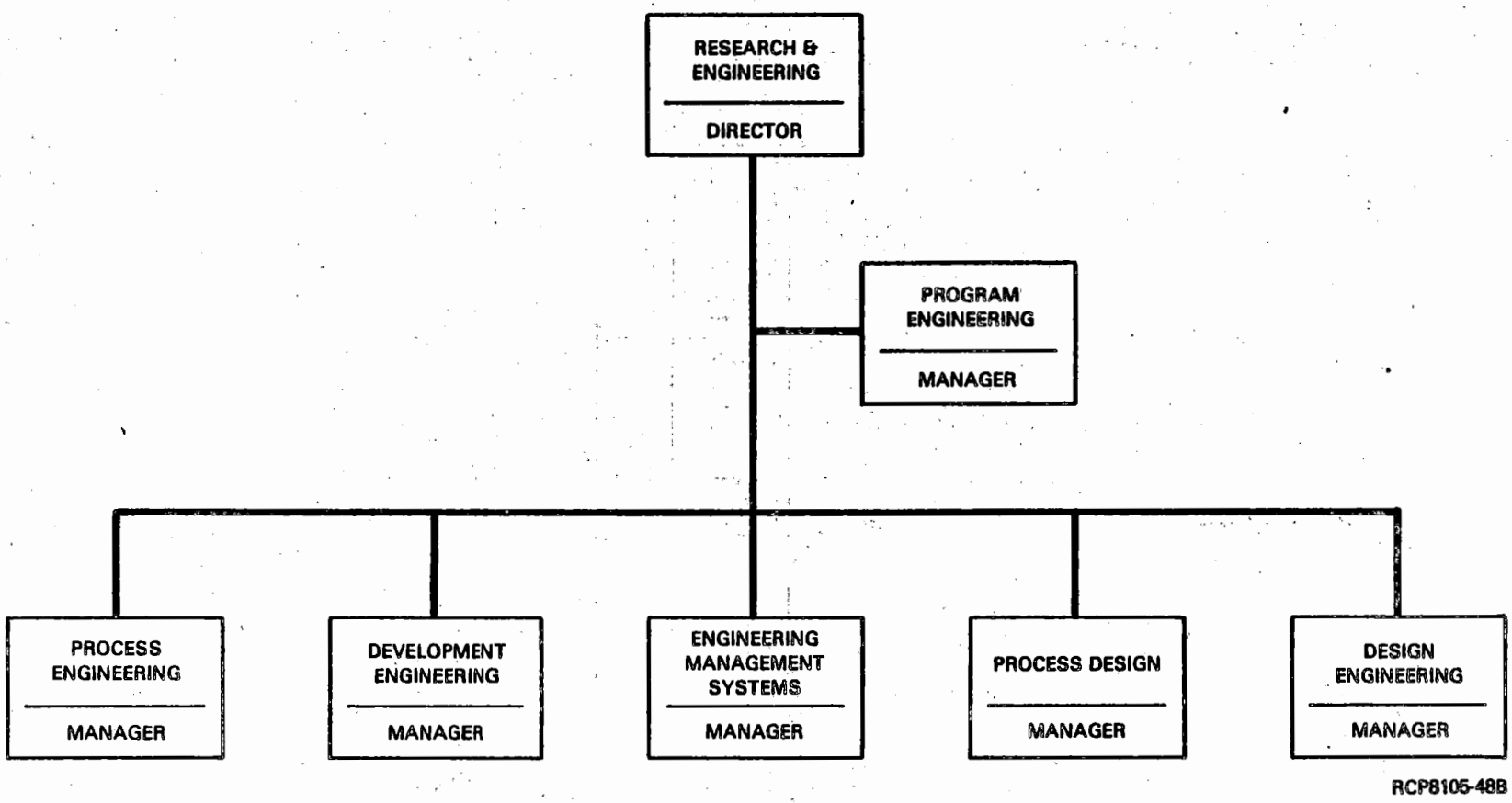


FIGURE 10-9. Research and Engineering Function.

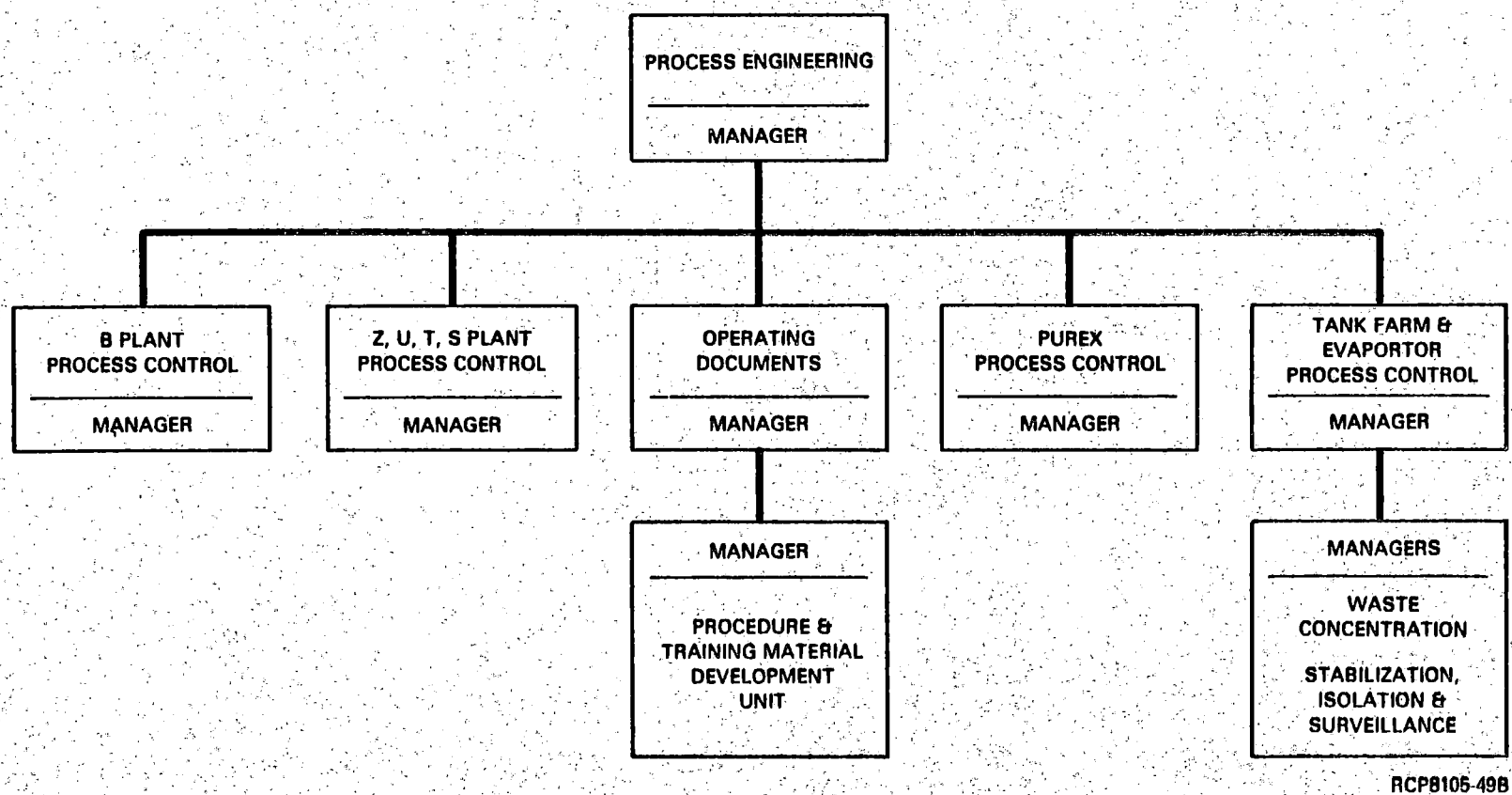


FIGURE 10-10. Process Engineering Department.

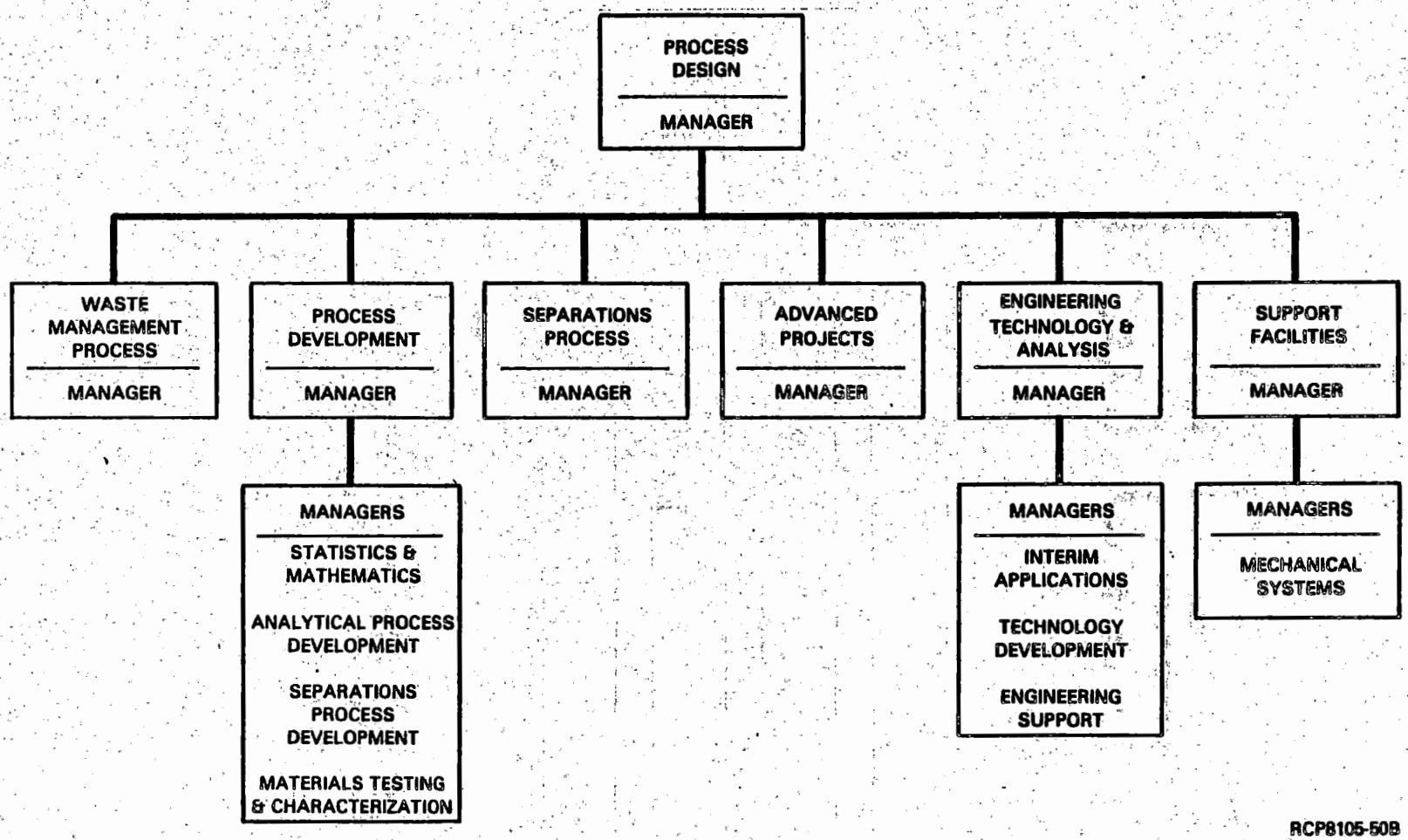


FIGURE 10-11. Process Design Department.

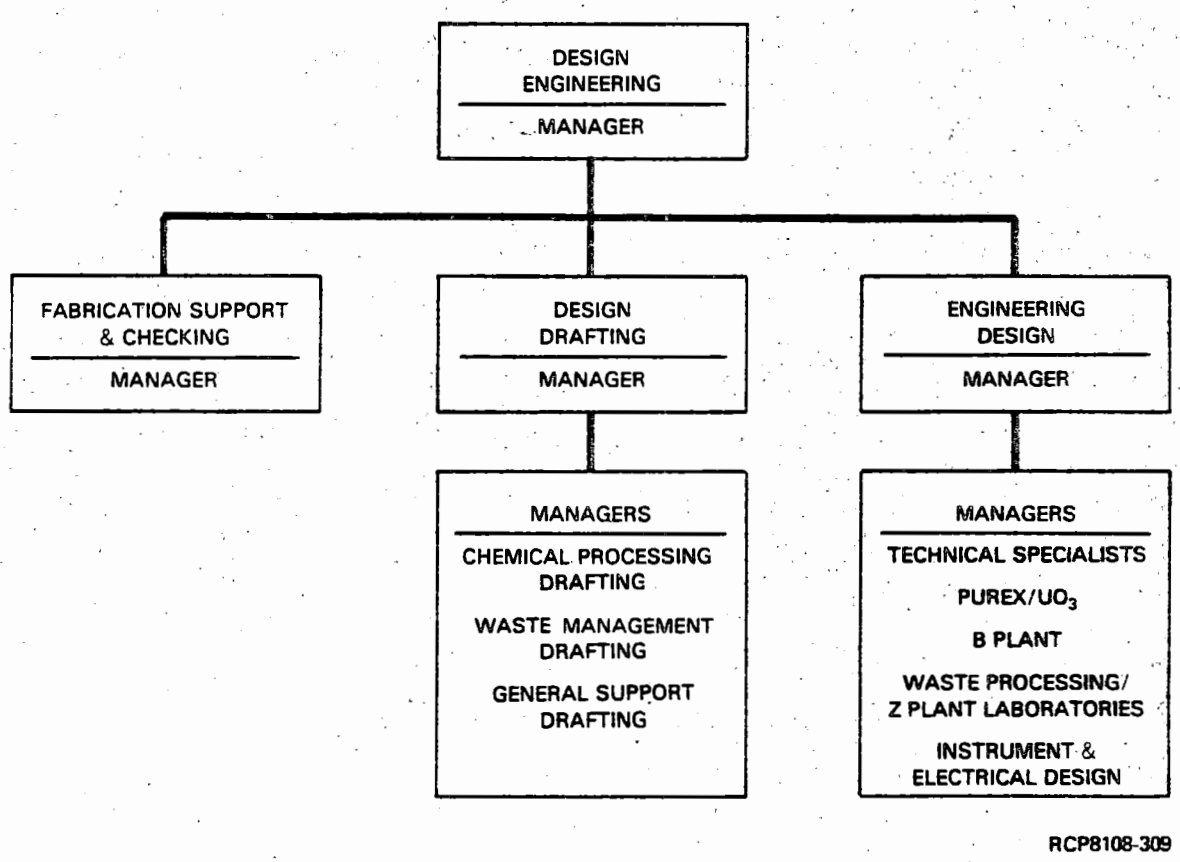


FIGURE 10-12. Design Engineering Department.

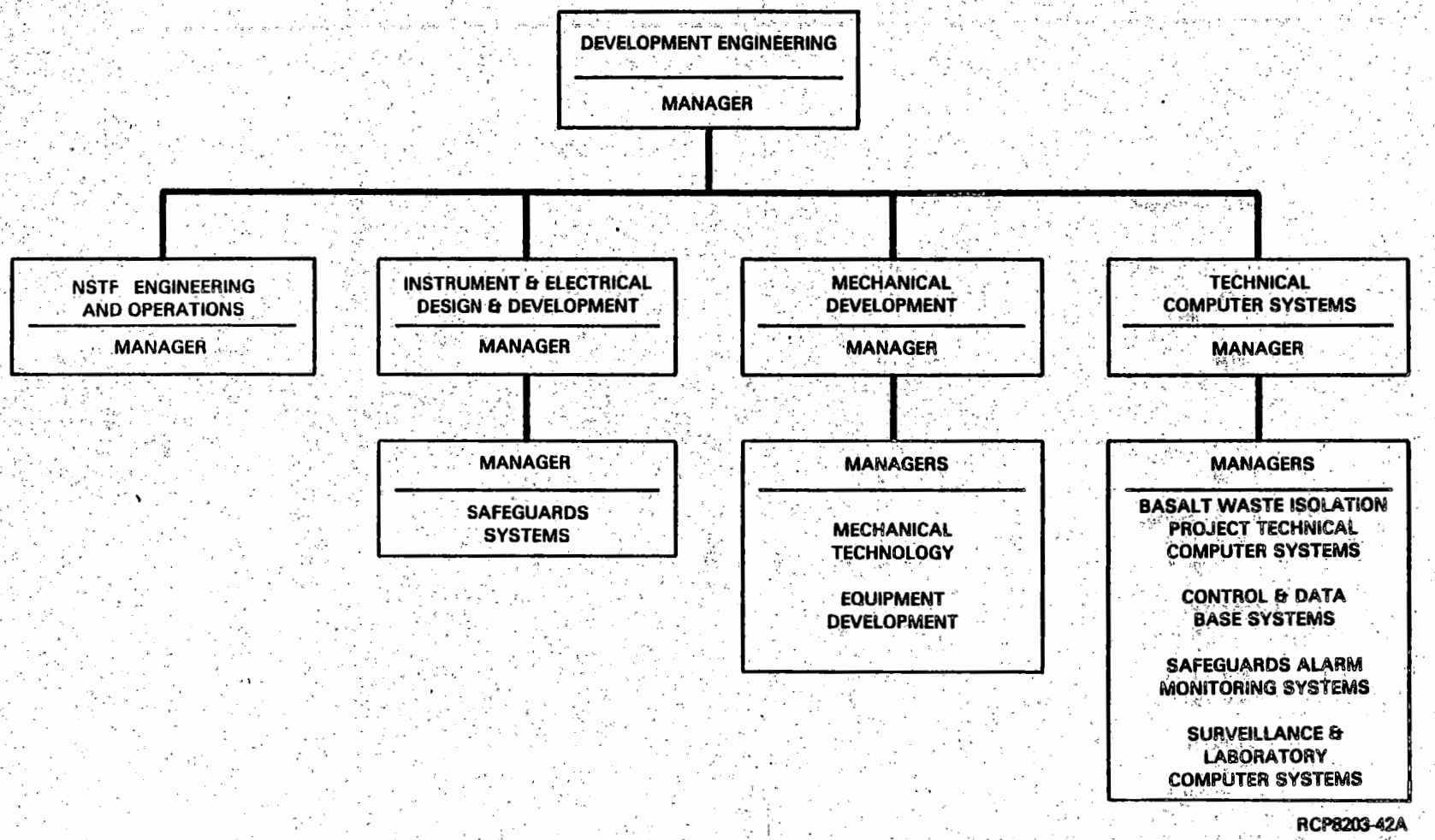


FIGURE 10-13. Development Engineering Organization.

Instrumentation and electrical design, development of equipment, and development of computerized systems are provided by the Design Engineering and Development Engineering Departments.

Additional engineering and design support is available in specialties within the R&E Function such as welding, metallurgy, corrosion, and stress analysis.

10.1.6 Qualifications of Management Personnel

Table 10-2 gives a brief summary of the responsibilities of managerial personnel associated with the principal, safety-related functions described in the above support organizations. Qualifications of supervisory personnel assure that the succession of authority can safely and competently proceed directly in accordance with the line organization. At least one individual under each manager or supervisor is specifically designated as an alternate in the event of illness or emergencies. The qualifications of key personnel in the operating and resident functional organizations are given in Table 10-3.

10.1.7 Staffing and Manpower Requirements

Manpower requirements from the various Rockwell functions in support of PUREX are provided in Table 10-4. The numbers include all craft, professional, management, and clerical personnel.

10.2 TRAINING PROGRAMS

Rockwell has established a program for the selection, training, and retraining of all individuals who operate, maintain, or supervise activities in the PUREX Facility. The PUREX Training Program Plan [15] was developed by the Personnel and Public Relations Function, Employee and Organizational Development Department and complies with the requirements of DOE Order 5480.1A, Chapter V.[2]

The PUREX Training Program Plan defines the following:

- Who is responsible for Rockwell training
- What are the operational levels and assignments of responsibility for PUREX operations
- How were the mandatory training courses and programs determined
- What are the training programs qualification and certification requirements

Table 10-2. Qualifications of Management Personnel Associated with Other Support Organizations.

Position	Education	Management experience	Industrial experience	Nuclear experience	Reports to	Responsibilities
Director, Research and Engineering	College degree in engineering, physical, science, or equivalent	Demonstrated administrative and managerial skills	12-yr minimum in related industry	6-yr minimum in related industry	Vice President and General Manager	Direct and control Research and Engineering assignments
Manager, Process Engineering	(same)	(same)	10-yr minimum in related industry	5-yr minimum in related industry	Director, Research and Engineering	Direct and control process engineering assignments
Director, Safety and Quality Assurance	(same)	(same)	12-yr minimum in related industry	6-yr minimum in related industry	Vice President and General Manager	Direct and control Safety and Quality Assurance assignments
Manager, Radiological Protection	(same)	(same)	(same)	(same)	Director, Safety and Quality Assurance	Direct and control Radiological Protection assignments and programs
Manager, Criticality Engineering and Analyses	(same)	(same)	10-yr minimum in related industry	5-yr minimum in related industry	Manager, Radiological Protection	Define, administer, and evaluate nuclear criticality safety
Manager, Industrial Hygiene and Safety	(same)	(same)	(same)	(same)	Director, Safety and Quality Assurance	Develop and administer industrial safety and hygiene and fire protection programs
Manager, Environmental Analysis and Monitoring	(same)	(same)	(same)	(same)	Director, Safety and Quality Assurance	Develop and administer environmental analysis and monitoring programs
Manager, Radiation Monitoring	(same)	(same)	(same)	(same)	Manager, Radiological Protection	Administer a program to minimize radiation exposure

Table 10-3. Qualifications of Key PUREX Operating and Resident Support Personnel.

Position	Education	Management experience	Industrial experience	Nuclear experience	Position accountable to
Operational Components					
Manager, PUREX Operations Department	College degree in engineering or physical science	Demonstrated administrative and managerial skills	10-yr minimum in related industry	5-yr minimum in related nuclear field	Director, Production Processing
Manager, PUREX Processing	(same)	(same)	8-yr minimum in related industry	4-yr minimum in related industry	Manager, PUREX Operations Department
Manager, PUREX Oxide and UO_3 Operations	College degree or equivalent in engineering or physical science	(same)	(same)	(same)	(same)
Shift managers	(same)	Demonstrated supervisory skills	6-yr minimum in related industry	3-yr minimum in related industry	Minimum, PUREX Process or Manager, PUREX Services and UO_3 Operations
Shift support managers	High school graduate	Demonstrated effectiveness in inter-personnel relations	4-yr minimum in related industry	2-yr minimum in related industry	Shift managers
Resident Functional Support Components					
Manager, PUREX Process Control	College degree in engineering or physical science	Demonstrated administrative and managerial skills	10-yr minimum in related industry	6-yr minimum in related nuclear field	Manager, Process Engineering
Manager, Separations Process	(same)	(same)	(same)	(same)	Manager, Process Design
Manager, PUREX Maintenance	College degree or equivalent in engineering or physical science	(same)	(same)	(same)	Manager, East Area Maintenance
Supervisor, PUREX Radiation Monitoring	(same)	(same)	(same)	(same)	Manager, Radiation Monitoring
Manager, Quality Engineering Process Facilities	(same)	Demonstrated administrative and managerial skills	(same)	(same)	Manager, Quality Engineering
Manager, PUREX Laboratory	College degree in chemistry or physical science	(same)	(same)	(same)	Manager, Analytical Laboratories

TABLE 10-4. Manpower Requirements for the PUREX Plant.

Classification	Total manpower[a]	Day shift at plant[b]	Typical offshift at plant	Support away from plant
Production Processing				
Process Operators	(17) 96	36	17	-
Power Operators	(2) 11	4	2	-
Crane Operators	(1) 5	2	1	-
Oxide Process (XYZ)	(4) 12	13	4	-
Other	(2) 10	8	5	-
Other Plant Operations Organizations				
PUREX Plant Maintenance	(10) 74	50	18	2-6
Production Control	26	22	3	1
Other (Shops and Services)	(11) 36	36	-	-
Research and Engineering				
Process Control Engineering	(40) -	32	6	2
Design Engineering	(25) -	25	-	
Development/Engineering/ Process Design	(45) -	45	-	-
Other	- -	-	-	-
Safety and Quality Assurance				
Radiation Monitoring	(5) 38	22	12	4
Lab Technicians	(4) 44	9	35	9
Lab Chemists	(1) 7	7	-	-
Shift Supervisors	(14) 14	6	4	4
Other	(4) 6.5	5.5	-	-
Other Functions				
Total (Program Office, PBM, Finance and Information Services, Safeguards/Material)	(23) -	23	-	-

[a]Denotes management/supervisory personnel which are included in the total manpower.

[b]"At Plant" includes only those personnel whose normal work location is 202-A building. Other personnel who may visit 202-A are included in the "support away from plant" column.

- How were PUREX personnel identified for compliance to DOE Order 5480.1A, Chapter V[2]
- What methods were implemented to develop and complete a training needs analysis
- What documentation is required for mandatory training and retraining
- Where is the required documentation stored
- How is Rockwell assured of satisfactorily trained personnel for startup and continued operations of the PUREX Facility.

10.2.1 General Training Program Description

All new Rockwell employees attend New Employee Orientation. This program gives the new employee insight into Rockwell's purpose and concentrates on subjects related to the employee's well-being (e.g., safety programs, fire protection, conduct in radiological areas, etc.).

Employees working at PUREX for the first time receive immediate basic training in radiation protection as well as plant-specific orientation. Emergency response training, including security, diversion, and terrorist awareness, is also provided.

10.2.1.1 Training in Radiation and Criticality Safety. Generic training in radiation and criticality safety is provided for all PUREX operations and maintenance personnel involved in routine radiation work and fissile material handling. Thus, operating and maintenance personnel may be categorized as radiation workers or nonradiation workers and fissile material handlers or nonfissile material handlers.

10.2.1.1.1 Radiation Workers. Personnel who require routine or emergency access to posted radiological areas, as determined by their respective managers, will be "qualified" radiation workers. However, persons not qualified as radiation workers may enter posted radiological areas if escorted by a qualified radiation worker. The nonqualified individual is not permitted to handle, process, or work on radioactive materials.

Radiation workers become officially qualified upon successful completion of radiation safety training courses and by satisfactory fitting and training in the use of available respiratory protection devices.

The radiation safety training requires, in part, that each participant demonstrate the following to the satisfaction of a designated instructor:

- The ability to read the commonly used self-reading dosimeters
- The proper procedure for putting on and removing protective clothing and protective respiratory devices

- The proper procedures for entering and leaving a contaminated area
- An understanding of the theoretical basis of radiological protection standards.

10.2.1.1.2 Fissile Material Handlers. A fissile material handler is defined as a person whose work assignment may require the handling of more than 15g ^{239}Pu, or equivalent, (3% or more of the minimum critical mass), and who has successfully completed the required training courses.

Fissile material handlers and all others assigned to a fissile facility such as PUREX are required to receive criticality safety training appropriate to their level of involvement and responsibility. Individuals who have successfully completed formal (classroom) training may be allowed to handle fissile materials as part of on-the-job training.

Radiation worker and criticality safety training provide operating personnel with a basic understanding of criticality and radiation safety requirements. General requirements for prevention of criticality and control of radiation hazards are also taught.

10.2.1.2 Specialized Personnel and Training. Specialized personnel are required to obtain additional plant and process specific skills. Specialized personnel include nuclear process operators, process control engineers, radiation monitors, analytical laboratory personnel, systems power operators, and maintenance personnel. The following summaries have been excerpted from Reference 15 to represent the methods used for training, skill maintenance, and retraining of specialized personnel.

- Nuclear Process Operators

  This training program consists of two parts:progression/promotion and job certification.

  Under progression and promotion, operators are required to successfully complete written examinations at 6, 18, 30, 42, and 54 mo. After completion of the 54-mo Nuclear Process Operator examination, retesting is required every 2 yr to maintain this responsibility level.

  Job certification is the practical, on-the-job portion of the training program. Operators are required to successfully complete a written examination and a walk-through evaluation demonstrating competence in performing a specific job. Recertification is required every 2 yr.

- Process Control Engineers

  This training program is divided into three phases: phase I (site-generic) includes training pertinent to all Rockwell-operated Hanford nuclear processing facilities; phase II

(plant-specific) covers generic information for the PUREX Facility; phase III (process-specific) covers detailed information on "specialized areas" within the PUREX Facility.

Upon successfully completing the first two phases, including the required examinations, the candidate can be certified as a Process Control Engineer (PCE) for the specific facility. However, this is a limited certification in that the PCE may not function in those areas designated as "specialized areas." Successful completion of the phase III training, including examinations, is required for functioning in such areas.

Training is a mixture of self-study and on-the-job experience under the guidance of qualified personnel and is supplemented with technical workshops and seminars. Self-study is based on technical documents, training manuals, administrative manuals, and operating procedures. Study guides, including specific learning objectives, are provided to the candidate.

- Radiation Monitors

This program establishes training requirements and methods for certifying the technical competence of Radiation Monitors [Rockwell job title: Radiation Protection Technologists (RPT)]. The purpose of certification is to assure that RPTs are fully qualified to perform the required duties in a safe and proficient manner.

Required training includes general requirements, academic training, on-the-job training and facility training and is evaluated by examination. Levels of RPT are Radiation Protection Trainees, Junior RPT, RPT, and Senior RPT.

The requirements for progression to Senior RPT include formal training of increasing depth, continued on-the-job training and experience, and formal testing leading to certification. Once certified as a Senior RPT, recertification will occur at 2-yr intervals.

- Laboratory Technologist

This training program consists of general laboratory familiarity training and on-the-job training. Both of these areas require demonstration of proficiency through the use of documented performance testing. The general laboratory familiarity training also requires written testing.

Retraining is conducted when a procedure undergoes a significant change, every 5 yr, or when a technologist demonstrates difficulty in performing a procedure.

- Chemist

  Laboratory chemists are expected to possess an academic background that provides a theoretical base and analytical experience. Specialized training consists of laboratory operating procedure familiarization and self-study guided by reading lists and is a combination of formal and on-the-job training designed to ensure that required duties are performed efficiently and safely.

  Performance is evaluated, during the conduct of laboratory procedures, by observation, and verbal questioning. Need for additional training and retraining is determined by periodic performance evaluation.

10.2.1.3 Operations Managers. This training program applies to all first-hand, second-level managers of operating personnel. Operations Manager (OM) candidates are expected to attain certification within 12 mo after entry into the training program. Training for certification consists of completion of the required reading/study topics; thorough understanding of plant and technical information as evidenced by passing written examinations; demonstration of on-the-job knowledge as evidenced by walkthrough examinations of all areas of the operations, equipment, and procedures within the area of future responsibility; and successful completion of written examinations on emergency procedures and abnormal plant conditions.

The OM recertification is required every 2 yr by written examinations and walk-through evaluations. (To maintain certification, OM are required to successfully complete the emergency procedures and abnormal plant conditions examination annually.)

10.2.2 Administration and Records

The Employee and Organizational Development (EOD) Department maintains all official training files. These files contain the original record of all individual training results, as well as the original copy of job lists, tasks, and documents relating to qualification and certifications. Duplicate records for PUREX employees will be kept at the PUREX Facility.

The EOD Department maintains the official records of criticality safety training status. A copy of the criticality training records are transferred with the employee whenever the employee is assigned to another organization within Rockwell.

10.2.3 Adequacy of Training

The adequacy of training is assessed in several ways. Occurrence reports are reviewed by each organization to determine if there is a need for training. The review may require a change in some aspect of training. Training and documentation would then be developed and implemented.

Evaluation sheets are periodically presented to employees to evaluate the training courses'/programs' contents, materials, and instructor. The evaluation sheets are reviewed by the instructor, plant personnel, and EOD personnel.

Audits by S&QA, EOD, and functional internal audits substantiate that training courses/programs are being conducted and documented according to government regulations and Rockwell policies and practices.

10.3 NORMAL OPERATIONS

10.3.1 Plant Procedures

Responsibility is assigned and authority is delegated to each member of the Vice President and General Manager's staff for issuance of formalized procedural manuals covering their respective assigned activities to implement policy, functional programs, and DOE management directives affecting the manner in which contractual activities are performed.

The OSRs define the acceptable conditions, safe boundaries and bases, and management controls required to assure safe operation of the PUREX Plant. The OSRs include both technical and administrative matters. The technical matters relate to the features (process variables, systems, or components) of the facility that are of controlling importance to safety. The technical matters also include effluent and environmental specifications addressed to the attainment of "as low as practicable" levels of releases and exposures. Administrative matters address the organizational and functional requirements that are important to the achievement and maintenance of safe operations of the PUREX Plant. The OSRs provide the basis for stable plant operations with a manageable margin between the normal operating range and the safety limits. The OSRs are incorporated into guidelines for Plant Operations.

Guidelines for Rockwell's processing activities are generally written as operating procedures. An operating procedure is a written, step-by-step instruction designed to provide guidance and advance approval for the safe and efficient performance of a specified task. Operating procedures may take one of the two following forms.

- Standard Operating Procedure (SOP)

  A SOP is a procedure for a selected process operation that provides sequenced, step-wise, and explicit instructions. In format, the SOP provides a brief explanation of the system and the operations to be conducted, with attention to the specific safety requirements for the job being done, including radiological, chemical, and criticality safety hazards and environmental risks.

- Plant Operating Procedure (POP)

  A POP is a procedure structured more formally than the SOP. Development is controlled by specifications. This is the primary format for procedure development for PUREX.

Several other types of procedures directly contribute to safety and efficiency of operations at PUREX.

- Laboratory Operating Procedure (LOP)

  The LOP describes routine laboratory practices, administrative practices, facility operation, and equipment operations requiring considerable special operator technique. The LOP contains a section on safety that discusses unique or unusual safety items that pertain to the use of the procedure. The LOP is administrative in nature.

- Analytical Procedure (AP)

  An AP describes work methods and technical considerations for laboratory analyses used in determining the quality or quantity makeup of samples to be analyzed. An AP is technical in nature.

- Radiation Work Permit (RWP)

  An RWP provides a detailed, sequential description of the conduct of radiation work. Specific procedures are in place for all aspects of PUREX work.

- Health Physics Procedures (HPP)

  The HPP is a procedure that governs specific radiological protection practices. These include instrumentation calibration checks, site dosimetry, personnel dosimetry, skin decontamination, etc. The HPP is approved by the Radiological Protection Department.

- Process Test Procedure (PTP)

  A PTP permits testing of significant changes in process flowsheets, specification, or equipment. A PTP outlines the implementation of a process test plan, which includes an informal safety analysis of the proposed changes, defining potential personnel safety hazards or detrimental effects on the equipment, process, facility, or environment.

- Acceptance Test Procedure (ATP)

  An ATP describes procedures to test newly constructed systems or components by checking the response of indicators, instruments, and controllers. Since operational conditions may not exist, the signals may have to be simulated.

- Operating Test Procedure (OTP)

  An OTP provides instructions for checking and testing of a process system--the entire system, its components, or its interrelationships to other systems.

The Process Engineering Department is responsible for the development of operating procedures for PUREX. The task is assigned to the Operating Documents Control Unit. The Operating Documents Control Unit analysts collect applicable technical and process information from knowledgeable sources to prepare each procedure. Procedures are in a format that lends itself to safe, efficient operation; this format also is consistent with the Rockwell policy statement of safe operation and with the OSR developed for the PUREX Plant. Technical and functional information from PPC and operations personnel is structured into user-oriented operating procedures. Additional inputs to and approval of these procedures are provided by Production Processing or the applicable S&QA and Nuclear Material Control organizations prior to issuance, so as to provide sets of written instructions designed to enable the user to perform a specific task safely and efficiently. The appropriate operating procedures are available in all work and process control areas.

The Production Processing Function is responsible for determining the need for new or revised operating procedures and for approving operating procedures for use. Procedures require strict compliance to ensure safe operation and are structured in compliance with Reference 2 for both radiation and criticality safety.

The Operating Documents Control Unit is responsible for issuance, removal, record keeping, auditing, and all other activities associated with controlling operating procedures for PUREX. The operating document control system is set up to provide the following:

- A uniform system of procedure control assuring that there are no obsolete, void, or outdated procedures in the operating areas of the PUREX Facility

- Procedure placement and removal services

- Procedure change notices to affected personnel

- A listing of all PUREX procedures for review or audit and the status of any procedure.

All procedures are reviewed and formally changed at any time a change is made in the affected area.

10.3.2 Records

The establishment of a system for creating and preserving records is a fundamental management responsibility of Rockwell. The Rockwell records policy is contained in Reference 1. The Records Department of the Finance and Information Services Function has the responsibility for assuring the existence of a records system that meets the needs of Rockwell and DOE.

Records are maintained for PUREX by various support organizations (e.g., historical operation by PUREX Operations and Process Engineering, radiation exposure by Radiological Engineering, environmental samples/surveys by Environmental Analysis and Monitoring, quality assurance by S&QA, etc.). Records to be maintained include the following:

- Records of personnel exposure
- Records of plant modifications
- Records of personnel qualifications and training
- Records concerning special nuclear materials inventories and accountabilities, other hazardous materials inventories, plant radioactive effluent releases, radioactive waste disposal, and plant radiation and contamination surveys
- Records of operations relative to the constrictions and boundaries of the safety limits and requirements, including sample logs, instrument settings or charts, activity logs, and maintenance and surveillance logs.

Record retention requirements are defined in conjunction with the description of the QA program provided in Section 10.7.

10.4 EMERGENCY PLANS

Rockwell has a complete set of emergency procedures(5) that are responsive to Rockwell policy pertaining to emergency preparedness. The emergency plan manual is published so that managers may have it available for reference at all times. Each member of management is familiar with its contents and uses it for training of subordinate personnel.

Changes in facilities, in Rockwell organization, or in government requirements necessitate frequent revisions of individual procedures. To assure that manuals are up to date, managers promptly initiate appropriate revisions. In addition, the emergency plan is subject to review and revision at least annually by the Emergency Planning Review Committee. This committee reports directly to the Vice President and General Manager.

The manual discusses general emergency responses and plans, assigns responsibilities, details methods of and pathways for communication in an emergency, describes Rockwell's emergency facilities, and contains a directory and emergency log. Types of emergencies covered include criticality, fire/explosion, natural forces, civil disobedience, and bomb threats.

Under the emergency plan, the Manager, Safeguards and Security (alternative: Director, S&QA) is considered to be the Emergency Director who is supported by trained, local emergency teams and plant-wide emergency and support groups.

Incorporated into the emergency plan is the identity of offices of the emergency organization and a description of the system for notifying appropriate persons both within and outside of Rockwell. Rockwell Emergency Control, under the direction of the Vice President and General Manager, has primary responsibility for maintaining an overview of onsite emergency response and for fulfilling corporate reporting requirements and local reporting to DOE-RL; DOE, in turn, is responsible for notifying other appropriate agencies. Detailed information is contained in the emergency plan communications section.

The PUREX Plant is equipped with personnel decontamination rooms and the necessary decontamination supplies, showers, and alpha and beta-gamma radiation-detection equipment (see Chapter 8.0). Personnel decontamination is covered in detail in written internal procedures. An RPT trained in personnel decontamination techniques is always on duty at the PUREX Plant. Additionally, in the event of an emergency, trained personnel are always available in other facilities in the 200 East and 200 West Areas.

Emergency drills are scheduled and conducted for PUREX Plant personnel. These include nuclear excursion evacuation, area evacuation, and fire drills. Each type of drill is conducted at least annually. The nuclear excursion detector systems are checked weekly during plant operations. Plant telephone crash alarm systems are checked weekly. The sirens that signal area evacuation are checked monthly. Fire alarm systems are checked every 6 mo.

10.4.1 Fire Protection

The Fire Protection Department of the Safeguards and Security Function has the responsibility for preparing prefire plans in conjunction with plant management for each facility in PUREX. Prefire plans have been prepared for all PUREX Plant facilities. The plans for major facilities are updated at least every 3 yr, and for lesser facilities, every 5 yr. Fire protection personnel review these plans in their training sessions three to four times

each year. Fire protection personnel and equipment from the 200 Areas Fire Station are available to react to any emergency in the PUREX Plant. They are located within 3 mi of the plant and can be available at any PUREX Plant Facility within 6 min after receipt of signal. Additional backup is available at the 100 Area Fire Station, 7 mi from the 200 Areas Fire Station, and at the 300 Area Fire Station, 20 mi from the 200 Areas Fire Station. In addition, there is a mutual aid agreement with the City of Richland Fire Department making personnel and equipment available in the case of extreme emergency. All members of the Fire Protection Department have had advanced first-aid training.

All Patrol personnel have also had advanced first-aid training. The Patrol Department has about 50 people available in the 200 Areas during the normal work week and 46 people available on shifts, weekends, and holidays to assist in an emergency.

10.4.2 Emergency Shutdown of Process Systems

Specific emergency shutdown procedures are prepared for each processing system or area by Process Engineering. These procedures are located and maintained within the operating procedure system. They are issued and controlled in the same manner as operating procedures.

10.4.3 Emergency Medical Services

Medical services are supplied by the Hanford Environmental Health Foundation. The 200 East Area First-Aid Station is located less than 1 mi from PUREX. The station is fully equipped with first-aid supplies. A registered nurse is stationed in the 200 East Area First-Aid Station during the normal work week and is located in either the 200 East or 200 West Areas on shift, weekends, and holidays. Ambulances are located at the 200 Areas Fire Station. Additional ambulances are stationed in the 100 N Area and the 300 Area, and are available in an emergency. Fire personnel serve as ambulance drivers and assist the nurse as required.

Additional medical support provided by the Hanford Environmental Health Foundation includes a competent medical staff knowledgeable in radiation-contaminated injury treatment techniques, as well as decontamination methods. The foundation has a well-equipped radiosurgery facility, located near Kadlec Hospital in Richland, for treatment of injured, contaminated personnel. A medical doctor is always "on call" in the event of extreme emergency.

Details of established radiation exposure guidelines and arrangements for the formation of an emergency technical advisory group are also included in the emergency plan.

10.5 DECOMMISSIONING

Decommissioning of PUREX is anticipated to be accomplished as part of an overall program effort for DOE facilities.[12]

Rockwell personnel have been involved in both in-plant and onsite decommissioning efforts involving radioactively contaminated facilities. In-plant experience includes decommissioning of the Recuplex Plutonium Facility and the Division of Military Applications Plutonium Fabrication Facilities. Onsite experience includes the decommissioning of the P11 Plutonium-Contaminated Critical Testing Facility. External radiation exposure and internal plutonium deposition were well controlled during these activities. No applicable limits were exceeded.

Rockwell International personnel have experience in organizing, managing, and conducting decontamination and decommissioning programs. Initial activities began in 1969 with decommissioning of the Hallam and Piqua Reactors. These activities have evolved into a comprehensive program established to decommission eight facilities at Atomics International's Santa Susana Nuclear Development Site.

10.6 REFERENCES

1. Rockwell, Rockwell Policies Manual, RHO-MA-100, Rockwell Hanford Operations, Richland, Washington (July 1977).

2. DOE, Safety of Nuclear Facilities, DOE Order 5480.1A, Chapter V, U.S. Department of Energy, Washington, D.C. (August 1981).

3. DOE-RL, Quality Assurance, DOE-RL Order 5700.1, U.S. Department of Energy, Richland Operations Office, Washington (August 1980).

4. DOE, Safety Analysis and Review System, DOE Order 5481.1A, U.S. Department of Energy, Washington, D.C. (August 13, 1981).

5. Rockwell, Rockwell Hanford Operations Emergency Plan, RHO-MA-111, Rockwell Hanford Operations, Richland, Washington (October 1977).

6. Rockwell, Quality Assurance Manual, RHO-CD-150, Rockwell Hanford Operations, Richland, Washington (May 1978).

7. Rockwell, Engineering Procedures Manual, RHO-MA-115, Rockwell Hanford Operations, Richland, Washington (March 1978).

8. Rockwell, Project Quality Requirement Plan, RHO-QA-PL-1, Rockwell Hanford Operations, Richland, Washington (October 1981).

9. Rockwell, Environmental Protection Standards, RHO-MA-139, Rockwell Hanford Operations, Richland, Washington (June 1981).

10. Rockwell, Radiological Controls, RHO-MA-220, Rockwell Hanford Operations, Richland, Washington (November 1978).

11. Rockwell, Analytical Laboratories Operating Instructions Manual, RHO-MA-138, Rockwell Hanford Operations, Richland, Washington (October 1977).

12. DOE, General Design Criteria for DOE Facilities, DOE Order 6430 (Draft), U.S. Department of Energy, Washington D.C. (June 1982).

13. Rockwell, Production Support Maintenance Manual, RHO-MA-137, Rockwell Hanford Operations, Richland, Washington (December 1978).

14. Rockwell, Instrumentation Calibration Control Manual, RHO-PS-MA-1, Rockwell Hanford Operations, Richland, Washington (February 1983).

15. Rockwell, PUREX Training Program Plan, RHO-PR-PL-1, Rockwell Hanford Operations, Richland, Washington (September 1982).

16. Rockwell, Material Policy and Procedure Manual, RHO-MA-135, Rockwell Hanford Operations, Richland, Washington (March 1978).

17. Rockwell, Accident Prevention Standards, RHO-MA-221, Rockwell Hanford Operations, Richland, Washington (March 1979).

THIS PAGE INTENTIONALLY
LEFT BLANK

9413219.1112

PUREX FSAR

CHAPTER 11.0 OPERATIONAL SAFETY REQUIREMENTS

THIS PAGE INTENTIONALLY
LEFT BLANK

This page intentionally left blank.

CONTENTS

11.0 Operational Safety Requirements 11-1

11.1 Introduction 11-1
11.1.1 Applicability and Objectives 11-1
11.1.2 Definitions 11-1
11.1.3 Operational Safety Requirements Violations 11-2
11.1.4 Corrective or Responsive Action 11-2
11.1.5 Revisions 11-4
11.1.6 Audit Link 11-4

11.2 Safety Boundaries 11-4
11.2.1 Iodine-131 Content of Fuel 11-5
11.2.2 Ammonia Concentration in Offgas Systems 11-6
11.2.3 Hydrogen Concentration in Offgas Systems 11-8
11.2.4 Fuel Receipt and Handling 11-9
11.2.5 Flammable Solvent Vapors and Aerosols 11-10
11.2.6 Concentrator Temperatures 11-13
11.2.7 Waste Treatment--Sugar Denitration 11-15
11.2.8 Waste Treatment Operations 11-16
11.2.9 Plutonium Conversion and Oxide Powder Handling 11-18

11.3 Safety Conditions 11-19
11.3.1 Nuclear Criticality Prevention 11-20
11.3.2 Authorized Materials 11-23
11.3.3 Gaseous Effluents 11-24
11.3.4 Liquid Effluents 11-26
11.3.5 Administrative Control System 11-27

11.4 References 11-29

9413219.1115

This page intentionally left blank.

11.0 OPERATIONAL SAFETY REQUIREMENTS

11.1 INTRODUCTION

11.1.1 Applicability and Objectives

The Operational Safety Requirements (OSR) herein, in accordance with U.S. Department of Energy (DOE) Order 5480.1A, Chapter V, define acceptable conditions, safe boundaries and bases, and management controls required to ensure safe operation of the Plutonium-Uranium Reduction Extraction (PUREX) Plant during the processing of irradiated fuels.(1) Operations outside of the specified boundaries and conditions could result in an unacceptable level of risk to the public, site workers, or environs or a compromise of facility integrity.

9413219.1116

11.1.2 Definitions

The OSR are divided into the three following categories.

- Safety Boundaries

  Safety Boundaries refer to the values of safety-related process variables which are observable and controllable. Operation outside of a safety boundary may result in serious consequences.

- Safety Conditions

  Safety Conditions refer to those technical conditions or restrictions essential for safe operation.

- Control Features

  Control Features supporting each safety boundary and safety condition are delineated in this document. Specific Limiting Control Settings (LCS) and Limiting Conditions for Operation (LCO) to implement each of these control features are defined in internal Rockwell Hanford Operations (Rockwell) documentation.(2) The LCS and LCO define the minimum equipment operability requirements, repair schedules, operability testing or calibration schedules, and more conservative operating limits.

11.1.3 Operational Safety Requirements Violations

The following items are OSR violations:

- Operating outside of a safety boundary
- Failing to meet the requirements of a safety condition
- Failing to have a specific control feature in place.

11.1.4 Corrective or Responsive Action

A violation of a safety boundary or safety condition would normally be detected at the time of occurrence since the specific control features provide early warning. Violation of a safety boundary or safety condition could also be detected after the fact during routine process control review or periodic audit. Failure to have a specific control feature in place should be detected and corrected during the prestartup review or would be detected after the fact. Actions to be taken in response to OSR violations detected at the time of occurrence are delineated in Section 11.1.4.1. Response to violations detected after the fact is defined in Section 11.1.4.2.

11.1.4.1 Violation Detected at Time of Occurrence. When an OSR violation is detected at the time of occurrence, the following action shall be taken.

- The affected process shall be brought to an orderly shutdown; this may require shutdown of the entire facility. If the violation or response is addressed in an emergency procedure, that procedure shall be followed.
- The following people (by position) shall be promptly notified: Manager, PUREX Process Operations; Manager, PUREX Operations; Manager, PUREX Process Control; appropriate Department Manager, Safety and Quality Assurance (S&QA) Chief, Nuclear Processing Branch, DOE-RL.
- Rockwell shall investigate the violation, recommend corrective actions to prevent further violations, and prepare an unusual occurrence report. This report shall be reviewed by the appropriate Rockwell safety committee and the DOE-Richland Operations Office (RL).

- The facility shall not restart until the following have been completed:

  - The facility system, or process has been returned to a safe condition

  - Actions or controls have been established to reduce probability of a reoccurrence.

  - Approval to restart has been obtained from the following: Director, Production Processing; Director, S&QA and DOE-RL Director, Nuclear Materials Production Division.

11.1.4.2 Violation Detected After the Fact. When an OSR violation is detected after the fact, the following action shall be taken.

- An immediate investigation shall be made by a PUREX Process Control Engineer and process supervision to determine the safety of the processing operation and the immediate corrective action to be taken to assure safe conditions.

- The Manager, PUREX Operations Department shall decide, based on the results of the preliminary investigation, whether or not to shut down the affected process or operation.

- The following people (by position) shall be promptly notified: Manager PUREX Process Operations; Manager, PUREX Operations; Manager, PUREX Process Control; appropriate Department Manager, S&QA; Chief, Nuclear Processing Branch, DOE-RL.

- Rockwell shall investigate the violation, recommend corrective actions to prevent further violations, and prepare an unusual occurrence report. This report shall be reviewed by the appropriate Rockwell safety committee and DOE-RL.

- The facility shall not restart (if shutdown is required) or begin the next scheduled processing campaign (if operation is continued) until the following have been completed:

  - Actions or controls have been established to reduce probability of a reoccurrence

  - Approval to resume operation has been obtained from the following: Director, Production Processing; Director, S&QA; and Director, Nuclear Materials Production Division, DOE-RL.

11.1.5 Revisions

Revisions to the OSR document to correct discrepancies, reflect process flowsheet or equipment changes, etc., shall be initiated by the PUREX Process Control Group. Review and approval of revisions to the OSR shall be as follows:

- Bases

  Same level, within Rockwell, as original document

- Safety Boundaries, Safety Conditions, and Control Features

  Same level, including DOE-RL, as original document

- Section 11.1, PUREX Final Safety Analysis Report (FSAR)

  Same level, including DOE-RL, as original document.

11.1.6 Audit Link

An audit link exists between this OSR chapter, the accident safety analysis (Chapter 9.0) and the process control documents. The link within this FSAR is given in Table 9-3. The link between the OSR and process control documents is provided by an addendum to RHO-RE-MA-5, PUREX Process Control Manual (PCM).[2] The PCM addendum contains each safety boundary and safety condition as presented in the OSR, followed by Control Feature Requirements (CFR) that implement the specific control features. The CFR contains the LCS and LCO which provide more conservative operating limits, minimum equipment operability requirements, repair schedules, and operability testing/calibration schedules. Revisions to the CFR to correct discrepancies, reflect process flowsheet or equipment changes, etc., shall be initiated by the PUREX Process Control Group. Review and approval of CFR revisions shall be at the same level, within Rockwell, as for the original PCM document. The S&QA approval authority for both the original PCM and revisions is the Safety Review Coordinator.

11.2 SAFETY BOUNDARIES

The safety boundaries apply to those safety related process variables and conditions which are observable and controllable. The specified safety boundaries represent the maximum (minimum) values for the respective variables where it can be said with confidence that no serious consequences will occur. Operation of the plant or process under conditions outside the scope of the safety boundaries could result in serious hazard to the public, environment, or operating personnel. An infraction of a safety boundary requires the immediate notification of DOE-RL and the prompt, orderly shutdown of the affected portion of the processing operation. Depending upon the affected operation, it may be necessary to shut down all production processing activities.

11.2.1 Iodine-131 Content of Fuel

11.2.1.1 Applicability. This safety boundary applies to any nuclear fuel that may be dissolved for reprocessing at the PUREX Plant.

11.2.1.2 Objective. The objective is to define the maximum ^{131}I contained in one dissolver charge of irradiated fuel.

11.2.1.3 Safety Boundary. The amount of ^{131}I contained in one dissolver charge is limited to 1.3×10^5 Ci.

11.2.1.4 Bases. The ^{131}I quantity stated is that analyzed in the short-cooled fuel accident in Chapter 9.0. The actual quantity normally processed will be much lower as described in Section 11.3.2.4.

11.2.1.5 Control Features. Processing of only authorized material will assure low levels of ^{131}I in the PUREX Plant. The green fuel monitor will permit detection of short-cooled fuel prior to charging in the event administrative controls at both UNC Nuclear Industries and Rockwell break down. Temperature limits on the dissolver offgas (DOG), silver reactor regeneration, and acid absorber operation assure that the minimum decontamination factor (DF) assumed in the accident analysis is maintained. The radiation devices in the offgas system are to detect abnormal ^{131}I releases and to permit prompt corrective action. The testing/calibration program assures the mitigating conditions assumed in the accident analysis are available and functioning. Specific control features follow.

A. Only material authorized in Section 11.3.2 shall be processed.

B. A radiation measuring device (green fuel monitor) shall be located in the PUREX Plant railroad tunnel. The radiation emanating from the fuel will be readily detected and measured as the canisters of fuel are removed from the cask cars and transported to the dissolver.

C. The DOG temperature operating range shall be defined to assure efficient operation of the silver reactor and preclude the formation of silver azide.

D. The silver reactor regeneration frequency shall be defined to maintain an acceptable DF for ^{131}I.

E. Radiation measuring devices shall be installed in the DOG systems to provide early warning of higher than normal ^{131}I releases as the fuel is dissolved.

F. Operability requirements for the 293-A backup facility acid absorbers shall be established to recover nitric acid and attain additional ^{131}I decontamination from the DOG during fuel

dissolution. Operability requirements shall also be established for other equipment necessary to support these specific control features.

G. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

11.2.2 Ammonia Concentration in Offgas Systems

11.2.2.1 Applicability. These safety boundaries apply to the ammonia concentration in the dissolver, E cell, and the ammonia scrubber waste concentrator (E-F11) offgas streams.

11.2.2.2 Objective. The objectives are to prevent the occurrence of an ammonia fire in the listed systems and to limit personnel exposure to ammonia.

11.2.2.3 Safety Boundaries. Safety boundaries follow.

A. The maximum concentration of ammonia in the effluent gas from the dissolver and E cell ammonia scrubbers shall be 14 vol%.

B. The moisture content of the vapor space in TK-E3, -E5, and -F12 shall be maintained above 10 vol% whenever ammoniacal solutions are in the tanks.

C. The discharge of ammonia from the PUREX Plant shall be controlled so as to limit the concentration at ground level at the project boundary to 25 ppm parts of air on a weekly average basis.

11.2.2.4 Bases. Ammonia is flammable in the range of about 15% to 30% ammonia in dry air. Water vapor effectively decreases the range of flammability, so that at about 44°C, the vapor pressure over an aqueous solution is sufficient to render all ammonia and air mixtures nonflammable. At a temperature of 44°C, the moisture content of saturated air is 10 vol%. Ammonia reacts with nitric acid or oxides of nitrogen to form ammonium nitrate, which is flammable and, under extreme conditions, potentially explosive. Ammonia may also react with silver nitrate (present in the offgas iodine removal facilities) to form silver azides, which are flammable and explosive. This reaction is prevented by controlling the DOG temperature as stated in Section 11.2.1.5. The limit in Section 11.2.2.3, item 1, is intended to ensure that the lower flammability limit is not exceeded in those systems where the presence of moisture cannot be assured. In those systems where the presence of moisture can be ensured, the requirement of Section 11.2.2.3, item 2, applies.

The limit specified in Section 11.2.2.3, item 3, for ammonia in the air is the Threshold Limit Value (TLV) established by the American Conference of Governmental and Industrial Hygienists for areas of uncontrolled occupancy.(3)

11.2.2.5 Control Features. Ammonia monitors on the DOG, E cell, and 296-A-24 stacks combined with the established temperature and cooling water flow limits assure ammonia concentrations are maintained below the safety boundaries. Minimum operating temperatures in tanks containing ammoniacal solutions maintain sufficient water vapor to assure nonflammable air-ammonia-water mixtures. Maintaining TK-E5 under vacuum during the reaction of decladding waste with caustic minimizes the amount of ammonia released to the cell and subsequently the amount of ammonium nitrate formed and deposited on the main canyon exhaust filters. Air dilution combined with efficient operation of condenser E-F11 assures that the ammonia concentration at ground level is below the TLV. The operability testing/calibration program assures that the mitigating conditions assumed are functioning. Specific control features follow.

A. The maximum allowable ammonia concentration in the effluent gas from the dissolver and E cell ammonia scrubbers shall be defined to allow corrective actions to control the ammonia concentration within the safety boundary.

B. The maximum temperature of the effluent gas from the downdraft condenser towers during decladding shall be defined to enhance ammonia removal from the gas stream.

C. Minimum water flow rates shall be established for the DOG ammonia scrubbers during decladding and for the E cell ammonia scrubber when ammoniacal solutions exist in E cell and during the chemical adjustment of cladding waste with sodium hydroxide in TK-E5.

D. A minimum operating temperature shall be established for tanks that contain or may contain ammoniacal solutions.

E. Controls shall be established to minimize the transfer of ammoniacal solutions into vessels serviced by the regular vessel vent system.

F. Personnel entry shall be restricted from any onsite location where the ammonia concentration exceeds an established limit, on an 8-h average basis.

G. Minimum equipment operability requirements and repair schedules for the ammonia offgas system blowers, ammonia monitors, and other necessary equipment shall be defined to support these specific control features.

H. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be defined and executed.

11.2.3 Hydrogen Concentration in Offgas Systems

11.2.3.1 Applicability. These specifications apply to the hydrogen content of the DOG system due to the chemical decladding reaction, and in all process vessels due to radiolysis.

11.2.3.2 Objective. The objective is to prevent the occurrence of a fire or explosion due to hydrogen accumulation exceeding the flammability or explosive limits.

11.2.3.3 Safety Boundaries. Safety boundaries follow.

- The maximum concentration of hydrogen in the offgas from a dissolver shall be 6.0 vol%.
- The calculated concentration of hydrogen gas formed by solution radiolysis shall not exceed 4.0 vol% in all other process vessels.

11.2.3.4 Bases. The hydrogen generated in the PUREX Plant during fuel element decladding and by radiolysis of hydrogen-bearing material is controlled to prevent a hydrogen fire or explosion. The lower flammability limit of hydrogen in dry air is 4.1 vol% for upward flame propagation. The flame produced at this concentration is erratic and barely self-sustaining. The limit for horizontal flame propagation is 6.0 vol% hydrogen. Ignition (presumably an open flame would be required at this concentration to raise the local temperature to the 350°C ignition point) of a 6.0 vol% hydrogen-air mixture in a room would result in window breakage and personnel burns.(4) A significantly higher concentration (>16 vol%) would be required for an explosion to occur.(5,6)

At concentrations of <10 vol% in air, less than one-half of the hydrogen is burned.(4) At 6 vol% hydrogen, there is a near zero probability of ignition in the DOG system (9.3 vol% hydrogen in dry air could not be ignited with a spark plug source).(5) If ignition did occur in some unforeseen manner, damage to the DOG equipment would be minimal (maximum calculated pressure increase is 22 lb/in.^{2} g) and there would be no danger to personnel as the equipment is located in a remote shielded area.

An additional safety margin is provided in the DOG system by maintaining a LCS of 4.0 vol% hydrogen and stopping the reaction by adding drown tank water to the dissolver if the uncorrected reading reaches 5.0 vol%. The readings are normally biased high by 0.2 to 0.4 vol% hydrogen due to the presence of water vapor in the sample. Also, under dissolver decladding conditions at a temperature of >86°C, all hydrogen-air-water vapor (saturated) mixtures are nonflammable.(7)

The more conservative 4.0 vol% safety boundary is applied to other process vessels to allow for uncertainties in the hydrogen generation calculations.

11.2.3.5 Control Features. The concentration of hydrogen in the offgas from the decladding step is measured. The concentration is controlled by adjusting the rate of the cladding dissolution reaction and diluted, if necessary, by the in-bleed of cell air to the DOG system.

Hydrogen is a by-product of the reaction between Zircaloy and ammonium fluoride. In the presence of nitrate ion the hydrogen reacts to form ammonia. Hydrogen in other PUREX Plant vessels is formed by radiolysis of water. The concentration is readily diluted by air leakage into the tanks and discharged to the various vent systems. Specific control features follow.

A. The maximum allowable measured hydrogen concentration in the DOG system shall be defined to allow corrective actions to control the hydrogen within the safety boundary.

B. A minimum nitrate ion concentration shall be defined for the decladding solution at the start of the decladding step to control the maximum hydrogen generation rate.

C. A maximum hydrogen concentration in the atmosphere of all process vessels except those vented by the DOG systems shall be defined.

D. Minimum operability requirements shall be defined for equipment necessary to support these specific control features.

E. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

11.2.4 Fuel Receipt and Handling

11.2.4.1 Applicability. The safety boundary applies to the receiving, handling, and charging of nuclear fuel to the dissolvers.

11.2.4.2 Objective. The objective is to minimize the probability of spontaneous ignition of fuel during the fuel handling and charging operations.

11.2.4.3 Safety Boundary. The temperature of any portion of a fuel element shall not exceed 300°C.

11.2.4.4 Bases. Exposure of irradiated fuel elements to the atmosphere for excessive periods could result in overheating and subsequent ignition. Both uranium and zirconium are pyrophoric. The uranium fuel element ignition temperature was estimated to be 300°C in the Chapter 9.0 accident analysis.

Calculations show that the 150 d cooled fuel would self-heat to 446°F (230°C) in 15 h in the absence of a mechanism for heat removal. This would characterize a filled double-barrel canister with a total of 14 fuel

elements. The canister is assumed to contain 12% ^{240}Pu fuel and to be on its side within the canyon.[8] Combustion would not occur at this temperature even if the cladding is breached.

11.2.4.5 Control Feature. The control features assure the fuel element temperature is maintained within conservatively safe limits at all times. Specific control features follow.

A. In the event irradiated fuel is held in the cask longer than the time recommended by PUREX Process Control on the shipping papers, control shall be established to ensure adequate cask cooling.

B. Controls shall be established to ensure that fuel in the dissolver is adequately cooled to prevent ignition of uranium or zirconium.

C. The requirements shall be defined for developing a recovery plan as well as restriction for fuel movement following the dropping or spilling of a container of irradiated nuclear fuel.

D. Minimum equipment operability requirements and repair schedules shall be defined to support dissolver charging operations.

E. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

11.2.5 Flammable Solvent Vapors and Aerosols

11.2.5.1 Applicability. This safety boundary applies to the prevention of solvent fires in all process vessels in the solvent extraction (including concentrators/condensers) and solvent waste treatment systems (F, G, H, J, K, L, and R cells).

11.2.5.2 Objective. The objective is to prevent the occurrence of an organic solvent fire or explosion in a process vessel.

11.2.5.3 Safety Boundary. The concentration of organic in vapors and/or aerosols containing <33 vol% water vapor shall not exceed 46 mg/L.

11.2.5.4 Bases. The PUREX process solvent consists of materials with predictable vapor pressure/temperature relationships. The safety boundary organic concentration in vapors and/or aerosols in dry air of 46 mg/L is the established lower flammability limit for hydrocarbons of chain length C_8 or greater.[4] This limit therefore applies to normal paraffin hydrocarbon (NPH). The similar limit for pure tributylphosphate (TBP) is 88 mg/L and is not reached until the temperature in an unagitated vessel reaches 146°C.[9]

The 46 mg/L concentration due to the vapor pressure contribution of NPH is reached at a minimum temperature of 80°C. Calculations based on conservative assumptions for available vapor space, agitator aerosol generation rates, and aerosol settling rates are made to determine the maximum aerosol contribution towards the vapor phase organic concentration.(10-12) A temperature LCS based on the worst-case tank is then selected to assure the 46 mg/L value is not exceeded in any tank.

Very fine aerosols or "mists" with particles in the size range of 10 to 60 µm may be flammable at organic concentrations of less than 46 mg/L and at temperatures from well below the flashpoint (80°C minimum for NPH) to 90°C. At 90°C, mist flammability is limited by lack of sufficient oxygen in the mixture. Engineering calculations have shown that NPH particles in aerosols which may be formed as a result of agitation in PUREX vessels would have a mean diameter of greater than 300 microns.(10) Thus, the 46 mg/L is applicable when both vapors and aerosols are present.

For wet air systems, such as in a concentrator/condenser or the discharge from a steam jet, flammability is limited by the water vapor concentration. For kerosene-type hydrocarbons, gaseous mixtures containing greater than 33 vol% water vapor are not flammable under any conditions, When water is present as a separate phase, dilution to 33% water vapor is reached at 72°C. At this temperature, the vapor pressure contribution of pure NPH is well below the 46 mg/L flammable concentration.

11.2.5.5 Control Features. The concentration of organic in the vapor phase is primarily controlled by establishing upper temperature limits for the liquid phase, controls on the volatility of the organic diluent (NPH) purchased for use, and operational controls which limit formation of aerosols. In agitated tanks, which contain organic, a lower allowable operating temperature is established to ensure that the sum of the concentration contributed by vapor pressure and by any aerosol formed by a partially submerged agitator is less than the concentration allowed by the safety boundary. The use of steam jets for transferring organic is avoided whenever possible or controlled to prevent gasing which could result in exceeding the temperature limits near the jet suction leg. Controls are also established over agitator operation in tanks containing organic to avoid spark generation. Solvent spills and leakage from a pressurized system are limited to avoid aerosol formation and to assure that airflow through the ventilation system filters would not be significantly reduced by soot resulting from a potential fire. The equipment operability requirements and the operability testing/calibration program assure that the mitigating conditions (e.g., aqueous film-forming fire protection system) assumed in the accident analysis are functioning. Specific control features follow.

A. Any potentially organic-containing process stream fed to an evaporator (or other vessel) in which solutions are heated above an established temperature shall be steam stripped or decanted.

B. Vessels that contain or have contained solutions that have contacted an organic solvent and have not been steam stripped or decanted shall be operated above the established temperature only under special conditions and procedures reviewed and approved by the Director, Production Processing.

C. Maximum temperatures shall be established for all aqueous feed, scrub, and stripping streams to the solvent extraction columns and for all vessels containing organic solvent during both normal and flushing operations.

D. The potential formation of large quantities of organic aerosol shall be avoided by using pumps for transferring organic wherever practicable.

E. Special precautions shall be taken to avoid gasing of steam jets used to transfer organic.

F. Interlocks shall be provided to prevent operation of an uncovered bottom agitator blade in tanks containing solvent to minimize the possibility of agitator/shaft failure and the resulting spark source.

G. Temperatures potentially resulting in vapor phase organic concentrations exceeding the safety boundary of 46 mg/L from both vapor pressure and aerosol contributions shall be established for each tank.

H. A temperature limiting control setting shall be established based on the worst case tank identified in item G.

I. Controls shall be established to limit the volatility of NPH purchased for use (minimum flashpoint).[13]

J. Air jets shall be used for sampling PUREX tanks that normally contain a separate organic phase.

K. Controls shall be established to limit organic solvent spills or leaks from a pressurized part of the system.

L. Minimum equipment operability requirements and repair schedules for the temperature measuring systems, temperature-steam (or cooling water) interlocks, agitator interlocks, and other equipment to support solvent extraction operations shall be defined.

M. Operability testing/calibration schedules for equipment and instruments required to support these specific features shall be established and executed.

11.2.6 Concentrator Temperatures

11.2.6.1 Applicability. These safety boundaries apply to the following process vessels:

- 1WW concentrator, E-F6-1
- 3BP stripper, T-L6
- 3WB concentrator, E-H4-1
- 3BP concentrator, E-L7-1
- 1CU concentrator, E-J8-1
- Uranium rework tank, TK-P6 1
- 2EU concentrator, E-K4-1
- Waste hydrolysis tank, TK-F8.

11.2.6.2 Objective. The objective is to prevent the formation and potential violent decomposition of nitrated organic compounds.

11.2.6.3 Safety Boundaries. Safety boundaries for concentrator temperatures follow.

- The maximum solution temperature in vessels E-H4-1, E-J8-1, E-K4-1, TK-F8, T-L6, E-L7-1 and TK-P6 shall be 135°C.
- The maximum solution temperature in concentrator E-F6-1 shall be 140°C.

11.2.6.4 Bases. Vigorous exothermic reactions can occur between nitric acid or heavy metal nitrates and some TBP diluent mixtures under conditions that might arise during the concentration of waste and product solutions. Violent reactions resulting in equipment damage were experienced during early developmental work and were verified in laboratory studies.[14-16] More recent laboratory work in which various diluents were studied has extended the knowledge of the nature of solvent nitration reactions.[17] While the exact reactions have not been determined, certain characteristics have been observed and the conditions under which these can be expected to occur have been defined as follows.

- The explosions or rapid reactions have involved the evolution of oxides of nitrogen, and the two incidents occurred only after concentration had proceeded to the point of incipient calcination of the uranyl nitrate (greater than 135°C).
- 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 and organic nitrate compounds.

- In the laboratory studies, the temperature at which the reactions were initiated (as evidenced by the evolution of oxides of nitrogen) varied, depending on the nature of the diluent and whether or not heavy-metal nitrates were present. With a naphthenic-type diluent (such as was used during the first 10 yr PUREX operated), rapid reactions have occurred at temperatures of 135°C in the presence of heavy-metal nitrates and 150°C in the absence of heavy-metal nitrates. In more recent laboratory studies straight-chain-saturated or NPH-type diluents (such as the NPH diluent used during the last 7 yr of plant operation), no vigorous, exothermic reactions were encountered at temperatures up to 150°C in the presence of heavy-metal nitrates.[17] These later studies did not investigate the solvent nitration reaction in the absence of heavy-metal nitrates. However, it is reasonable to assume that the relationship reported earlier (i.e., higher initiation temperature in the absence of heavy-metal nitrates) would be equally valid with normal paraffin-type diluents.[16]

11.2.6.5 Control Features. The control features are designed to preclude violent solvent-nitrate reactions by preventing solvent from entering the concentrators and maintaining an always-safe temperature. Decanting or steam stripping prevents solvent entry and the interlocks assure a safe temperature. The use of the straight chain NPH diluent provides an additional margin of safety. The equipment operability requirements and the operability/calibration program assure the required systems are functioning during operation to provide the mitigating conditions assumed in the accident analysis. Specific control features follow.

A. Maximum allowable steam pressures in the reboilers which are consistent with the temperature limits shall be established.

B. Interlocks shall be provided to shut off the steam upon high steam pressure or high temperature readings.

C. The feed to concentrators shall be decanted or steam stripped (minimum weight ratio of steam to liquid shall be specified).

D. Steam stripping shall be initiated before addition of process streams to a concentrator.

E. Decanting shall be performed from unagitated tanks with a specified minimum pump submergence.

F. Minimum equipment operability requirements and repair schedules for the concentrator control systems and other necessary equipment shall be defined to support concentrator operations.

G. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

11.2.7 Waste Treatment--Sugar Denitration

11.2.7.1 Applicability. These safety boundaries apply to the sugar denitration of high-level waste.

11.2.7.2 Objective. The objective is to prevent entry of radioactive solution into the pipe and operating (P&O) gallery lines which may result in excessive radiation exposure to personnel and/or contamination spread.

11.2.7.3 Safety Boundary. The maximum pressure shall not exceed 6.5 lb/in.^{2} g in E-F6 and 8.2 lb/in.^{2} g in TK-F15 and TK-F16.

11.2.7.4 Bases. Sugar is added to the 1WF solution in the concentrator feed tank (TK-F7) to form nitrite ion in the concentrator and suppress ruthenium oxidation and volatilization. The primary safety concern with this operation is the possibility of uncontrolled pressurization as a result of pumping an excessive quantity of unreacted sugar-nitric acid into a cold concentrator, then heating to boiling temperature. Similar reactions could occur in TK-F15 and TK-F16 where sugar is added to denitrate the waste solution. The approximate pressure that will force solution with a 1.0 specific gravity from these vessels into the P&O gallery is given as the safety boundary. The actual solution specific gravity is considerably higher and the vessels are controlled at much lower pressures to avoid plugging of the vent line and/or ejecting solution to the canyon floor. The TK-F15, TK-F16, and E-F6 vessels are equipped with seal pots which limit pressurization to <25 in. water gauge (WG) (<0.9 lb/in.^{2} g).

11.2.7.5 Control Features. The control features are designed to prevent overpressurizing a vessel as the result of a rapid sugar-nitric acid reaction. Use of sugar is restricted to only three canyon vessels. There the reaction temperature, addition rates, and reaction times are limited to maintain a vigorous, but complete and controlled reaction. These vessels are routinely operated under vacuum. Minimum equipment operability requirements and the operability testing/calibration program assure the needed equipment is functioning during operation. Specific control features follow.

A. Solutions containing sugar or any other chemical denitration agent shall not be added to process vessels other than TK-F7, -F15, and -F16.

B. The maximum rate of sugar addition and the minimum HAW flow to the 1WF Tank (TK-F7) shall be established.

C. Minimum operating temperatures shall be established for adding waste solution containing sugar to E-F6 and during sugar addition/digestion in TK-F15 (TK-F16).[18]

D. Interlocks shall be provided to shut off the sugar at or below the established minimum TK-F15 (TK-F16) operating temperature.

E. Maximum pressure limits shall be established for E-F6 and TK-F15 (or TK-F16) which shall not be operated routinely at pressurization with respect to the process cell.

F. The minimum digestion period for waste solution containing sugar, including the time of sugar addition, shall be established.

G. Minimum equipment operability requirements and repair schedules for the temperature, pressure, and flow measurement systems and other necessary equipment shall be defined to support sugar denitration operations.

H. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

11.2.8 Waste Treatment Operations

11.2.8.1 Applicability. These safety boundaries apply to the operation of the waste treatment and neutralization operations.

11.2.8.2 Objective. The objective is to maintain the integrity of underground waste transfer lines and storage tanks.

11.2.8.3 Safety Boundaries. Safety boundaries for the waste treatment operations follow.

- Solutions (excluding water flushes) discharged directly to mild steel storage tanks or through mild steel lines shall have a pH greater than 8.0 before being transferred from the plant.

- Underground transfer lines 4001, 4002, 4003, or 4004 shall not be subjected to an average temperature exceeding 66°C.

11.2.8.4 Bases. Containment of the radionuclides in the aqueous waste streams requires that the integrity of the transfer lines and storage tanks be maintained. The underground storage tanks are fabricated of mild steel which is not resistant to nitric acid. Except for the short section of carbon steel pipe from the valve pits to the tank, the transfer line from the PUREX Plant to the underground storage tank is stainless steel, and sufficient highly basic solution is in the tank to neutralize any waste discharged from the plant. Because of inadequate mixing in the large, relatively quiescent tank; however, the absence of pockets of acid cannot be

assured unless the pH of the waste (excluding water flushes) is greater than 8.0 prior to transfer. Also, the equilibrium nitrite concentration in the stored waste solutions is enough that the addition of acidic solution to stored waste would evolve copius quantities of oxides of nitrogen gases. This gas would be partially converted to nitric acid in the mild steel tank farm ventilation system, causing excessive corrosion. In practice, the aqueous waste to be transferred to the underground storage tanks is adjusted to a pH of greater than 12.0. Excessive heating of the underground transfer lines by contact with the low pressure superheated steam remaining from the 90 lb/in.^{2} g steam supplied to operate the transfer jets subjects the lines to undue stress and could result in line failure (one instance of line failure was attributed to this phenomenon). The 66°C limit allows normal transfers and short-term heating, as can result when a transfer is slow to start, yet protects the line from excessive heating. The lines are designed to withstand a maximum operating temperature of 93°C.

11.2.8.5 Control Features. The control features are designed to assure that the integrity of the waste transfer lines and tank farm receiving tanks is maintained. The weight factor of the sending tank will be observed closely during waste transfers to minimize operation of the steam jet without solution flow as evidenced by no weight factor decrease within a short time after turning on the jet, leveling of the WF during the transfer, or continued operation of the jet after the WF has "zeroed," Interlocks are installed in some cases to shut off the steam to the jet at an established minimum WF. The minimum operability requirements and the operability testing/calibration program assure the needed equipment is functioning during operation. Specific control features follow.

A. Prior to transfer to underground waste storage tanks, the waste sampling and chemical adjustment requirements to meet tank farm needs shall be defined and implemented.

B. A means for measuring temperatures in the 4000 series lines shall be provided.

C. Minimum equipment operability and repair schedules for the WF instruments in the sending tank and other necessary equipment shall be defined to support waste transfer operations.

D. The use of WF instrumentation to minimize operation of the transfer jets without solution flow shall be defined.

E. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

11.2.9 Plutonium Conversion and Oxide Powder Handling

11.2.9.1 Applicability. These safety boundaries apply to the processing operations involved in the conversion of plutonium nitrate to plutonium dioxide product and to the handling of the product.

11.2.9.2 Objectives. The objectives are to prevent the spread of contamination and minimize exposure of personnel to radiation during the plutonium conversion and product handling operation.

11.2.9.3 Safety Boundaries. Safety boundaries for plutonium conversion and oxide powder handling follow.

- The plutonium nitrate feed solution to the oxide production facility shall be free of separate organic phase.
- The prereduction tank (TK-N5) shall not be operated at greater than 1.0 lb/in.^{2} g pressure
- Plutonium compounds known to be capable of gas generation which could cause pressures in excess of 40 lb/in.^{2} g in a sealed container during storage shall be stored in a vented configuration.

11.2.9.4 Bases. Because of the high operating temperatures, a slight possibility exists of a fire or explosion occurring in the calciners if a significant quantity of organic should be admitted. This organic could either flash vaporize and, possibly, ignite or be nitrated to red oil in the calciner and explode. While the probability of a significant quantity of organic passing through the final plutonium concentrator is low, it could happen as a result of flooding of the 3B column.

Pressurization of any of the process vessels in the gloveboxes in conjunction with a defective instrument air line could feasibly result in the discharge of plutonium solution into the operating area. The vessel most likely to pressurize is the prereduction tank. The pressure limit specified in Section 11.2.9.3 is to prevent sufficient pressurization to cause the above contamination. In practice, the vessels are expected to operate under a slight vacuum at all times.

The probability of container failure and the magnitude of the resultant controlled contamination release is increased by container pressurization. A typical container (food pack can) ruptures at approximately 95 lb/in.^{2} g.[19] The 40 lb/in.^{2} g limit allows a sufficient margin of safety for sealed containers. Container venting eliminates pressure buildup.

11.2.9.5 Control Features. Plutonium dioxide powder is hygroscopic and readily adsorbs sufficient moisture from the air to cause container pressurization. The dry product powder is exposed to the glovebox atmosphere in the powder handling and the loadout gloveboxes. To ensure that the air in those gloveboxes is sufficiently dry to prevent the adsorption of moisture by the product, the atmospheric moisture is limited.

The rate of the prereduction reaction and therefore the rate of gas generation can be controlled by placing restrictions on nitric acid and hydrogen peroxide concentrations and reaction temperature. This will limit pressurization of the prereduction tank.

Tests such as "loss on ignition" readily detect the presence of unstable compounds which could result in container pressurization.

The operability/calibration program assures that the mitigation conditions assumed are functioning. Specific control features follow.

A. Prior to further processing in the plutonium oxide production facility, a sample of each plutonium nitrate feed solution batch shall be visually inspected to verify there is no separate organic phase.

B. The maximum allowable moisture content shall be defined for the powder handling and loadout glovebox atmospheres.

C. Operational controls shall be defined to limit pressurization of the prereduction tank, calciners, and other process vessels.

D. Operational controls shall be established to avoid storage of unstable plutonium compounds in sealed containers.

E. Minimum equipment operability requirements and repair schedules shall be established to support these control features.

F. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

11.3 SAFETY CONDITIONS

The safety conditions apply to those technical conditions or restrictions which are essential for safe operation. The safety conditions pertain to those process or operating conditions necessary to preclude the occurrence of a serious incident or to ensure that normal operations can be maintained within the confines of release guides. Operation of the plant or process under conditions outside the scope of the safety conditions could result in hazard to the public, environment, or operating personnel. An infraction of a safety condition, therefore, requires the immediate

notification of DOE-RL and the prompt, orderly shutdown of the affected portion of the processing operation. Depending upon the affected operation, it may be necessary to shut down all production processing activities.

11.3.1 Nuclear Criticality Prevention

11.3.1.1 Applicability. This safety condition applies to all processing and handling of fissile material in the PUREX Plant.

11.3.1.2 Objective. The objective is to prevent the occurrence of a nuclear criticality accident.

11.3.1.3 Safety Condition. A criticality prevention program shall be established to provide the capability for ensuring that the accumulation of fissile material is maintained in a subcritical configuration (i.e., with an acceptable margin of subcriticality). The program shall consist of equipment and system design review, criticality prevention limits and restrictions, configuration control, personnel training, inspections, and independent reviews.

11.3.1.4 Bases. Department of Energy Order 5480.1A, Chapter V[1] requires that fissile material be processed, stored, transferred and handled in a manner that minimizes the possibility of a nuclear criticality accident. Within Rockwell, the criticality safety program is defined in the Nuclear Criticality Safety Standards manual.[20] Each fissile material operation is considered in detail. Credible contingencies that could potentially lead to a criticality accident are identified, evaluated, and documented. Criticality prevention limits are then established and implemented.[21]

- In shielded facilities, criticality prevention is based on the contingency principle that at least two unlikely, independent, and concurrent changes or contingencies must occur before criticality is possible.

- In unshielded facilities, criticality prevention is based on the contingency principle that at least three contingencies must occur before criticality is possible.

- In establishing controls to assure compliance with the contingency criteria, the margin of subcriticality meets one of the following:

  - The k_{eff}* is less than 0.95, including allowances for biases in the calculations (if the method of calculation can be shown to be accurate by comparison to experimental data for a similar system, a higher value may be justified and approved; under no circumstances shall a k_{eff} of 0.98 be exceeded)

*If a Monte Carlo method is used to calculate K_{eff}, the reported value will be the calculated $K_{eff} + 2\sigma$, where σ is the standard deviation.

- The fraction of critical dimension, volume, mass, etc., including an allowance for accuracy, is equal to or less than 0.90

- The evaluated parameter is equal to or less than a corresponding subcritical limit given in American Nuclear Society standards or guides.[20]

- Criticality prevention may be achieved by limiting one or more of the following items:

  - Equipment geometry--shape and dimensional limitations--including allowances for corrosion, tolerances, etc.

  - Fissile material mass, including allowances for measurement accuracy (particularly using NDA methods)

  - Fissile material volume

  - Fissile material form

  - Fissile material concentration or density

  - Fissile material moderation

  - Presence of fixed or soluble neutron absorbers (poisons)

  - Arrangement and spacing of equipment

  - Reflection.

  Where practical, reliance is placed on equipment design in which dimensions are limited, rather than on administrative control. Full advantage may be taken of any nuclear characteristics of the process materials.

- Criticality limits take into account neutron interaction between systems or individual units (e.g., individual fissile material batches, containers, or vessels), unless the systems are isolated.

9411.6123.1136

11.3.1.5 Control Features. The control features assure that all fuel types proposed for processing in the PUREX Plant are analyzed for criticality implications and specifications are established before approval to proceed is given. Controls on the dissolver operation are established to prevent excessive plutonium concentrations, plutonium precipitation, and plutonium polymer formation. Limits on plutonium rework added to TK-E6 maintain the plutonium concentration in the solvent extraction feed within an acceptable range. The sample schedule and instruments are used to monitor process performance and prevent plutonium reflux or undesired accumulation in solvent extraction equipment. The equipment operability requirements and the operability testing/calibration program assure that the mitigating conditions assumed in the analysis of postulated criticality accidents are functioning. Specific control features follow.

A. Analyses shall be performed for materials with specific ^{235}U enrichments and minimum ^{240}Pu isotopic contents prior to processing.

B. The procedures shall require dissolvers to be inspected following dissolution to prevent overcharging.

C. The dissolver sequence selector switch and specific gravity interlocks shall be used to control allowable additions to and transfers from the dissolvers per flowsheet sequence.

D. The nitric acid concentration and plutonium content of the metal solution at the end of the dissolution step shall be controlled based on dissolver specific gravity readings.

E. The amounts, concentrations, and conditions for rework of plutonium via feed makeup tank TK-E6 shall be specified.

F. An approved sample schedule shall be issued which identifies samples required to support the criticality prevention program.

G. Nitric acid monitors, flow monitors, plutonium concentration monitors, and interlocks shall be identified to supplement the sample schedule and provide an additional safety margin.

H. Minimum equipment operability requirements and repair schedules shall be defined to support the criticality prevention program.

I. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

11.3.2 Authorized Materials

11.3.2.1 Applicability. These safety conditions apply to nuclear fuel that may be received for storage or processing in the PUREX Plant and other nuclear materials that may be received, stored, or used, such as laboratory standards and calibration sources.

11.3.2.2 Objective. The objective is to define those nuclear materials for which safe handling and processing conditions have been established in the approved flowsheets and this FSAR.

11.3.2.3 Safety Conditions. Materials scheduled for processing--source, special nuclear, and by-product materials--may be received, subject to the following limitations.

- Uranium and/or plutonium to be processed shall be in a form compatible with approved flowsheets.
- The annual average (per charge) radioactivity of the fuel scheduled for processing at the PUREX Plant shall be no greater than that of fuel irradiated to 2,500 MWD/t in N Reactor and aged 180 d after discharge from the reactor. No single charge shall contain more fission product (FP) radioactivity than that associated with N Reactor fuel irradiated to 3,500 MWD/t and aged 180 d.
- Standards, calibration sources, etc., and other nuclear material shall not exceed limits developed in the FSAR analysis.

11.3.2.4 Bases. Detailed safety analyses performed in this FSAR on those processes covered by an approved flowsheet have shown that operations can be conducted safely. The analysis currently covers only Zircaloy clad fuel and must be amended before aluminum clad fuel is processed.

The FP inventories for the fuels to be reprocessed have been calculated as a function of exposure and cooling time.(17) The average FP inventory permitted has been shown by this FSAR analysis to result in acceptable radiation exposures during reprocessing. The maximum FP inventory permitted in a single charge allows some processing flexibility without a significant increase in radiation exposures. Activity of Activity of ^{131}I in the fuels to be processed will be on the order of 4×10^5 Ci/t of fuel at the time of discharge from the reactor(22) and will have decreased to about 0.1 Ci/t at 180 d. A review of the decontamination performance of the DOG system during the last processing campaign prior to plant shutdown in 1972 (at which time the silver reactors had not been regenerated for at least 48 mo) indicates that fuel cooled 180 d or longer can be processed continuously at the maximum sustained production rate without exceeding the ^{131}I environmental release guide(s), on a weekly average basis, even with an ineffective silver reactor.

The limits specified in Section 11.3.2.3 for other nuclear material are established to ensure administrative control of hazardous materials without unduly restricting the use of those materials for necessary and legitimate purposes.

11.3.2.5 Control Features. The control features assure that the material processed at the PUREX Plant is within the constraints of the FSAR and the Environmental Impact Statement (EIS).[23] The required reviews listed below assure processing of only approved materials and that radiation exposures resulting from use of miscellaneous standards or sources are maintained at an as-low-as-reasonably-achievable (ALARA) level. Specific control features follow.

A. Prior to acceptance for processing at PUREX, the identity of all nuclear fuel received shall be verified in accordance with approved procedures.

B. Plutonium oxide, nitrate, or other reactive scrap to be transferred into the PUREX Plant for processing shall require the prior review and approval of the managers of PUREX Process Control and Criticality Engineering and Analysis.

C. The nuclear material standards, calibration sources, etc., shall be approved for use by the managers of PUREX Process Control, Radiological Engineering, and Criticality Engineering and Analysis.

11.3.3 Gaseous Effluents

11.3.3.1 Applicability. This safety condition applies to the gaseous effluents from the PUREX Plant.

11.3.3.2 Objective. The objective is to ensure the protection of the public, environs, and operating personnel from radioactive materials and oxides of nitrogen released via the PUREX exhaust stacks.

11.3.3.3 Safety Condition. Neither the general public nor onsite personnel shall be exposed to excess (see Bases) airborne radiation or oxides of nitrogen concentrations resulting from PUREX gaseous effluent streams.

11.3.3.4 Bases. The calculated impacts of radionuclides and oxides of nitrogen released in PUREX Plant gaseous effluents during both routine operations and under postulated accident conditions were found to be acceptable in the PUREX EIS and this FSAR. Actual releases must remain inside the limits evaluated[23] which are below the radiation exposure guidelines contained in Reference 24. The PUREX Plant effluent control system is based on sampling and controlling at the point of release per DOE-RL Order 5820.2.[25]

11.3.3.5 Control Features. The control features assure radionuclide and oxides of nitrogen releases remain within the limits evaluated in the EIS and FSAR. The limits established comply with the requirements contained in the PUREX Plant startup criteria and assure that radiation exposures to both onsite and offsite individuals from the PUREX Plant gaseous effluents are well below the level of concern. The action to be taken in response to abnormally high releases is specified in the emergency plan listed as a specific control feature in 11.3.5.5. The routine stack samples and/or monitor readings provide early warning of off-standard conditions and allow for corrective action to be taken well before the safety condition is approached. The equipment operability/calibration requirements assure that the mitigating conditions assumed in the accident analyses are functioning. Specific control features follow.

A. Limits shall be established for the annual average concentrations of radionuclides from PUREX operations measured in samples taken at ground level at the Hanford Site boundary.

B. Limits shall be established for the annual average concentrations of radionuclides from PUREX operations measured in onsite samples taken at ground level.

C. Radioactive effluent releases shall be sampled, monitored as required, analyzed, measured for volume, and reported (total quantities).

D. Performance limits (e.g., vacuum requirements) shall be established for the vessel and condenser vent systems.

E. A program for testing filters shall be established which includes performance limits.

F. A program for limiting and monitoring the oxides of nitrogen released from the PUREX Plant shall be established.

G. Minimum equipment operability requirements and repair schedules for the HVAC systems and gaseous effluent control and monitoring equipment shall be defined to support processing operations.

H. Minimum operability requirements and repair schedules for the fire protection systems shall be defined to support plant operations.

I. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

11.3.4 Liquid Effluents

11.3.4.1 Applicability. This safety condition applies to the liquid effluents from the PUREX Plant, except sanitary waste.

11.3.4.2 Objective. The objective is to ensure the protection of the public and environs from radioactive materials and noxious material in otherwise potable water supplies.

11.3.4.3 Safety Condition. Neither the general public nor onsite personnel shall be exposed to excess radiation or to hazardous concentrations of chemicals resulting from PUREX liquid effluents (see Bases).

11.3.4.4 Bases. The DOE Order 5480.1A, Chapter XI,[24] requires that the radionuclide content in liquid wastes discharges to the uncontrolled environment be ALARA and not exceed the concentration guides of Table II, Attachment XI-1. The effluent cadmium concentration is controlled to maintain the value in the pond and groundwater well below the concentration limits established by federal regulations.[26] The groundwater beneath the Hanford Site is considered to represent the uncontrolled environs, since it is not in an isolated aquifer contained within the site boundaries. The liquid wastes are therefore considered to reach the uncontrolled environment when the groundwater is reached.

Compliance with the requirement of Section 11.3.4.3 is determined by analysis of groundwater samples taken from monitoring wells.

11.3.4.5 Control Features. The control features assure the radionuclide and noxious chemicals in PUREX Plant liquid effluents are maintained within the levels analyzed in the FSAR and EIS. The weekly samples and/or monitor readings provide early warning of off-standard conditions and allow for corrective action to be taken well before the safety condition is approached. The action to be taken in response to abnormally high releases is specified in the Emergency Plan listed as a specific control feature in Section 11.3.5.5. Controls on pH and organic content prevent leaching of radionuclides previously loaded onto the soil column. Minimum equipment operability requirements and the operability testing/calibration program assure the mitigating conditions assumed in the accident analysis are functioning. Specific control features follow.

A. Limits shall be established for weekly and annual average concentrations of radionuclides in aqueous effluent streams from the PUREX Plant.

B. Radioactive liquid effluent releases shall be sampled, monitored as required, analyzed, measured for volume, and reported (total quantities).

C. Capability shall be provided for automatically diverting the chemical sewer, cooling water, and steam condensate streams containing levels of radioactivity exceeding established trip levels of radiation monitoring equipment to a lined retention basin.

D. Limits shall be established for the chemical composition of the process condensate (pH) and chemical sewer (pH and cadmium) streams.

E. Controls to prevent the presence of a separate organic phase in any radioactive liquid effluent stream shall be established.

F. Minimum equipment operability requirements and repair schedules shall be defined to support the liquid effluent disposal program.

G. Operability testing/calibration schedules for equipment and instruments required to support these specific control features shall be established and executed.

9413259.1142

11.3.5 Administrative Control System

11.3.5.1 Applicability. This safety condition applies to the administrative control system required for operation of the PUREX Plant.

11.3.5.2 Objective. The objective is to assure safe operation of the PUREX Plant during both normal operations and under accident conditions.

11.3.5.3 Safety Condition. An administrative control system shall be defined for the PUREX Plant which assures safe operation under both normal and accident conditions.

11.3.5.4 Bases. The safe operation of a processing facility depends upon personnel of various talents, each performing the assigned task in accordance with established procedures, rules, and controls. For the staff to operate the facility successfully, there must be an assignment of responsibility and delegation of authority. To ensure a safety consciousness throughout the organization, safety standards and objectives plus a procedural system for implementation and an education and training program covering the content and application thereof must be provided.

11.3.5.5 Control Features. The specific control features are listed below. The criticality prevention, gaseous effluents, and liquid effluents controls discussed in Sections 11.3.1, 11.3.3, and 11.3.4; and the Accident Prevention Standards Manual[27] are required to assure safe operation of the PUREX Plant. These controls assure that a qualified and trained staff is available at all times and is capable of safely shutting down the plant if necessary to remain within limits evaluated in the EIS and FSAR. Procedures are provided for both normal and emergency shutdown conditions. The mitigating conditions assumed in the accident analyses are assured by the design control and operability testing/calibration requirements. Adequate records and audits are available as proof of safe operation and compliance with this OSR. Specific control features follow.

A. An organization plan shall be established which provides an effective management system for the PUREX Plant and supporting organizations.

B. Minimum qualifications shall be established for operating and support personnel.

C. A minimum shift staffing plan shall be established which assures that the plant can be shut down safely under accident conditions.

D. A documented training program shall be established.

E. Plant operating procedures shall be prepared to ensure operation within the scope of the OSR specified herein.

F. A surveillance, operability testing/calibration program for all equipment and instruments required to support specific control features shall be established and executed.

G. An emergency plan and emergency shutdown procedures shall be established and maintained in appropriate locations.

H. An ALARA program plan shall be established.

I. Records shall be maintained by PUREX Operations Department to provide auditable proof that operations are conducted within the safety boundaries and safety conditions.

J. An internal Rockwell document containing LCS and LCO shall be prepared to implement the control features contained in this OSR.

K. Audits of the conduct of the operations and support activities shall be made.

L. The design or design modifications of any equipment item or system related to environmental, personnel, or process safety shall receive prior engineering and safety review and approval.

M. The requirements for storing and handling chemicals shall be defined.

N. The requirements for disposal of solid waste materials with the potential for containing radionuclides shall be defined.

O. Minimum operability requirements and repair schedules for pressurized air supply systems and essential services (electricity, steam, water) equipment shall be defined.

11.4 REFERENCES

1. DOE, "Safety of Nuclear Facilities," DOE Order 5480.1A, Chapter V, Department of Energy, Washington, D.C. (August 1981).

2. Rockwell, PUREX Process Control Manual, RHO-RE-MA-5, Rockwell Hanford Operations, Richland, Washington (1983).

3. Threshold Limit Valve (TLV) for Chemical Substances-and Physical Agents in the Work Environment with Intended Changes for 1982, American Conference of Governmental and Industrial Hygienists, (Current Version).

4. Limits of Flamability of Gases and Vapors, U.S. Bureau of Mines Bulletin 503, U.S. Government Printing Office, Washington D.C. (1952).

5. L.V. Carlson, R. M. Knight, J. O. Henrie, Flame and Detonation Initiation and Propagation in Various Hydrogen-Air Mixtures, with and without Water Spray, A1-73-29, Rockwell International Atomics International Division, Canoga Park, California (May 12, 1973).

6. F. R. Belles, Detonability and Chemical Kinetics: Prediction of Limits of Detonability of Hydrogen, 7th Symposium on Combustion at London and Oxford, the Combustion Institute, Butterworth Scientific Publications (August 28 to September 3, 1958).

7. J. S. Yeaw and L. Schnidman, The Extinction of Gas Flames by Steam, American Gas Association Twentieth Annual Convention, Atlantic City, N. J. Proceedings, American Gas Association, Inc., 420 Lexington Avenue, New York, N. Y. (October 10 to 13, 1938).

8. Rockwell, Heat Transfer Analysis - Fuel Canisters Stored in PUREX Canyon, SD-RE-TI-034, Rockwell Hanford Operations, Richland, Washington (July 23, 1982).

9. Flammability Characteristics of Combustible Gases and Vapors, Bureau of Mines Bulletin 627, U.S. Department of the Interior, Bureau of Mines, Washington, D.C. (1965).

10. Rockwell, Bases for PUREX Solvent Tank Temperature Limits, SD-RE-TI-103, Rockwell Hanford Operations, Richland, Washington (July 1983).

11. Rockwell, Essential Material Specification EMS-G-00669, Normal Paraffin Hydrocarbon (NPH), Rockwell Hanford Operations, Richland, Washington (May 1983).

12. Richard Dennis, Handbook of Aerosols (1976).

13. Rockwell, Liquid Mixing and Processing in Stirred Tanks, Rockwell Hanford Operations, Richland, Washington (March 1978).

14. TNX-Evaporator Incident, DP-25, E. I. DuPont Co., Savannah River Laboratory (May 1953).

15. GE, Overconcentration in Initial Operation of Uranium Evaporator - 321 Building, HW-28690, General Electric Company, Richland, Washington (July 1953).

16. GE, Investigation of Explosive Characteristics of PUREX Solvent Decomposition (Red Oil), HW-27492, General Electric Company, Richland, Washington (March 1953).

17. Safety Aspects of Solvent Nitration in HTGR Fuel Reprocessing, GA-A14372, UC-77, General Atomic Company (June 1977).

18. GE, Denitration of PUREX Wastes with Sugar, HW-76973, General Electric Company, Richland, Washington (April 1963).

19. Rockwell, Evaluation of Food Pack Cans as Plutonium Storage Containers, ARH-CD-635, Atlantic Richfield Hanford Operations, Richland, Washington (March 1976).

20. Rockwell, Nuclear Criticality Safety Standards, RHO-MA-136, Rockwell Hanford Operations, Richland, Washington (July 1978).

21. Rockwell, Criticality Prevention Specifications--Uranium Plutonium Separations at the PUREX Plant, CPS-P-465-40000, Rockwell Hanford Operations, Richland, Washington (April 1983).

22. UNC, Isotopic Composition of Irradiated Mark IV and Mark 1A Fuel, UNI-436, UNC Nuclear Industries, Richland, Washington (September 29, 1975).

23. DOE, Operation of PUREX and Uranium Oxide Plant Facility, DOE/EIS-0089D, Draft Environmental Impact Statement, U.S. Department of Energy, Washington, D.C. (May 1982).

24. DOE, "Requirements for Radiation Protection" DOE Order 5480.1A, Chapter XI, U.S. Department of Energy, Washington, D.C. (August 1981).

25. DOE-RL, Radioactive Waste Management, DOE-RL Order 5820.2, U.S. Department of Energy - Richland Operations Office, Richland, Washington (July 1979).

26. Code of Federal Regulations, 40-CFR-61.224, U.S. Government Printing Office, Washington, D.C.

27. Rockwell, Accident Prevention Standards Manuals, RHO-MA-221, Rockwell Hanford Operations, Richland, Washington (March 1979).

THIS PAGE INTENTIONALLY
LEFT BLANK

PUREX FSAR

APPENDIX A GLOSSARY OF TERMS

THIS PAGE INTENTIONALLY
LEFT BLANK

This page intentionally left blank.

APPENDIX A

GLOSSARY OF TERMS

The definitions of terms used in the Plutonium-Uranium Reduction Extraction Final Safety Analysis Report (PUREX FSAR) are those used at the Hanford Plant. Where possible, definitions are those that are in general use.

The information is arranged alphabetically, including terms which are constructed by joining more than one word. For example, "airlock" is listed under A and "Lock and Tag Procedure" under L.

A glossary of acronyms used in the PUREX FSAR occurs at the end of this appendix.

9413219.1148

A

<u>absorbed dose</u>: the quantity of energy imparted to a mass of material exposed to radiation expressed in rads (1 rad = 100 ergs/g)

<u>abnormal occurrence</u>: an unwanted flow of energy

<u>absorption</u>: the process by which radiation imparts some or all of its energy to any material through which it passes; the taking up of substance through the interior of another substance; (not to be confused with adsorption which is the adhesion of one substance to the surface of another)

<u>abused</u>: used wrongly or improperly

<u>accident</u>: the detrimental impingement of an unwanted energy flow upon a target as a result of the lack of barriers or barrier failure

<u>accountability</u>: material balance, usually of a valuable material (e.g. plutonium and uranium, see SNM Material) encompassing all significant inventory, incoming, and outgoing amounts of the material

<u>accuracy</u>: conformance with standard; as compared to precision, which is a measure of ability to duplicate or quality of reproducibility

<u>acid</u>: a hydrogen-containing substance which dissociates on solution in water to produce one or more hydrogen ions. Acids turn litmus paper red, react with bases to form salts; solutions of acids have a pH below 7.0

<u>acid absorber</u>: equipment used to recover nitric acid

<u>acid deficient</u>: a term applied to caustic (basic) solutions

acid suit: acid resistant clothing, usually rubber or plastic, worn in conjunction with hood and face shield, gloves and boots, to provide a complete cover to all parts of the body

actinides: the elements with atomic numbers above 88

activity: (1) chemical activity--the relative rate, degree, readiness, or ease with which an element, compound, or agent acts; (2) nuclear activity--the rate at which a material is emitting nuclear radiations, usually given in terms of the number of disintegrations occurring in a given mass per unit of time [the standard unit of nuclear activity is the curie (Ci)]; (3) operational activity--a planned program of work

administrative control: control enforced by management directive, usually in written form, and transmitted to the operating personnel in an appropriate directive instrument such as a procedure or a posting

adsorption: a surface phenomenon in which one substance is taken up at the solid surface of another resulting in a change of concentration at the interface

aeration: charging a liquid with air; bubbling air through a liquid or spraying liquid into the air

aerosol: a dispersion of colloidal (very finely divided) particles in air, e.g., mist, fog

aging: usually refers to decay of radionuclides; also refers to chemical or physical changes brought about by time

air change: the rate of change of ventilation air, assuming bulk displacement. Usually referred to as "air changes per hour"

air lift: device for raising liquids to a higher level by air current

air-lift circulator: a device using the air-lift principle for agitation and/or mixing of liquids

airlock: (1) airlock (ventilation)--a compartment between areas of different pressure level used to provide transportation between areas without upsetting pressure differentials; (2) airlock (pipe) stoppage of liquid flow caused by air trapped in the pipe

air monitor [constant air monitor (CAM)]: air sampler with associated electronic circuitry for the continuous detection of radioactivity collected on the filter paper; alarm circuits are also provided to warn personnel of changes in activity level and instrument malfunctions

air operated valve (AOV): a valve actuated by air for controlling fluid flow

air sample: a sample of air for content analysis. (When measuring radioactivity, a known volume of air is passed through a filter paper and the total activity collected on the surface of the filter paper is measured)

airborne: supported by air. Usually applied to small particles whose free-fall velocity is so slow that they are carried by air currents. Airborne contamination applies to contamination loose in, filtered from, or deposited from the air as contrasted with contamination spread by such means as splashing, dripping, or leaking

alarm limits: prescribed maximum and/or minimum quantities to determine when a warning signal is to be given

alignment: adjustment; arrangement in a line

aliquot: in chemistry, a measured proportion of a total volume of a solution

alkali: see base; a generic term for sodium or potassium hydroxide

alkaline solution: aqueous solution having more OH^- ions than H^- ions; turns red litmus paper blue; pH above 7.0

alpha particle: nucleus of helium atom, emitted by some radioactive elements; each alpha particle consists of two protons and two neutrons, therefore, having a nuclear mass number of four and a charge of plus two

alpha radiation: the emission of alpha particles (helium nuclei) from a material undergoing nuclear transformation by alpha decay

alpha total (AT): total net alpha counts per minute per unit volume

alpha waste: waste material contaminated by radionuclides which emit alpha particles, particularly transuranic elements

analogy: likeness or similarity between two or more things

anhydrous: dry, contains no water

anion: a negative ion

annular tank: a doughnut-shaped tank; the annulus provides favorable geometry for nuclear safety

annunciator: an electrically controlled signal or indicator that audibly or visually (or both) announces an event

antifoam: an agent that prevents foaming or disperses foam

appreciable: considerable

aqueous solutions: solution in which water is the solvent

aqueous makeup: chemical preparation of water solutions; sections of process building used for this purpose

aqueous phase: in solvent extraction, the water-containing layer, as differentiated from the organic phase

aquifer: a formation, a group of formations, or part of a formation which contains sufficient saturated permeable material to yield water to wells or springs

areal concentration: in units of grams per unit area. Mass (usually of plutonium) within a vessel divided by the effective floor area of the vessel

array: an arrangement of objects, such as spacing of tanks, cans, etc., as in nuclear safety

as-built: modifications of the master drawings according to actual plant installation, including dimensional changes, substitution of materials, deletions, etc., so that an accurate record is available for future maintenance and design activity

aspirator: any apparatus for producing a movement of fluids by suction

Atomic Energy Commission (AEC): the government agency established to administer the nation's atomic energy program replaced by ERDA and then by DOE

atomic number: the number of protons in the nucleus of each chemical element

attenuation: a reduction in intensity

automated process: any portion of operations, either chemical or mechanical, that does not require manual operation

auxiliary: equipment used in support of a process or activity; also the small hoist on a crane

available radioactive contamination: the radioactive nuclides which are accessible and can be transported by natural environmental forces

B

backcycle: returning some portion of process solution to a previous step in the process (in PUREX, the portion of the facility involved in aqueous waste recycle)

9413219.1152

background: the amount of radiation present at a given location, due to natural or induced radiation; the ambient level of radiation

background radiation: the level of radioactivity in an area, which is produced by sources other than the one of specific interest; in the Hanford area, the background radiation is produced by naturally occurring radioactive materials in the crust of the earth, cosmic radiations, and the fallout from nuclear weapons tests

backscattering: the deflection of radiation by scattering process through angles greater than 90 with respect to the original direction of the motion

backup: usually emergency equipment or spares readily available, also refers to secondary HNO_3 recovery from PUREX dissolver offgas

baffle: a plate in a piping or duct system used to control or direct the flow of fluids

bag in and out: procedure used to introduce and to remove material from process hoods, without loss of containment, via plastic bag procedure

batch operation: each of the process steps is completed and stopped before proceeding to subsequent steps--as opposed to continuous operation where process streams flow continuously through a series of process steps

batch-size control: a method used to control the amount of fissile material in each batch or vessel to prevent a nuclear criticality

beam, radiation: a stream of radiation, such as a stream of beta particles; the radiation escaping through an opening of a shield

beta decay: radioactive decrease by emission of beta particles

beta particle: an electron or positron emitted from an atomic nucleus or neutron in the process of transformation

beta radiation: essentially weightless charged particles (electrons and positrons) emitted from the nucleus of atoms undergoing nuclear transformation

beta ray: a stream of beta particles

beta radiation: essentially weightless charged particles (electrons and positrons) emitted from the nucleus of atoms undergoing nuclear transformation

beta ray: a stream of beta particles

boron trifluoride BF_3: gas used in ion chambers for detection of a neutron flux by the neutron-alpha reaction; also denotes both the tube and the entire instrument assembly

biological: pertaining to science of living matter

biological hazards: the dangers involved in exposure of living things to radioactive and other toxic material

block valve: normally refers to the valve in a gang valve system which is the last one in line to the jet

Blue Print File (BPF): engineering files containing certified vendor information (CVI file), such as vendor drawings, specifications, prints, tracings, catalogs, etc.

body burden: the amount of radioactive material in the body at the time of interest

boiling point: the temperature at which the saturation vapor pressure of a liquid is in equilibrium with the external pressure on the liquid at standard conditions (BP of water is 100°C or 212°F at 760 mm of Hg)

breakthrough: point at which retention of "product" is no longer maintained; e.g., in ion exchange, a breakthrough occurs when the resin is saturated and permits the product to pass through

breathing air: separate air supply for workers who wear fresh-air masks as respiratory protection

Bremsstrahlung: secondary photon radiation produced by deceleration of charged particles passing through matter

bridge crane: a crane in which a beam or bridge running on overhead tracks carries the hoisting apparatus

bucket: a metal container for storage, transport, and dumping of irradiated aluminum-clad fuel elements at PUREX

burial box: any metal, wood, concrete, or cardboard container specifically designated for containing contaminated wastes or equipment for burial

burial ground: a land area specifically designated to receive packaged contaminated solid waste or worn-out equipment for shallow land burial. Solid waste repositories on the site are of several types:

- Dry waste trench: a linearly oriented excavation which receives solid waste in cardboard boxes, barrels, concrete boxes, fiberglass-impregnated boxes, etc. The trench is backfilled and covered with 8 ft of earth

- Concrete-lined V-trench: a trench which receives barrels of solid waste which are covered by metal and 4 ft of earth

- Transuranic slab: a concrete-pad storage facility containing stacks of 55-gal drums of transuranic dry wastes. The stacks of drums, 24 by 24 by 12 ft, are covered with plywood, plastic-coated nylon sheeting, and 4 ft of earth

- Caissons: vertically oriented cylindrical structures of metal pipe or concrete which are buried below grade for receipt of small items of contaminated solid waste

- Tunnel: oval-shaped steel structures supported by reinforced concrete beams and covered with a soil overburden. A railroad traverses the tunnel, and large pieces of contaminated equipment are placed on railroad flatcars and stored in the tunnels.

9411.6125.1.154

burst: an exceptionally large release of uncontrolled radiation energy is a very short period of time

butt: an adjustment to bring a chemical solution into specifications

button: the mass of plutonium metal formed in the plutonium reduction operation

by-product material: radioactive material produced in a nuclear reactor, ancillary to the reactors main purpose of producing power or fissile materials; fission products are usually considered to be by-product material

C

calcine: to drive off volatile matter and produce dry solids by action of heat; also, the product produced by calcining

calibrate: to determine, check, or correct the graduation of

cam: a device for converting regular rotary motion into irregular rotary or reciprocating motion

CAM: see continuous air monitor

campaign: the time frame and series of events associated with a period of active or continuous operation

canyon: that portion of a chemical separations plant that contains the processing cells and maintenance craneway

capacity factor (CF): the ratio of the actual capacity compared with the design capacity

capillary: a fine bore tube, of any material

car puller: a device, usually consisting of a fixed capstan, drive mechanism and cable, used for moving railroad cars to a specific spot for loading or unloading

cartridge: the assembly or package of stacked perforated plates and spacers used in the solvent extraction columns; a filter element in assault masks (same as canister)

cascade: equipment arrangement where process streams flow by gravity from vessel to vessel, column to column; also, radioactive decay scheme

cask: a shielded container used to transport radioactive material

cask car (well car): a railroad car design to transport casks

cation: a positive ion

caustic: sodium or potassium hydroxide

cave: a shielded hood, usually containing a viewing window or overhead mirror and some sort of manipulators for handling radioactive materials

cell: a confining room or enclosure, often shielded, where radioactive materials are processed or stored

centrifugal pump: a pump for transferring liquids, consisting of an impeller rotating within a casing

centrifuge: rotating equipment using centrifugal force to separate solid-liquid mixtures or liquid-liquid mixtures of different densities; act of using a centrifuge

Cerenkov radiation: electromagnetic radiation emitted by a charged particle moving in a material at a velocity greater than the speed of light in that material

chain reaction: a series of nuclear reactions in which one of the agents necessary to the series is itself produced by the same reactions. Depending on whether the number of reactions directly caused by one reaction is on the average less than, equal to, or greater than unity, the chain reaction is convergent (subcritical), self-contained (critical), or divergent (supercritical)

charging: feeding raw material into an apparatus, e.g., dissolver or furnace

chemical processing: chemical treatment of materials to separate specific usable constituents. At Hanford, the separation by chemical means of plutonium, neptunium, and uranium from each other and fission products resulting from irradiation of the uranium in a nuclear reactor

circulator: a device for introducing air into the bottom of a self-boiling waste storage tank to provide agitation and to minimize thermal stratification

clarified: made clear or intelligible; free of solids

clearance (Security): access permit covering type of information one can receive and areas and buildings one is permitted to enter

coating removal: an operation involving removal of the metal sheath from reactor fuels

codecontamination: removal of fission products and impurities from plutonium and uranium streams simultaneously in solvent extraction; i.e., first solvent extraction cycle at PUREX

cold: uncontaminated; free of radioactivity; not irradiated when referring to uranium

column (packed): extraction column which contains packing (Burl saddles or Raschig rings)

column (pulse): solvent extraction column in which the two liquid phases (organic and aqueous) flow countercurrently to each other and are pulsed to attain good mixing

column frequency: or pulse frequency, is the time rate of pulsing in a column, expressed in cycles per minute

compound (chemical): a substance composed of two or more elements combined in definite proportions by weight, the individual properties of the constituents having disappeared

compound column: a solvent extraction column having more than two influent streams, such as an organic stream, feed stream, and scrub stream, all introduced at different points

compressed: condensed; pressed into less space

concentration: the mass per unit volume or the strength of a solution

concentration guide: the average concentration of a radionuclide in air or water to which a worker or member of the general population may be continuously exposed without exceeding radiation dose standards

9413219.1157

concentrator: equipment used to evaporate solutions to reduce volume and increase concentration

condensate: liquid formed from cooling vapors

condenser: equipment used to condense vapors, usually some form of heat exchanger that removes heat from vapor, causing condensation. The cooling medium is usually water

confinement: to keep radioactive material within some specified bounds, differing from containment in that no absolute physical barrier exists

connector, remote: a device for connecting process piping, auxiliary lines, or electrical conduits by an operation performed remotely

consequence: an effect or result

constant monitoring: continuous determination of a parameter, such as the amount of ionizing radiation or contamination present

contact maintenance: maintenance work involving contact or touch with equipment being repaired, as differentiated from remote maintenance

contactor: apparatus in which batch or continuous solvent extraction operations are performed

containment: that which prevents escape and dispersion of radioactive material beyond specified confines

contaminated waste: solid, gaseous, or liquid waste which contains radioactive materials

contamination (radioactive): the presence of radioactive materials where they are normally undersirable, harmful, or cause interference

contingency, double/triple (as referred to criticality safety): the philosophy that at least two unlikely, independent, and concurrent changes (contingencies) in processing or operating conditions are required before a criticality is possible in shielded areas. For unshielded areas, criticality prevention is based on the occurrence of three such changes (contingencies) before a criticality is possible

continuous air monitor (CAM) - instrument which continuously monitors radioactive levels in the air and alarms at a set point

continuous phase: the larger volume of liquid in a column through which a smaller volume (dispersed phase) travels in the form of droplets or globules

controlled area: an area in which access, occupancy, and working conditions are controlled to minimize exposure of personnel to radiation and contamination, or to maximize security

controlled exposure: the limiting of occupational exposures to ionizing radiation by administrative means

controlled injury zone (CIZ): a zone containing radioactive material, in which employees with improperly protected or healed skin injuries are not permitted to work

convection: natural movement of heated parts of a liquid or gas

cooling (fuel): storage of fuels after discharge from reactors to allow for the decay of short-lived isotopes to acceptable radioactivity levels

corrode: to eat away gradually by chemical action

count: act of measuring radioactivity; the specific amount of radioactivity present

count rate meter (CRM): a device which gives a continuous indication of the average rate of ionizing events

counter: an instrument that records a portion of the disintegrations from which the total can be calculated

counts per minute (cpm): the number of events per minute recorded by an instrument designed to detect radioactive particles; especially used to indicate the relative amount of radioactive contamination

cover block: a removable concrete shield over cell openings and diversion boxes

crane maintenance platform (CMP): canyon maintenance work area behind a shielding wall where direct maintenance on the crane is accomplished

crash alarm: notification of an emergency made simultaneously to telephones on an automatic conference circuit

crib: a porous underground structure for disposal of low-level liquid wastes

critical: fulfilling the condition that a medium capable of sustaining a nuclear chain reaction has an effective multiplication factor equal to unity. (A nuclear reactor is critical when the rate of neutron production, excluding neutron sources whose strengths are not a function of fission rate, is equal to the rate of neutron loss.)

- Delayed critical: identical with critical; the term is used to emphasize that the delayed neutrons are necessary to achieve the critical state

- Prompt critical: fulfilling the condition that a nuclear chain reacting medium is critical utilizing prompt neutrons only

critical mass: the minimum mass of fissile material which can be made critical with a specified geometrical arrangement and material composition

critical organ: the critical organ for internally deposited radionuclides is that organ of the body receiving the specified radioisotope that results in the greatest physiological damage to the body

criticality: state of being critical; a self-sustaining neutron chain reaction in which there is an exact balance between the production of neutrons and the losses of neutrons in the absence of extraneous neutron sources

criticality alarm system: a network of active gamma radiation or neutron-sensitive detectors connected to audible alarms that will signal a personnel evacuation if a criticality occurs

criticality accident, nuclear criticality accident: the release of energy as a result of accidentally producing a self-sustaining or divergent neutron chain reaction

criticality safety: protection against the consequences of an inadvertent nuclear chain reaction, preferably by prevention of the reaction

crud: solids, precipitates, and accumulations of unidentified materials not wanted in the process

curie (Ci): one curie is a unit of radioactivity equal to 3.7×10^{10} disintegrations per second; also equal to 3.7×10^{10} becquerels

cut: portion of metal dissolved at one time from a dissolver charge; a portion of a product material eluted from an ion-exchange column

CVI file: certified vendor information file

cycle: a distinct and complete portion of a process operation, such as a partition cycle or final decontamination cycle in solvent extractions

D

daughter (element): the product of the radioactive decay of a radioactive element (called the parent)

daughter products: the nuclides formed by the radioactive disintegrations of a first nuclide (parent)

de minimus levels: the contamination levels acceptable for unrestricted access or release

deactivation: the permanent removal from active service of a facility or portion of a facility, possibly accompanied by removal or partial removal of equipment or services essential to the operation of the facility or system

decanting: to separate a liquid from a residue or a liquid of higher density by pouring off carefully the supernatant liquid without disturbing the heavier material settled at the bottom; a vacuum or pump transfer is generally made instead of pouring

decay (radioactive): the spontaneous transformation of one nuclide into a different nuclide or into a different energy state of the same nuclide. The process results in a decrease, with time, of the number of the original radioactive atoms in a sample. It involves the emission from the nucleus of alpha particles, beta particles (or electrons), or gamma rays; or the nuclear capture or ejection of orbital electrons; or fission; also called radioactive disintegration

decay chain: the sequence of radioactive disintegrations in succession from one nuclide species to another until a stable daughter is reached

decay heat: the thermal energy produced in a material by its own radioactive disintegrations

decommissioning: the management or disposition of worn out or obsolete nuclear facilities or contaminated sites; at Hanford, four alternatives have been defined for decommissioning contaminated excess sites and facilities

- Layaway: the facility or site is preserved essentially in its current state, as a temporary measure. Continuing surveillance and maintenance are anticipated as long as the facility is in the layaway mode

- Protective storage: the facility is prepared to be left in place safely for an extended period which might range from decades to centuries. Often this mode will require engineered improvements to strengthen the containment of contamination for the duration of protective storage

- Entombment: the facility or site is prepared to be left in place safely, with a long-lasting barrier surrounding all contaminated materials and with no intent to allow for their subsequent retrieval

- Dismantling: as a minimum, all radioactive components and materials which exceed the criteria for unrestricted release are removed from the site. In addition, nonradioactive components may be removed, structures dismantled, and the area prepared for alternative use

decontamination: the selective removal of radioactive material from within another material

decontamination agents: those solutions or other agents used to effect decontamination

decontamination factor (DF): the factor by which the concentration of contaminants is reduced; it is the ratio of the initial contamination to the contamination present at the completion of the process step

deep well pump: turbine-type pump designed to meet a wide range of head and capacity conditions; such pumps usually contain a number of stages to meet the high head requirement

degas: to remove uncombined gases that may be present in a liquid or solid

demineralized water: water which has been treated by ion exchange or material such as zeolites to remove the dissolved salts

demineralizer: equipment used to produce demineralized water; the demineralizing material, such as zeolites

demister: a trap for removing liquid particles (mist) in a gas stream

denitration: the removal of nitric acid, nitro groups, or nitrogen oxides (opposite of nitration)

density: the mass of any substance per unit volume at any definite temperature

depleted uranium: uranium containing less than 0.7111% ^{235}U

deposition: process of depositing or laying down; generally refers to radioactive materials deposited within the body

detrimental: harmful; causing injury

diaphragm: a membrane separating two areas or cavities

differential pressure (DP): pressure difference between two points in an enclosed fluid or gaseous system

diluent: a diluting agent

dip tube or leg: a pipe or combination of pipes extending into a vessel as the suction leg for solution sampling, for transfer of vessel contents, or as a sensing line for instrument functions

disengaging: the separation of contacted phases; e.g., organic and aqueous phases

disintegration: (radioactive) - see decay

dismantling: the disassembling, dismounting, and removal of equipment or systems

dispersed phase: see continuous phase

dispersion: in solvent extraction, the mixing of two immiscible liquids

disposal: safe disposition of radioactive waste, residues, or contaminated equipment, tools, or other materials

dissolve: to make a solution of; to break up or undo

distillation: the process of removing impurities from a liquid by heating it to its boiling point and then condensing the vapors

distribution coefficient (E_a^o ; D): the ratio of the equilibrium concentrations of a given component in two immiscible solutions; e.g., organic/aqueous. A similar term, L/V, simply refers to the ratio of the total aqueous flow to a column to the organic flow to that column

ditch: a linearly oriented excavation often used to divert waste waters to ponds

diversion box: a below-grade concrete enclosure containing the remotely maintained jumpers and spare nozzles for diversion of waste solutions to storage tank farms or disposal sites

Do Not Operate tag: a tag placed on any equipment where a possibility of personnel injury or equipment damage would exist, in the event that the equipment were to be operated

door stop: a shielded transport container for highly radioactive process samples

DOP test: a standard test for determining the efficiency of HEPA filters and consists of introducing a quantity of dioctyl phthalate mist upstream of the filters and determining the fraction that passes through the filters

dose: a general term indicating the amount of energy absorbed from incident radiation by a specified mass; the unit of absorbed dose is a rad

dose commitment: the integrated dose which results from an intake of radioactive material when the dose is evaluated from the beginning of intake to a later time (usually 50 yr); also used for the long-term integrated dose to which people are considered committed because radioactive material has been released to the environment

dose rate: radiation dose received per unit of time

dosimeter: an instrument for measuring accumulated radiation dose; common dosimeters are TLD badges and pencils

double contingency: for shielded areas, the requirement that at least two unlikely, independent, and concurrent changes (contingencies) in processing and/or operating conditions must happen before a criticality accident is possible (referred to as the double contingency principle)

downcomer: a pipe or other vertical flow channel that allows liquid to flow downward

drop-out: method of checking flow measuring instruments in which the volume removed from a tank (drop-out) over a period of time is compared to the measured flow

drum: a metal or composition cylindrical container used for incoming chemicals and for the transportation, storage, and disposal of waste or chemical materials

dunnage: the structure used to support equipment; the crating or supports used to stabilize railroad shipments

E

effluent: a stream leaving a process or facility

emergency power: electrical power available from an alternate station and/or incoming line, in the event of normal power loss. Switchgear located at power substations and in each production facility complete the transfer. Additional backup is provided from steam turbine generators located in each powerhouse

emergency procedure: a special procedure for personnel to follow in the event of an abnormal occurrence or emergency

emitted: discharged, sent forth, given out

emulsion: a dispersion of immiscible liquids; e.g., a suspension of oil in water or water in oil

encased piping: piping enclosed in another pipe or confining enclosure used to doubly enclose process solution containing radioactive material when it is pumped or otherwise transferred in pipes

endothermic reaction: a chemical reaction which absorbs heat

en echelon: adjective to describe geologic features that are in an overlapping or staggered arrangement

Engineering Flow Diagram (EFD): line sketches prepared as a guide for detail design

enriched uranium: uranium containing more than 0.711% ^{235}U

entombment: the act or process of entombing or burying equipment or materials to the extent that their enclosure will remain safe for an indefinite period with minimal required surveillance

entrainment: the carrying of liquid droplets, gas bubbles, or fine solid particles by liquid or gaseous streams

eugeosyncline: a geosyncline in which volcanism is associated with clastic sedimentation

exclusion area: the fenced area enclosing the individual production facilities

excursion: a sudden rapid increase of power produced when a reactor or other system of fissile material undergoes a sudden increase in reactivity

exothermic reaction: chemical reaction which gives off heat

explosive limits: the concentration range of a compound in a defined atmosphere within which an explosive reaction is possible

exposure: total radiation dose received by a person, chemical, or any other material

exposure record: records which date, list, and evaluate the occupational exposure of persons to ionizing radiation

extraction: the mass transfer of an element or compound between two immiscible phases, usually referring to transfer of uranium and/or plutonium from the aqueous to the organic phase

F

Facility Change Notice (FCN): an administrative form used to obtain approval for and recording physical changes to plant and equipment, same as Engineering Order or Engineering Charge Order

fail-safe: design and installation of added safety to processing systems and equipment, so that in case of failure the system will shut down in a safe or stable condition

fast flux (fast neutron): a stream of neutrons having energies (velocities) near that imparted to them by a fission event; when applied to nuclear reactors refers to those using high velocity neutrons to cause successive fission events

9413219.1164

feed/product specifications: detailed requirements such as form, purity, concentration, and limits of impurities

fertile: capable of being transformed into a fissionable substance by capture of a neutron

film badge, thermoluminescent dosimeter (TLD): a badge containing masked photographic film. Nuclear radiation darkens the film and exposure to the wearer can be determined by developing the film and comparing it with a standard.

fissile: capable of being split; fissionable, especially by neutrons of thermal energies

fissile material: one of several actinides which under proper conditions fission spontaneously, producing sufficient neutrons to sustain a chain reaction

fission (nuclear): the splitting of a nucleus into two or more fragments

fission product (FP): a nuclide produced by the fission of a heavy element

fissionable: a nuclide capable of undergoing fission by any process

fissionable material: any material fissionable by neutrons of some energies, such as certain isotopes of uranium and plutonium, that have the property of capturing neutrons and thereupon splitting into two or more particles and releasing additional neutrons

five fold: a personnel contamination detection instrument for monitoring of hands and soles of shoes. The counter uses several gamma and alpha monitoring instruments in a fixed position as detecting elements; now called hand and foot counters

flex jumper: nonrigid connecting pipe used in process cells in place of a rigid jumper

flooding: a solvent extraction column condition in which dispersion of the phases fails and countercurrent flow ceases

flowsheet: an outline or diagram of successive operations specifying stream compositions, ratio of fluid flows, acidities, and densities as optimum for producing product to meet specifications

fluvial: ground formations of, found in, or produced by a river or stream

foldbelt: a linear region that has been subjected to folding and deformation during mountain building

four fold: a personnel alpha contamination detector instrument for monitoring hands

fractionation: internal reflux within a column resulting in a gradual separation between the high and low boiling fractions

fractionization: separation of cesium and strontium from waste streams the B Plant operation

french drain: a subsurface soil drain formed by partly filling a drainage hole in the ground with loose, broken tile, bricks, or crushed rock

fresh air mask: a tight-fitting breathing mask supplied with fresh air from an external source

fuel (nuclear, reactor): fissionable material used as the source of power when placed in a critical arrangement in a nuclear reactor

fuel element: a tube, rod, or other form into which fissionable material is fabricated for use in a reactor

fuel separation (fuel reprocessing): processing of irradiated (spent) nuclear reactor fuel to recover useable materials as separate products; usually separation into plutonium, uranium, and fission products

full-face respirator: a respiratory protective device worn to prevent inhalation of radioactive particles

G

gamma burst: spontaneous disintegration of radioactive substance signifying the sudden release of a large quantity of electromagnetic radiation; a rapid and substantial increase in fission product activity associated with organic product stream from an extraction column. It may be specific to one column, one cycle, or general to the entire solvent extraction train

gamma radiation: electromagnetic energy emitted in the process of a nuclear transition

gamma ray: electromagnetic radiation emitted by the nuclei of radioactive substance during decay, similar in nature of X-rays

gamma scan: process of measuring the energy spectrum of the gamma rays emitted by a material in order to determine its constituent nuclides

gang valve (jet): piping assembly, associated with jet transfers, consisting of four valves performing steam, air, vent, and block service

gaseous waste: stack gas effluents discharged from the production facilities

Geiger-Muller Tube: a gas-filled tube used to detect low-level beta and gamma radiation by the ionization pulse produced in the gas; used on a GM counter

geometrically favorable: subcritical by virtue of neutron leakage, under the worst foreseeable process conditions

geometrically safe: geometrically safe equipment is subcritical by virtue of neutron leakage under all possible conditions

geometry: dimensions and shape, particularly the effect on criticality; favorable geometry indicates shape and size is such that neutrons readily escape system, improving criticality safety; also defines the efficiency of radionuclide counting instruments

geosyncline: a mobile downwarping of the crust of the earth, either elongate or basin-like, measured in scores of kilometers, which is subsiding as sedimentary and volcanic rocks accumulate to a thickness of thousands of meters

Gilmont: ultramicroburrette - the sampler device for obtaining measured volumes of samples for anlaysis and measured drops for specific gravity determinations

glaciofluvial: pertaining to the meltwater from washing glacier ice and to deposits and landforms produced by meltwater

glovebox: an enclosure having openings fitted with gas-tight gloves by means of which certain radioactive or other special materials may be safely handled. The enclosure is kept under slight negative pres-sure. Ventilation, enters when provided from a nonradioactive zone, and is exhausted through high-efficiency filters.

glove port: an opening into a hood or glovebox for attaching an arm-length rubber glove which maintains the hood seal but permits manual hand operations inside the hood

GM instrument: instrument for detecting low-level beta and gamma radiation using a Geiger-Muller tube

grab sample: representative sample obtained by collecting loose solids in equal quantities from each part or package

graphic panel: an operating panel board equipped with recorders and controllers and laid out by means of lines, marks, and characters to give a diagram of process flow

green fuel (green metal, short cooled fuel): reactor fuel elements that have not radioactively decayed the minimum amount of time required for processing

H

half-life: the time required for the radioactivity of a substance to decay to half its original value

hazard: a condition with the potential of causing injury to personnel, damage to equipment or structures, loss of material, or lessening of the ability to perform a prescribed function

hazard review: a detailed, critical assessment of operation, functions, and equipment to determine the hazards present and the degree of risk involved

headend: that portion of the process in which reactor fuels are declad and dissolved and product feed solutions are prepared for extraction battery processing

head tank: a vessel used for direct flow (gravity or pump) of solutions to process

header: common supply or discharge line which leads to or converges from many smaller lines

Health Physics: that branch of radiological physics dealing with the protection of personnel from ionizing radiation

heat exchanger: any of the various pieces of processing equipment used for the controlled heating or cooling of liquids and vapors

heel: the amount left in a vessel or container after the bulk of the contents have been removed

high efficiency particulate air (HEPA) filter: an air filter capable of removing at least 99.97% of the radioactive particulate material greater than 0.3 in diameter in an air stream

high-level alarm: an alarm activated by instruments indicating that an established upper operating limit of some physical variable of the contents of a process vessel, sump, or column has been reached, such as the weight factor, interface, differential pressure

high-level liquid waste: liquid materials from Hanford operations stored in underground tanks; the liquid wastes are contaminated by greater than 100 μCi/mL of mixed fission products or more than 2 μCi/mL of ^{137}Cs, ^{90}Sr, or long-lived alpha emitters

high-level waste: radioactive waste which contains the bulk (99%) of fission products

hold up: the amount of material in plant equipment at any time

hood: a canopy and exhaust duct used to confine hazardous materials in order to reduce the exposure of industrial workers

"hot": a colloquial term meaning radioactive

Hot Shop: a shop or room designed for handling or repair of radioactively contaminated equipment; the space may have special shielding, ventilation, or manipulating equipment

I

immiscible: incapable of being mixed to a state of homogeneity

impact wrench: a power-driven wrench delivering torque through successive sudden applications of radial force to the wrench socket

inaccessible: not capable of being accessible, reached, or approached without resort to deliberate or determined action

incident: unwanted flow of energy

incredible: unbelievable; not reasonable

industrial (radioactive) waste: radioactive wastes which would be sent to low-level radioactive waste burial grounds, e.g., pumps, pipes, crates, concrete, paper towels, etc.

ingestion: refers to being taken into the stomach, especially taking radioactive materials into the digestive tract

inhibitor: a restrainer; a material which stops or retards a chemical or nuclear reaction

in-line instrumentation: instruments designed to continuously measure physical or chemical properties of fluids flowing through pipes

Instrument Flow Diagram (IFD): a line diagram of the instrumentation associated with process equipment and flows

interaction: in nuclear safety, the effect of neutron absorption between two or more containers or masses of fissionable material

interface: the common boundary of two materials either immiscible or of different phases, such as organic and aqueous phases in extraction column operation

interlock: a device to prevent activation of equipment until desired conditions have been met or to make the operation of a piece of equipment dependent on that of another to prevent operations hazardous to personnel or equipment

ion exchange: process for selectively removing a constituent from a liquid stream by reversibly transferring ions between an insoluble solid and the liquid stream; the exchange medium (usually a column of resin or soil) can then be washed to collect the separated constituent if so desired

ionization chamber: a device consisting of an enclosed volume of gas between charged electrodes, used in measuring ionizing radiation in terms of electric charge associated with ions produced in the volume

ionizing radiation: any electromagnetic or particulate radiation (contamination) capable of producing ions, directly or indirectly, in its passage through matter

irradiation: exposure to radiation by being placed near a radioactive source; usually in the case of fuel materials being placed in an operating nuclear reactor

isotope: nuclides with the same atomic number (i.e., the same chemical element), but with different atomic masses. Although chemical properties are the same, radioactive and nuclear properties may be quite different for each isotope of an element

J

jacket: an outer casing of a tank through which water or steam may be passed to heat or cool tank contents

jet: a solution transfer device; also called a siphon, exhauster, or eductor

job hazard breakdown: a written analysis of the job and stepwise procedure that points out necessary equipment and methods to perform the job safely

joint windspeed: joint frequency of occurrence of wind speed, direction, and stability

JPA: Job Performance Aid; detailed written instructions for performing a task

jumper: pipe assembly designed for remote installation and removal, used to connect in-cell canyon equipment

K

k_{eff}: the ratio of the total number of neutrons produced during a time interval (excluding neutrons produced by sources whose strengths are not a function of fission rate) to the total number of neutrons lost by absorption and leakage during the same interval. When the quantity is evaluated for an infinite medium or for an infinite repeating lattice, it is referred to as the infinite multiplication factor (k_∞), and when the quantity is evaluated for a finite medium, it is referred to as the effective multiplication factor (k_{eff})

kilogram: a weight in the metric system equal to 1,000 g or approximately 2.2 lb

knock-out pot: a deentrainment vessel used to separate liquid droplets from vapor

L

labyrinth: as used in this safety analysis report, a scheme or figure depicting the path through the various spatial and physical boundaries between the process solutions in PUREX Plant and the public

lacustrine: anything found or formed or having to do with a lake or lakes

lead glass: glass containing lead oxide in amounts up to 50%, used because of its increased density and consequently its radioactivity shielding properties

let-down valve: an air-operated diaphragm valve, which is in the drain piping of extraction columns and is controlled by the column interface to maintain the desired liquid level in the column

limited area: security area in which guards or other internal controls can prevent access by unauthorized persons to classified matter

limits (operating): volume, concentration, weight limits, etc. in process vessels or areas designed to preclude personnel hazards, equipment damage, or processing upsets

Lock and Tag Procedure: a procedure for locking out and tagging equipment where the possibility of injury exists in the event the equipment is used or operated

long-lived isotope: a radioactive nuclide which decays at such a slow rate that a quantity of it will exist for an extended period; usually radionuclides whose half-life is greater than 3 yr

9413219.1172

lost time injury: colloquial for disabling injury, which is defined as a work injury which results in death, permanent total disability, permanent partial disability, or temporary total disability as defined under OSHA

louver plate: an insert plate fitted with slatted openings designed to pass air across a barrier while deflecting impinging liquids or solids to redirect phases in a pulse column to minimize channeling

low-level alarm: an instrument device set to alarm at a predetermined minimum level

low-level liquid waste: fluid materials disposed of at Hanford which are contaminated by less than 5×10^{-5} µCi/mL of mixed fission products; wastes containing types and concentrations of radioactivity such that shielding to reduce personnel exposure is not required

lull: a temporary quiet or still condition

L/V: the ratio of aqueous to organic flows in extraction column operation

M

magenta: the color, purplish shade of red, of the symbol on a radiation zone sign

man-rem: used as a unit of population dose; often the average dose per individual expressed in rems times the population affected

manipulator: mechanical device for performing work behind a barrier or in a cave

manometer: one type of instrument for measuring pressure

material(s), nuclear: a generic term used to refer to those materials of interest to the nuclear industry, usually synonymous with Special Nuclear Material, but could also include zirconium, beryllium, and other materials of importance in nuclear reactors

material balance: the comparison of input and output material quantities for a process. Generally, the comparison over a specific time interval of beginning inventory plus receipts with ending inventory plus shipments plus discards.

9403219.1173

maximum permissible concentration (MPC): the quantity of radioactive material in air, water, etc. per unit volume or weight, from which a human should not (based on present knowledge) sustain appreciable body damage

maximum permissible limit (MPL): the largest dose of radiation or body burden of radionuclides which, under any combination of circumstances, is not expected to cause appreciable body injury to the average individual at any time during his lifetime

medical treatment injury: a work injury which requires medical treatment, but does not involve a disabling condition or loss of time other than required for medical treatment

Megawatt Day (MWd): usually used as MWd/t, which expressed the exposure in a reactor of a ton or uranium in terms of the energy produced (megawatt - MW) for a specified length of time (day) for a specified amount of material (tonnes - t)

metal: generic term referring to uranium metal and uranium-bearing solutions

metal ("spike" or spike metal): a fuel element whose outer cylinder is enriched to 1.25% ^{235}U, whose inner cylinder is enriched to 0.947% ^{255}U, yielding an average preirradiation enrichment of 1.15% ^{235}U

metal (uranium): generic term for any uranium metal processed in PUREX Plant

Metal (72): see natural uranium

Metal (94 or 95): uranium enriched in ^{235}U to a concentration of 0.947% (used interchangeably)

metathesis: reaction of two compounds involving the displacement and replacement of two elements, molecules, or radicals, and resulting in the formation of two new compounds for example; at PUREX $UF_4(s)$ + $KOH(L)$ $UO_2.2H_2O(s)$ + $KF_4(L)$ (s = solid, L = liquid)

microcurie: one-millionth of a curie

microgram: one-thousandth of a milligram

micron: one-thousandth of a millimeter; a particle 0.001 millimeter in diameter

millirad (mrad): 1/1,000 of a rad

minor injury: injury requiring no more than first-aid treatment

moderator: a material used to slow down fast neutrons

modified Mercalli scale: one of the earthquake intensity scales, having 12 divisions ranging from 1 (not felt by people) to XII (damage nearly total). (Wood and Neuman, 1931)

molecular weight: the sum of the atomic weights of all the constituent atoms in the molecule of an element or a compound

monitor: determination of the amount of radioactivity present; the act of determining or the individual doing the "monitoring"

MUF: a term used in accountability of nuclear materials meaning "material unaccounted for"

MWd: see Megawatt Day

N

natural uranium: uranium containing approximately 0.72% ^{235}U also called 72 Metal

neptunium (Np): synthetic radioactive element made by bombarding ^{235}U with neutrons; atomic number 93, the mass number of its most stable isotope is 237

neutralization: chemical reaction between an acid and a base in such proportions that the characteristics of each disappear ($NaOH + HNO_3$ $NaNO_3 + H_2O$)

neutron: a particle existing in or emitted from the atomic nucleus; it is electrically neutral and has the same mass as a proton

neutron interaction: exchange of neutrons between two or more fissile units that results in an increased reactivity of the system

nitration: the introduction of the nitro group, oxides of nitrogen, into an organic compound, usually with nitric acid or with some mixture containing nitric acid

noble gas: rare inert gas, e.g., helium, neon, argon, krypton, xenon

noncontrolled exposure: the exposure to ionizing radiation under conditions which are beyond those defined as controlled exposure; generally, exposures to the public

nozzle: a small spout, usually designed to control velocity

nozzle (as relates to PUREX): a male fitting to which a jumper is attached; it could be a liquid, steam, air, or electrical connection

9413219.1174

nozzle plate: sieve plates in a column which have the holes indented to cause them to act as tiny jetting nozzles

NPH: normal paraffin hydrocarbon - diluent for tributyl phosphate used as the organic phase for the solvent extraction columns in PUREX

N Reactor: the low pressure steam reactor (NPR) built under contract for the former Atomic Energy Commission for the dual purpose of plutonium and power production

nuclear blank: a means of preventing uncontrolled reagent addition through physical separation of a header from a process vessel by removal of a section of the interconnecting piping (criticality prevention measure)

nuclear energy: energy released in a nuclear reaction, such as fission

nuclear excursion: accidental criticality; an uncontrolled reaction event

nuclear facility: a facility where nuclear materials are processed, stored, and/or otherwise handled

nuclear reaction: result of the bombardment of a nucleus with atomic or subatomic particles or very high energy radiation; also the decay of radioactive materials

nuclear safety: the application of technical knowledge and administrative control to prevent an unplanned, uncontrolled nuclear chain reaction. Also refers generally to radiation safety

nucleus: the positively charged center of an atom

nuclide: a species of atom having a specific mass, atomic number, and nuclear energy state

O

offgas heater: a heater in an offgas system which heats up the air for a processing reason or for keeping vapors from condensing

operability test: trial to determine if designated equipment will perform its intended function

operating procedures (SOP, POP): written procedures describing a sequence of operating steps, including safety instructions

operating rate: the amount processed in a stated period of time, e.g., tons/day

Operations Safety Requirements (OSR): definition of the acceptable conditions, safe boundaries and bases, and management controls required to assure safe operation of the facility during operation

organic: the term generally refers to the solvent and hydrocarbon diluent used in solvent extraction, e.g., tributyl phosphate and normal paraffin hydrocarbon

organic saturation: a measure of the quantity of product(s) contained in the organic

orifice: a clean cut hole in a thin plate which restricts the channel (usually a pipe); used to restrict flow, i.e., permit a measured flow to pass at a given pressure

overbatch: planned or inadvertent exceeding of a quantity operating limit

overhead crane: a bridge crane which runs on tracks above head level, such as found in the canyon building

overheads: liquids or vapors coming off the top of a tower or column, the distillate or condensation of vapors of distillation

oxidation (oxidant, oxidizer): the chemical combination of an element with oxygen; any increase in valence owing to the loss of one or more electrons

oxidize: to combine oxygen with some substance

P

P&O gallery: pipe and operating gallery

package (cell): a grouping of equipment handled as a unit

packing: rings used in columns; asbestos-type material used to prevent leaks in valves and pumps; material used for protection during shipment

panel board: panels where recorders, indicators, switches, and other control devices are located

"panic button": an emergency shutdown switch

partition (cycle): to separate one element from others; in processing operations, the separation, such as plutonium from uranium

pencil: pocket dosimeter ionization chambers capable of recording gamma radiation received by the wearer

9413219.1176

9413219.1177

penetrating dose: an absorbed dose from gamma or X-rays or neutrons of sufficient penetration to affect the critical organs, principally the blood-forming organs

penetrating radiation: forms of radiant energy which are capable of passing through significant thicknesses of solid material; these usually include gamma rays, X-rays, and neutrons

percent of chart: the point on a recording or controller chart at which the pen is recording and from which flow, volume, etc. can be determined by referring to a calibration chart

percolate: pass a liquid through a porous substance for extraction or filtering purposes; the movement of moisture through sediments

permissible dose: that dose of ionizing radiation which, on the basis of present knowledge, carries negligible probability of causing severe somatic injury or genetic effect

pH: a measure of acidity and alkalinity, neutrality being at pH 7 with acidic solutions being less than 7 and basic solutions being higher than 7

phase: a solid, liquid, or gaseous homogeneous substance that exists as a distinct and mechanically separate portion in a heterogeneous system

phase flow ratio (L/V): ratio of the total aqueous flow to a column to the organic flow to that column

pipe trench: the canyon building trench which holds the pipes carrying radioactive process solutions from cell to cell

piping diagram: a line diagram of pipe systems

Plant Operating Procedures (POP): see Standard Operating Procedures

plutonium (Pu): artificially produced radioactive element, atomic number 94, atomic weight 242, m.p. 640°C, b.p. 3,235°C, d 19.84 g/cm^3. The isotope ^{239}Pu is an alpha emitter with a half-life of 24,000 yr, used in nuclear weapons and as nuclear fuel.

plutonium polymer: a polymer of plutonium as $[Pu(OH)_x(NO_3)_{(4-x)}]_y$, formed in low acid solutions

pneumatic transmitter: a device or instrument using air to transmit flow, temperature, pressure, liquid-level measurements from process locations to a central control board

9413219.1178

poison (neutron poison): a nonfissionable material which absorbs neutrons, thus reducing the flux available for fission

pollution: presence of contaminants in the environment

polymer: a compound formed by a chemical reaction in which two or more molecules of the same substance combine to form a compound from which the original substance may or may not be regenerated, and the new molecular weight is a multiple of that of the original compound

pool cell: a "cell" filled with water to provide shielded storage of contaminated equipment

population dose (population exposure): the summation of individual radiation doses received by all those exposed to the sources or event being considered

porosity: the ratio of the volume of void space in a rock or sediment to the total volume

power reactor: a nuclear reactor designed to produce heat for conversion to electrical energy or mechanical propulsion

PR (PR room): product removal; referring to the plutonium nitrate product removal facility at PUREX

PR can: a $Pu(NO_3)_4$ transfer container used in the Hanford 200 Areas.

precipitate: solids which separate from a solution and settle at the bottom of the vessel containing it; to cause such a separation to occur, usually by the addition of a chemical

prejob safety planning: written analyses of the job and procedure that points out necessary safety equipment and methods to perform the job safely

pressure drop: the difference in pressure between two points in a process or piping system

pressure vessel: a tank which can operate under pressure, such as a hot water tank

preventative maintenance: routine work done to maintain equipment in good operating condition

procedure: a written set of instructions, approved by appropriate management, detailing sequential steps for safe and efficient performance of a task

process air: air designated for process use, such as for jet operation

process cells: enclosures housing the radioactive process systems and other systems or services directly associated with the process equipment

process equipment: the functional equipment items or systems associated directly with the separation of fissile materials from fission products

process specifications: operating limits established for safe production of quality products in accordance with the Operations Safety Requirements (OSR)

process standards: written administrative controls for operating variables which are more stringent than the process specifications, designed to minimize the potential for the violation of the process specifications. Trackable to the Operations Safety Requirements (OSR)

process test: a formal written and approved procedure to test a new method of operation

process water: the water used in the process, generally demineralized or "softened"

product removal and handling (PR) can: a container used to transport plutonium solutions

product specifications: detailed product requirements, including such things as form, purity, concentration, and limits of impurities which products must meet to be acceptable to the customer

production reactor: a nuclear reactor designed for transforming one nuclide into another; usually a conversion of natural uranium into plutonium

project proposal: a document prepared to request approval and funds from the DOE for the design and/or construction of new facilities

proportional sampler: a sampling device designed to continuously sample a process stream to obtain a representative sample even though the flow rate or composition might change

protected area: a new term for an exclusion area which is a security area where mere access to the area constitutes access to classified information

protected process water: process water which has been pumped through special tanks which make backflow impossible

proton: a positively charged particle of about the same mass as a hydrogen atom (which is 1 on the chemist's scale) and about 1,800 times heavier than the electron. The electrical charge of the proton is equal and opposite to the charge of the electron. Protons and neutrons make up the nucleus of the atom.

pulsating: vibrating or throbbing

pulse: the up-and-down movement of liquid in a column

pulse amplitude: the magnitude of the up-and-down motion of the liquid contents of a pulse column, usually expressed in inches

pulse frequency: the time rate of pulsing, expressed in cycles per minute

pulse leg: the pulse transmission line which transmits the pulse from the pulse-generator piston to the bottom of the column; the leg needs to be of sufficient height to balance most of the hydrostatic head of the liquid in the column

PUREX (acronym for plutonium-uranium extraction): the facility and process which uses solvent extraction and ion exchange for the separation of plutonium, neptunium, and uranium from irradiated production fuels; the 202-A Building in the 200 East Area

purge: removal of material from a system or pipe by adding another, such as blowing with air

purgerator: an instrument used to regulate the flow of purge air

pyrophoric: taking fire spontaneously on exposure to air

Q

quick-sort: procedure for checking personnel after a criticality accident to determine extent of radiation exposures and contamination

R

rad: a special unit of measure for the absorbed dose of radiation; one rad equals 100 ergs absorbed per gram of material

radiation (ionizing): particles and electromagnetic energy emitted by nuclear transformations which are capable of producing ions when interacting with matter; gamma rays and alpha and beta particles are primary examples in Hanford waste

radiation (radioactivity): the emission of atomic particles or rays by nuclei

radiation incident: a condition or situation which resulted in personnel external exposure or internal deposition of radionuclides exceeding the standard or operational controls

radiation monitoring: a term covering application of a field of knowledge, including the determination of dose rates, surveys of personnel and equipment for contamination control, air sampling, exposure control, etc.

Radiation Monitoring group: a group with designated responsibility for adequate radiation protection practices; a group providing radiation monitoring service

radiation occurrence (RO): a condition or situation involving radiation which is an undesirable event of sufficient importance or widespread interest as to warrant investigative action by supervision

radiation safety: the state of awareness and understanding necessary in the safe handling of radioactive materials and safe conduct in areas where ionizing radiation is present

radiation survey: evaluation of an area or object with instruments in order to detect, identify, and quantify radioactive materials and radiation fields present

radiation work permit (RWP): a written procedure for work in radiation zones defining radiological conditions, precautions, protective clothing, and respiratory protection requirements

radiation zone: area which contains radioactive materials or radiation fields in quantities significant enough to require control of personnel entry to the area

radioactive (decay): property of undergoing spontaneous nuclear transformation in which nuclear particles or electromagnetic energy are emitted

radioactive liquid waste crib: a rock-filled, V-shaped trench with concrete (or soil) covers used for disposal of intermediate-level liquid waste

radioactive liquid waste tile field: tile field used for disposal of low-level radioactive liquid waste (see tile field)

radiolysis: the breaking up of a compound (usually H_2O) by radioactivity

radionuclide: a nuclide which is radioactive

9413219.1182

raffinate: in a solvent extraction process, the liquor which remains after the major portion of desired material has been removed by extraction

railroad cut: that portion of the railroad from the exclusion area fence to the process building

Raschig ring: column or scrubber packing consisting of metal or ceramic cylindrical rings of same length as diameter; rings function as a means to "break up" process streams and provide large surface areas for contact

raw water (RW): untreated and unchlorinated water used principally for cooling

reactivity: a measure of the capability of a system to maintain criticality; systems with high reactivity are capable of undergoing rapid excursions of increasing power; systems with low reactivity will undergo slower excursions; systems with negative reactivity will not become critical

reactor: a nuclear reactor

recorder-controller instrument: an instrument which measures (such as temperature, pressure, volume, flow) and from the measurement automatically adjusts the process measured to the preset limits

rectification: a distillation carried out in such a way that the vapor rising from a still comes in contact with a condensed portion of vapor previously evolved from the same still

recycle: process material routinely returned to the system for reprocessing

red oil: a product of degradation of the TBP-organic system in some form of an nitrated organic-metal complex; the material is a heavy, oily substance which is explosive at temperatures above 135°C

REDOX: a facility and the process for separating plutonium and uranium from irradiated reactor fuels by using successive steps of chemical reduction and oxidation together with solvent extraction. The plant, the 202-S Facility in the 200 West Area, was completed in 1952 and deactivated in 1967.

reduction: to change from higher to lower valence, such as to change Pu^{+6} to Pu^{+4} in solution by addition of chemicals such as ferrous ion; also to change a metallic compound to metal, such as Pu^{+4} to plutonium metal in the button line operation; also, the gaining of electrons by an atom, ion, or element

reflector: material which reflects neutrons. "Thick reflector" means material with a neutron reflecting capacity equivalent to more than 1 in. of water. "Normal reflector" means material with a neutron-reflecting capacity equivalent to not more than 1 in. of water. "Minimum reflector" means material with a neutron reflecting capacity equivalent to not more than 1/8 in. of stainless steel or other common metals.

reflux: the process of returning condensed vapors to the system; the repeated extraction and stripping of a material in a single solvent extraction column

regional monitoring: the monitoring of land areas and vegetation outside a plant area

regulated area: a radition zone, access to which is limited and controlled

regulated shop: maintenance repair shop within a radiation zone

regulated tools: tools and equipment used only in radiation zones

release limit (release guide): a control value which regulates the concentration or amount of radioactive material released to the environment in an industrial situation; usually dose to persons in the environment derived from environmental behavior to the released material so that the dose is kept below a selected level

rem (acronym for "roentgen equivalent man"): a unit of measure for the dose of ionizing radiation which gives the same biological effects as one roentgen of X-rays; one rem approximately equals one rad for X, gamma, or beta radiation.

remote maintenance: repair work done at a distance so that no contact by personnel is made; e.g., remote maintenance in PUREX by use of the canyon cranes

reprocessing: chemical processing of irradiated nuclear reactor fuels to remove desired constituents

reservation: as applied to the Hanford Works, refers to land, air, and water administered by the DOE

resin: material used in an ion exchange operation to separate by reversibly holding fixed certain specific materials in ionic form

respiratory protection: equipment, such as full-face respirators, self-contained breathing apparatus, or fresh air masks worn to protect personnel from inhaling harmful gases or particulate matter

restricting orifice: see orifice

9413219.1184

retention basin: open large-capacity solution storage area for keeping potentially contaminated solution until analysis indicates it meets disposal criteria

reverse strike: alkalinization process consisting of transferring acidic waste into caustic solution that has been previously added to a waste alkalinization tank

rework: to process again

risk: the multiplicative product of the probability of a particular event and its consequences; units of Ci/yr and rem/yr are commonly used

roentgen: a unit of measure of ionizing electromagnetic radiation (X and gamma); one roetgen corresponds to the release by ionization of 83.8 ergs of energy per gram of air

Rotameter*: an instrument used to measure the flow rate of a fluid. The fluid, passing up through a vertical tapered tube, moves a "rotor" (modified cylinder) within the tube to the height required to let the fluid pass between the tube and rotor, and flow is determined by rotor position

S

safe batch: quantity of material in which the total plutonium content is controlled (concentration and mass) to less than a minimum critical mass

safe geometry: shape of process vessels such that a nuclear reaction cannot be sustained with the process materials (changes in reflection, moderation, addition of dense plutonium might cause criticality)

safe ratio: ratio of plutonium concentration to natural uranium or enriched metal below which criticality is not possible in an aqueous homogeneous system

safe solution: solution in which the plutonium concentration is maintained within subcritical concentrations

salt cake: damp solids formed by evaporation of waste containing radioactive contaminants to concentrations exceeding solubilities in the evaporator-crystallizer (products are Hanford defense residual liquor and salt cake)

---

*Registered trademark

salting agent: a chemical which, when added to a two-liquid phase system, increases the transfer of materials into the desired phase

sample: a small representative portion of the total

sample gallery: specific area in the PUREX Plant where process samplers are located

sanitary waste: human wastes and other nonradioactive material for disposal to preserve public health

sanitary water: water which has been treated (filtered, purified, etc.) and is usable for drinking

saturated solution: a solution which contains the maximum possible amount of dissolved material under given conditions

saturation: point where solution will not absorb any more solute

scintillation (counter): light flashes produced in crystalline material by ionizing radiation (device for measurement of the level of activity of a source utilizing a photomultiplier to detect scintillation)

scintillation monitor: a radiation detection or measurement instrument based on the principle that light pulses are produced in some materials when they are exposed to radiation

scrubber: equipment (tank or column) used to clean process and effluent streams

seal pot: to prevent air entering, yet provides release of excessive pressures which may develop within the system

secondary containment: the mechanical means which provides the second line of defense in preventing spread of contamination

security area: physically defined space containing classified matter and subject to physical protection and personnel access controls

SEFOR: Southeast Fast Oxide Reactor

self-absorption: the absorption of radiation by the emitting substance itself

self-concentration: allowing radioactive waste to concentrate as material, through decay of radioactivity, self-heats to boiling and drives off water as vapor

shielding: the material interposed between a source of radiation and personnel for protection against danger of radiation; common shielding materials are concrete, water, and lead

shipping cask: specially designed container used for shipping radioactive products

short-lived isotope: a radioactive nuclide which decays so rapidly that a given quantity is transformed into its daughter products within a short period (usually those with a half-life of days or less)

sieve plate: perforated plate in an extraction column used to break up and disperse the liquid phases

silica gel: a colloidal form of silica used to remove zirconium-niobium from UNH streams

silver reactor: equipment used for removal of the radioactive iodine

simple column: a column having two streams flowing into it and two out of it; and designed to perform only one solvent extraction function (extraction, scrubbing, or stripping)

site: the geographic area upon which the facility is located and which is subject to controlled access

Site: the Hanford Site

sky shine: gamma radiation "scattered" or reflected back by the atmosphere, similar to scattering or diffusion of light by the air

slave crane: crane which may be operated directly and independently or operated remotely in conjunction with the master crane

slump: a landslide characterized by a shearing and rotary movement of a generally independent mass of rock or earth along a curved slip surface

smear: a means of measuring loose surface contamination on an object by wiping it with paper, gauze, etc. and then measuring the radioactivity on the wipe

solenoid: electrically operated device for opening and closing valves; the word solenoid is used with or without the word valve

solid wastes (radioactive): either solid radioactive material or solid objects which contain radioactive material or bear radioactive surface contamination

solvent extraction: contacting immiscible liquid phases for the purpose of transferring solute, preferentially from one phase to the other. In laboratories, batch and semicontinuous contacts are used in analytical procedures. Pulse columns have been devised as a continuous contacting system for PUREX

solvent recovery: reclamation of organic extractant for reuse in solvent extraction processes

solvent treatment: recovery process whereby degradation products are removed from the solvent extractant to render it acceptable for reuse. The TBP-NPH is regenerated with alkaline permanganate washes

source (radiation): calibrated sample of radioactive material used to check instrument response and ability to measure; any point or area from which radiation originates; a radiochemical standard with known characteristics

source data: report of nuclear material status at an accounting point

source material: in atomic energy law, any material except special nuclear material which contains 0.05% or more of uranium, thorium, or any combination of the two

source term: the quantity of radioactive material released by an accident or operation which subsequently causes exposure after some mechanism of transmission or deposition

sparger: perforated pipe or ring at the bottom of a tank through which air, steam, or gas is introduced so that the bubbling action of the fluid coming from the sparger agitates the solution in the tank

special nuclear materials (SNM): plutonium, ^{233}U, or uranium enriched to a higher percentage than normal of the 233 or 235 isotopes, or any material artificially enriched in any of these substances

specific gravity: weight of a given substance compared to an equal volume of pure water at 4°C

"spike" or spike metal: a fuel element whose outer cylinder is enriched to 1.25% ^{235}U, whose inner cylinder is enriched to 0.947% ^{235}U, yielding an average enrichment of 1.15% ^{235}U

stability (atmospheric): a description of the atmospheric forces on a parcel of air following vertical displacement in an atmosphere otherwise in hydrostatic equilibrium; if the forces tend to return the parcel to its original level, the atmosphere is stable; if the forces tend to move the parcel further in the direction of displacement, the atmosphere is unstable; if the air parcel tends to remain at its new level, the atmosphere has neutral stability

stack gas: gaseous discharge from a "smoke" stack, such as from the PUREX Plant stacks or gas from the powerhouse

stage: in solvent extraction, a single equilibrium contact followed by separation of phases is a stage. In a continuous contactor (pulse column), the height of column that represents a stage will vary, depending on process variables; i.e., temperature, pulse frequency, efficiency of mixing, etc.

Standard Operating Procedure (SOP): operating instructions, same as Plant Operating Procedures.

standby: the condition where a facility or burial ground, etc. is placed in a nonoperating condition but is maintained in readiness for subsequent operation

steam strip: removal of a compound from a solution by introduction of steam, e.g., organic from condensate or aqueous solutions

step-off (pad): area between designated zones, passage through which requires survey and may require removal of protective clothing such as shoe covers, coveralls, gloves

stoichiometric: the quantitative relationships between elements and compounds taking part in a chemical reaction. The exact amount of material that will react completely with another is a stoichiometric quantity

stream: flow of a liquid or gas; process stream is the flow of product materials, such as plutonium nitrate solution; a waste stream is the flow of effluent containing the unwanted materials

stripping: in the PUREX solvent extraction process, the transfer of product from the organic phase back into the aqueous phase

subcritical: less than a critical amount; the status of a system in which more neutrons escape than are being produced

sublimation: the change of a substance from a solid state to a vapor without passing through a liquid state, e.g., iodine crystals vaporize when heated

suction: reduction of pressure in a pipe so that pressure difference forces fluid to flow through a pipe (vacuum transfer)

sump: a pit, tank, or reservoir serving as a drain or receptacle for liquid, usually located at the lowest point in the system

supercritical: the condition resulting from a fission reaction where each fission causes more than one new fission

superheat: to heat a liquid above its boiling point without converting it into vapor; steam heated so that it possesses more than sufficient heat to remain vaporous at the given pressure

<u>surveillance</u>: those activities necessary to assure that the site remains in a safe condition (including periodic inspection and monitoring of the site, maintenance of barriers to access to radioactive materials left on the site, and prevention of activities on the site which might impair these barriers)

<u>survey</u>: to determine with instruments the amount of contamination or radiation present

<u>swamp</u>: ground area where uncontaminated or low-level waste water is intentionally discharged to seep into the ground

T

<u>tank</u>: a metal container commonly located underground (at Hanford) for storage of liquid wastes or a process vessel in a facility

<u>tank farm</u>: an installation of interconnected underground tanks for storage of radioactive wastes

<u>technical manual</u>: manual which describes the technical aspects of an operation and establishes operating conditions to maintain process-product specifications

<u>theoretical stage</u>: in aqueous-organic systems, the mixing and settling stage in which equilibrium between the two phases is attained

<u>thermohm</u>: temperature measuring device based on the principle that electrical resistance in a metal conductor changes with temperature; the element is usually nickel or platinum wire coil; temperatures can be read directly by the deflection of a millivolt meter in the system

<u>thermocouple</u>: a device for measuring temperature consisting of two dissimilar metal wires joined at one end (hot junction) with the free ends joined to measuring instrument; electrical potential changes due to temperature changes at the hot end are masured at the cold end and calibrated to read out as temperature

<u>thief sample</u>: sample taken by means of inserting a long sampling tube into the material (usually liquid)

<u>thoria</u>: thorium oxide

<u>threshold</u>: the point at which an effect is first produced, observable, or other indicated

<u>threshold dose</u>: the minimum absorbed dose that will result in a detectable degree of any given effect

9413219.1190

threshold neutron monitor: a neutron detection and measuring device which employs several detectors; each detector will react only with neutrons above a certain energy or "threshold," such that neutron spectrum measurements may be made

third-party inspection: inspection or audit made by person(s) not associated with the user group

tickler - tickler file: file card reminder; filed by date to be removed for doing periodic or routine jobs

tile field: area containing earthenware drain pipes from which sanitary or uncontaminated waste water drains to percolate into the soil; waste water drain area

titration: the volumetric determination of the strength of a solution of an acid, oxidizing or reducing agent by delivering from a burette an appropriate reagent of known strength using a suitable indicator or instrument to determine end point

ton: 2,000 lb

tonic effect: phenomenon observed in HA column operation associated with one of two changes. First, addition of rework of high-level waste or other rework material; the second being addition of trace quantities of fluoride being added to the feed. In either case, it is observed that fission products decontamination is markedly enhanced and, on occasion, for unidentified reasons, neptunium losses are markedly reduced.

tonne (t): 1,000 kg

trace: an amount of a substance present in negligible concentration; the record made by a self-registering instrument; to follow out a line so as to determine where it goes and where it starts

tracer: an element or atom having a peculiarity such as radioactivity whereby it can be traced through a chemical reaction or process; an auxiliary line, such as tubing, on the outside of a chemical line through which a stream may be passed to keep the chemical warm or prevent freezing

trap: a device used for removing or restraining a material flow; device to remove condensed steam from a steam line

trap pit: a covered pit housing steam traps

transuranic elements: those elements which have an atomic number higher than 92; these elements do not normally occur in nature and have to be produced artificially from uranium, either directly or indirectly by successive steps of transmutations. (^{93}Np, ^{94}Pu, ^{95}Am, ^{96}Cm, ^{97}Bk, ^{98}Cf, ^{99}Es, ^{100}Fm, ^{101}Md, and ^{102}No)

transuranic wastes: any waste material measured to contain more than a specified concentration (presently proposed as 10 nCi of alpha emitters per gram of waste) of transuranic elements

transuranium: nuclides having an atomic number greater than that of uranium (i.e., greater than 92)

trench: a ditch used for the disposal of solid radioactive waste or low-level liquid waste

Tri-n-butyl phosphate (TBP): an extractant used in processing nuclear fuels used in the PUREX process

trophic levels: pertains to groupings of organisms according to characteristics of their intake of nutrition

trunnion: either of two opposite projecting pivots over which a lifting yoke will fit to enable the equipment to be picked up and moved or turned

tube bundle: series of tubes made as a unit to fit inside a vessel and used for transferring heat

tunnel: railroad passageway into the PUREX processing building

turbomixer: a mixer which provides circulation (mixing) through centrifugal impellers

U

unit mass: a specified weight of fissile material in an easily recognized location (such as the contents of a single can, a group of containers at one location or the mass in a piece of equipment)

uranium: radioactive element, atomic number 92, atomic weight 238.07; natural uranium consists of isotopes 238 (99.3%), 235 (0.7%), and 234 (trace)

uranyl nitrate hexahydrate (UNH): uranium as a nitrate [$(UO_2)(NO_3)_2.6H_2O$] recovered an aqueous solution in the PUREX Plant. This term is also used interchangeably to mean the aqueous solution

V

vacuum fractionator: a nitric acid concentrator which operates under reduced pressure

vacuum transfer: transfer, usually of solutions, by means of a vacuum or suction

vapor lock: partial or complete interruption of fluid flow caused by bubbles of air or gas in the piping system

vault: room with reinforced walls and a steel door with a combination safe-type lock or other mechanism which assures physical security

ventilation balance: control of the building air supply to maintain proper airflow in a direction towards areas of potentially greater contamination

vessel vent system: system under a slight negative pressure through which a series of process vessels are vented and the offgas is treated and filtered

W

waste management: the planned storage and disposal of radioactive waste materials, including processing to recover valuable constituents

water table: upper boundary of an unconfined aquifer surface below which saturated groundwater occurs; defined by the levels at which water stands in wells that barely penetrate the aquifer

weight factor: a measure of the hydrostatic pressure in a dip tube below the liquid level in a tank; it is used with the vessel calibration data and solution specific gravity to determine the liquid volume

wet standby: refers to the plant in a nonoperating condition, but with flush solutions circulated through all plant vessels on a scheduled basis

"whites": Hanford jargon for white coveralls worn in radiation zones for protection against radioactive contamination

whole body count: radiation survey which measures the amount of radioactive material in the body

wind regime: the regular pattern of wind over a ground surface (i.e., Hanford Site)

X

X-rays: penetrating electromagnetic ionizing radiation which results from changes which occur in the electron shell of the atom

Z

zircaloy: the zirconium-tin alloy used for fuel element cladding

Zirflex: process which employs a boiling aqueous solution of ammonium fluoride and ammonium nitrate to dissolve zirconium or zircaloy

9113219.1193

GLOSSARY OF PUREX FLOWSHEET ACRONYM MEANINGS

INTRODUCTION

Generally, the flowsheet acronyms are descriptive of the process function but there are some incongruities: the flowsheet has been changed but the old acronym has been retained. Equipment numbers are included in the glossary for correlation with flowsheet acronyms. A pictorial schematic of the PUREX process with flowsheet acronyms and equipment numbers is shown on Drawing H-2-53497 (not included). Typical acronyms are given below:

TYPICAL ACRONYMS:

AAD: Acid absorber distillate. Vapor from the first cycle waste Concentrator E-F6 is stripped of acid in the acid absorber T-F5. The lean vapor leaving the acid absorber is condensed (AAD) in Condenser E-F5.

AAR: Acid absorber reflux. Part of the condensate from E-F5 is recycled as lean feed to the acid absorber.

AFD: Acid fractionator distillate. This is the condensed lean vapor from the acid fractionator T-U6.

AFR: Acid fractionator reflux. Part of the distillate from the concentration of nitric acid in the acid fractionator T-U6 will be reintroduced into the tower of the acid fractionator to strip the vapors of acid.

ASD: Ammonia scrubber distillate. This condensate is produced in the decontamination of ammonia scrubber waste by evaporation in the E-F11 Concentrator.

HAS: HA (column) scrub. Scrub solution for the HA column (T-H2) removes some of the fission products extracted in the organic phase.

1CU: One - "C" (column) uranium. After the initial decontamination of uranium and plutonium are separated into two separate streams-an aqueous and an organic. The uranium in the organic phase is back-extracted into the aqueous phase in the 1C column; the 1CU is the 1C column product.

1CX: One - "C" (column) extractant. This is a weak acid stream used to back-extract the uranium from the organic into the aqueous phase.

1UC: First (cycle) uranium concentrate. The 1CU is concentrated in E-J8 before the second cycle.

1UD: First (cycle) uranium distillate. This is the process condensate from Concentrator E-J8.

1WF: First (cycle) waste feed. Tank F7 is feed tank for the high-level waste Concentrator E-F6.

2AW: Second "A" (column) waste. This column is used for the intermediate purification of plutonium. The column equipment number is T-L2.

2DF: Two - "D" (column) feed. Final purification of the uranium is by aqueous-organic contact in column T-K2.

2DW: Two - "D" (column) waste. This stream is the largest flow in the 3WF.

2EU: Two - "E" (column) uranium. The uranium-rich organic from the 2D column is back-extracted into the aqueous phase in the 2E column (T-K3). The aqueous phase is the 2EU.

2EX: Two - "E" (column) extractant. This is a weak nitric acid stream used to back-extract the uranium from the organic phase in column T-K3.

2UD: Second (cycle) uranium distillate. The 2EU is concentrated to product specifications in the 2EU Concentrator E-K4. The 2UD is the process condensate from this concentrator.

3WB: Third waste backcycle. (The three is misleading.) This acronym is used in reference to the backcycle waste evaporator E-H4.

3WD: Third (backcycle) waste feed. Sort of misleading; the three is used to pertain to the fact that this was part of a third cycle opera-tion. All the second cycle wastes are concentrated and recycled for additional recovery.

3WW: (Backcycle) waste. The 3W prefix used to be consistent with other backcycle terms. The 3WW is concentrated bottoms from the E-H4 Concentrator.

NOMENCLATURE LIST

AEC	Atomic Energy Commission
AFAN	ammonium floride and ammonium nitrate
ALARA	as low as reasonably achievable
AMU	aqueous makeup
ANN	aluminum nitrate nonahydrate
AOV	air operated valve
AP	analytical procedures
ASG	audible signal generator
ASW	ammonia scrubber waste
ASD	ammonia condensate stream
AT	alpha total
ATP	acceptance test procedure
BP	before present
BPF	blue print file
BWIP	Basalt Waste Isolation Project
CAD	criticality alarm detectors
CADM	criticality alarm detector modules
CAM	continuous air monitor
CARM	criticality alarm relay module
CAW	current acid waste
CCR	central control room
CCTV	closed-circuit television system
CEO	change engineering order
CF	capacity factor
CIZ	controlled injury zone
CMP	crane maintenance platform
CP	cutie pie
CPS	criticality prevention specification
CRM	count rate meter
CSAR	criticality safety analysis report
DBT	Design Basis Tornado
DF	decontamination factor
DOE	U.S. Department of Energy
DOG	dissolver offgas
DOP	dioctylphthalate
DOV	diaphragm operated valve
DP	differential pressure
ECM	executive control meeting
ECMP	east crane maintenance platform
EFD	engineering flow diagram
EIS	enivronmental impact statement
EPA	Environmental Protection Agency
ERDA	U. S. Energy Research and Development Administration

FCN	facility change notice
FFTF	Fast Flux Test Facility
FSAR	Final Safety Analysis Report
FP	fission product
FPD	functional planning document
HCR	headend control room
HEHF	Hanford Environmental Health Foundation
HEPA	high efficiency particulate air
HMS	Hanford Meteorological Station
HRHE	Hanford Regional Historical Earthquake
HVACR	heating, ventilating, and air conditioning control room
IFD	instrument flow diagram
IHSD	Industrial Hygiene and Safety Department
LCO	limiting conditions for operation
LCS	limiting control setting
LOI	loss on ignition
LOP	laboratory operating procedures
MM	Modified Mercalli
NCSC	Nuclear Criticality Safety Committee
NCSS	nuclear criticality safety standards
NPH	normal paraffin hydrocarbon
NPR	new production reactor (N Reactor)
NRC	Nuclear Regulatory Commission
OSHA	Occupational Safety and Health Act
OSR	operational safety requirements
OTP	operability test procedures
PAPR	powered air purifying respirator
PAW	PUREX acid waste
PHA	Preliminary Hazards Analysis
PMF	probable maximum flood
PNL	Pacific Northwest Laboratory
POP	plant operating procedures
PPC	PUREX Process Control
P&O	pipe and operating
PR	product handling and removal
PRV	pressure relief valve
PSSC	Packaging and Shipping Safety Committee
PST	pacific standard time
PTP	process test procedures
PUREX	plutonium-uranium extraction

9413219.1197

QA	Quality Assurance
QAPP	Quality Assurance Program Plan
QE	Quality Engineer
QC	Quality Control
QE&C	Quality Engineering and Control
Rockwell	Rockwell Hanford Operations
R&E	Research and Engineering
RL	Richland Operations Office (of DOE)
RMA	remote mechanical aoxide line
RM/PD	radiation monitoring/personnel decontamination
RO	radiation occurrence
RW	raw water
RWP	Radiation Work Procedure
SAR	Safety Analysis Report
SEFOR	Southeast Fast Oxide Reactor
SNM	special nuclear materials
SOP	Standard Operating Procedure
S&QA	Safety and Quality Assurance
SSE	Safe Shutdown Earthquake
SWP	Special Work Permit
TBP	tributilized phosphate
TLD	thermoluminescent dosimeter
TLV	threshold limit value
TRU	transuranic
UBC	uniform building code
UNC	United Nuclear Industries
UNH	uranyl nitrate hexahydrate in acidic solution
USGS	U.S. Geological Survey
WBSK	work breakdown structure
WF	weight factor
wg	water gauge

9413219.1198

9413219.1199

PUREX FSAR

APPENDIX B HISTORY OF PUREX PLANT

This page intentionally left blank.

9413243.1200

APPENDIX B

HISTORY OF PUREX PLANT

This appendix presents a thorough discussion of the history of the Plutonium-Uranium Reduction Extraction (PUREX) Plant.

The PUREX Plant was committed to radioactive service in 1956 as a major producer of weapons-grade plutonium from irradiated uranium metal reactor elements. When it was shut down in 1972, the plant was a major facility of the then U.S. Atomic Energy Commission (AEC) effort to supply spent fuel separation capability to a variety of programs. Eight phases are discernible in the history of the facility (roughly 1954 to the present):

- A 2-yr building period
- A 4-mo testing period
- Startup
- Refinement
- Increasing versatility
- Standby
- Upgrading
- Startup.

Efforts to refine and improve flowsheets and processes preceded radioactive operation and continue to the present. Multipurpose use of the plant became a defined objective by 1960, and by the time the plant was placed on standby, it was processing entirely different materials from those originally processed and will continue in this manner for future operations.

PLANT MISSION

Originally, the PUREX Plant reprocessed irradiated, aluminum-clad, reactor fuel elements, employing a wet chemical process to separate fission products (FP) from the uranium and plutonium. This process resulted in uranium nitrate and plutonium nitrate, with both chemicals emerging in a highly purified state. The uranium nitrate was subsequently converted to uranium oxide (UO_3) and shipped offsite for enrichment and reuse as reactor fuel. The plutonium nitrate was further processed onsite and was eventually used in weapons fabrication.

The fuel elements involved in the original process were low-burnup elements in the 250-700 MWd/t range. These elements could be handled at the PUREX Plant following short cooling periods (90 to 120 d) and yielded small amounts of plutonium (a few grams per tonne). With this fuel, plant capacity was far in excess of the original design basis.

Table B-1 lists other materials processed by the PUREX Plant with the respective dates the various feed stocks were introduced. With minor exceptions, all of the processed materials were irradiated in Hanford Site reactors.

TABLE B-1. Fuels Reprocessed at the PUREX Plant.

Time period	Fuel
January 1956	"72" metal (natural U) Al-clad, low Pu g/t (weapons grade)
August 1958	First Np recovery
October to December 1960	Began recovery of FP, ^{90}Sr, and ^{137}Cs
August 1962	"72" metal (natural) Al-clad, moderate Pu g/t (fuels grade)
November 1965	"72" metal (natural) Al-clad, moderate Pu g/t (fuels grade); SEFOR[a] campaign(s)
February 1966	"95" metal (enriched) eventually both weapons
January 1965, May 1966, September 1970	Al-clad thoria target elements; oxide powder in 1965 and 1966; compressed wafers in 1970
June/July 1967	Zircaloy-clad "95" (enriched) metal and "1.25" (spike)--both from N Reactor
February 1968	Al-clad natural U from older Hanford Site reactors and Zircaloy-clad 210 (2.1% ^{235}U) "Coproduct" from N Reactor
February 1969	PRTR[b], Zircaloy-clad PuO_2 (initially less than 1% Pu), UO_2 (1970--0.5, 1.0, 2.0, 4.0)
January and June 1972	Mixed oxide scrap (PuO_2-UO_2) in mild-steel cans; Zircaloy-clad mixed oxide
May 1982	Pu-Al and U-Al alloy (PRTR and Phoenix Reactors)

[a]Southeast Fast Oxide Reactor.

[b]Plutonium recycle test reactor.

Representative plant performance is shown in Table B-2, giving the relative purity of products obtained.

TABLE B-2. Historical PUREX Plant Performance.

Process	1956[a]	1959[b]	1967	1971
Separation ratios				
U from Pu[c]	10^7	10^7	10^7	10^7
Pu from U[d]	4×10^4	4×10^4	6×10^5	6 x 10
Plutonium decontamination				
FP from Pu	10^7[e]	10^8	10^8	10^8
First-cycle decontamination factor	3.5×10^5[f]	2×10^4	5×10^3	2×10^3
Second-cycle decontamination factor	30	2×10^3	2×10^3	1×10^3
Anion exchange	-	3	10	-
Third-cycle decontamination factor	-	-	-	50
Uranium decontamination				
FP from U	10^6	5×10^6	10^7	10^7
First-cycle decontamination factor	4.5×10^5[f]	2×10^4	3×10^3	2×10^3
Second-cycle decontamination factor	13	5×10^2	1.5×10^3	5×10^3
Silica gel	(4 to 10)[g]	-	2	-
Acid recovery				
Dissolving	-	85%	-	85%
Solvent extraction	65%	95%	-	95%
Sugar denitration	-	-	-	75%
Pu/U recovery	98%	99.9%	-	-
Solvent recovery	99%	99.7%	-	99.8%

[a]HW-31000.(1)
[b]HW-60116.(2)
[c]Pu/U in feed divided by Pu/U in product.
[d]U/Pu in feed divided by U/Pu in product.
[e]Pu reworked at Z Plant.
[f]Based on codecontamination and partition cycles.
[g]Initial decontamination factors were not as good as planned; ZrNb was the primary problem.

PLANT FLOWSHEET EVOLUTION

Modifications have been made to the PUREX flowsheet as information became available to increase capacity, decrease operating costs, reduce waste losses, and improve product purity. Major flowsheet modifications fall into six categories: (1) optimization of the original flowsheet within the objectives of the original plant mission, (2) modifications for neptunium recovery, (3) modifications for thorium processing, (4) modifications for increased processing flexibility (primarily in the headend), (5) modifications for reduced environmental releases, and (6) modification for plutonium oxide production.

The following discussion is confined to the actual, documented flowsheets. A large number of process tests were performed over the years, but not all of the tests resulted in flowsheet modifications.

Brief discussions of PUREX-type flowsheets (of which there are innumerable variants) appear in Chapters IV and VI of the PUREX Technical Manual and Chapter IV of the Reactor Handbook, Vol. II.[3,4] The development of the PUREX process was a major AEC activity involving efforts at Oak Ridge National Laboratory (ORNL), Knolls Atomic Power Laboratory, and the Hanford Site. The Hanford Site PUREX Plant flowsheet was demonstrated on a hot pilot plant scale at ORNL and in cold pilot plant work at Hanford in the 300 Area. Cold run operability tests in the plant itself were performed on what was then designated the HW 3 flowsheet. An increased capacity flowsheet (HW 4) was developed prior to hot startup, but was never used. The HW 3 flowsheet was modified, based on operational experience during cold run testing, and reissued as the PUREX Plant flowsheet (PPF) I. This flowsheet was, in turn, replaced within 6 mo of hot startup by PPF II. The changes were made primarily to improve ^{95}Zr-Nb decontamination. Conversion of the solvent extraction system from a three-cycle to a two-cycle scheme, with separate concentration and reuse of second-cycle aqueous waste streams, was accomplished and documented. With some modifications, the PUREX Plant two-cycle flowsheet was used throughout the remaining uranium-plutonium production campaigns. Updated versions were issued in 1968 for aluminum-clad fuels and in 1972 for Zircaloy-clad fuels. The current flowsheet, including the "Zirflex" process for decladding and dissolution of Zircaloy-clad N Reactor fuel, was updated in 1977. The latest flowsheet incorporated the processing of 6%, 9%, and 12% ^{240}Pu material, eliminated the recovery of neptunium and added the production of plutonium oxide (see Figure 6-1).[5]

A thorium process flowsheet was developed in 1965, modified after a PUREX Plant process test, and used for thorium processing campaigns in 1966 and 1970.

FEED PREPARATION PROCESSES

In the original feed preparation process for aluminum-clad fuel, the fuel elements were declad with a sodium hydroxide solution, containing sodium nitrate to suppress formation of hydrogen gas. After rinsing the

bare metal fuel elements, two or more dissolution cuts were taken with nitric acid, the number depending on the total metal inventory remaining in the dissolver. The individual cuts or batches of the resulting uranyl nitrate solution were then transferred to feed storage tanks. The feed was clarified by batch centrifugation. After clarification, the dissolved metal solution was chemically adjusted to assure reasonably uniform nitric acid and uranyl nitrate concentrations and then used as feed for the solvent extraction system.

Feed clarification was discontinued in March 1959. Rinsing between coating removal and nitric acid dissolution, coupled with various jetleg-length changes and allowance for solids settling, eliminated the need for centrifugation. In 1963, chemical adjustment of the feed was discontinued after demonstrated uniformity of dissolved metal solution.

The Zirflex process for chemically decladding and dissolving Zircaloy-clad uranium metal fuel elements was developed to permit reprocessing of Zircaloy-clad fuels in the existing Hanford Site production fuel reprocessing plants. The Hanford reprocessing facilities do not possess mechanical means (i.e., chop, shear, or extrude) for breaching the fuel cladding.

The Zirflex process consists of the following steps:

1. Dissolution of the Zircaloy by a mixture of ammonium fluoride-ammonium nitrate (AFAN) solution

2. Methathesis of uranium and plutonium tetrafluorides to uranium and plutonium dioxides with potassium hydroxide

3. Centrifugation of cladding waste solutions, spent metathesis solutions, and various rinses for recovery of product prior to discharge of the cladding waste to underground storage

4. Dissolution of the uranium metal with nitric acid in one or more cuts (with aluminum nitrate added to complex residual fluoride)

5. Dissolution of the cake removed from the centrifuge in nitric acid-aluminum nitrate for eventual incorporation in the solvent extraction feed.

The only significant modification to the original process was to make up an AFAN solution of the desired concentration and add the amount to the dissolver prior to charging, rather than adding a concentrated AFAN solution to boiling dissolver water. This modification was necessary to provide sufficient nitrate ion concentration in solution at the start of the decladding reaction to suppress hydrogen evolution.

Dissolver offgas treatment for the decladding process consists of ammonia removal in the downdraft condenser and water scrubber, deentrainment, offgas heating, iodine removal by passage through a packed tower containing silver nitrate-coated Beryl saddles, filtration, and discharge via the 200-ft stack. Offgas treatment for the uranium dissolution process consists of condensate recyle to the dissolver, deentrainment, offgas heating, passage through a packed tower containing silver nitrate-coated Beryl saddles for iodine removal, filtration, and additional nitric acid and iodine removal in the backup facility.

The backup nitric acid recovery system was originally installed as a dual purpose unit: the first column performing additional nitric acid absorption, the second operating as a caustic scrubber for radioiodine removal. The iodine removal in the first tower proved satisfactory. This, coupled with no measurable decline in efficiency, resulted in elimination of the caustic scrub for iodine removal in favor of additional acid recovery. As a result of increased cooling time of the fuel (to at least 180 d), radioiodine (^{131}I) levels declined to the point where operation of the silver reactors was no longer required. These were last regenerated in April 1968.

9413219.1206

The chronology of the feed preparation processes is given in Table B-3.

SOLVENT EXTRACTION AND PRODUCT HANDLING, 1956-72

The basic startup flowsheet was a three-cycle system employing a codecontamination cycle; a partition cycle consisting of an extraction column, partition column, a scrub column ancillary to the partition column, and a uranium stripping column; and separate, final decontamination cycles consisting of extraction and strip columns for the plutonium and uranium streams. The breakdown is shown in Table B-4.

With the exception of the aqueous plutonium stream from the partition column, each aqueous product stream exiting from a cycle was concentrated. All extraction column losses were actual; however, high product content waste was generally recycled.

The basic operating features of this flowsheet were as follows:

- Rigid control of uranium saturation in the HA, 1A, and 2D Columns
- Uranium-plutonium separation by plutonium reduction to the inextractable III valence using ferrous ion (as ferrous sulfamate with sulfamic acid as a holding reductant) in the 1BX and 2DF streams, and an oxidizing step prior to the second plutonium cycle.

TABLE B-3. Feed Preparation Headend Chronology.

Time period	Feed preparation
March 1956	Switch from batch nitric acid addition to continuous acid addition during cut to eliminate ^{131}I surge
September 1957	Backup offgas treatment facility on-line for improved acid recovery and ^{131}I removal
June 1958	T-XB switched from caustic to acid scrubber
June 1958	Prototype silver reactor installed as temporary replacement for original
June 1958	NH_3 scrubber installed
March 1960	Discontinued feed centrifugation (free centrifuges and E cell tankage for FP recovery)
October 1963	Discontinued feed adjustment
August 1965	Thoria process test. The A3 annular dissolver installed (testing continued through September)
January 1966	B3 annular dissolver installed
January 1966	216-A-36-A crib out of service, distributor extended (became 36-B), a consequence of (1) solution carry-over during dissolution into the NH_3 scrubber catch tank and (2) contamination of NH_3 scrubber waste with dissolver solution via the downdraft condenser and piping. The problem noted with the A3 dissolver was solved on the B3 and subsequent dissolver vessels by raising the elevation of the dissolver vent connection on the dissolver and by installing isolation valves between the dissolver and NH_3 waste tank. In addition, routine condenser and scrubber flushing at the conclusion of the fuel dissolution step was begun.[6]
September 1966	Thoria campaign; C3 annular dissolver installed
January 1967	First test of Zirflex process
February 1968	First use of dissolver insert to reduce annulus (for nuclear criticality prevention) during coproduct processing
December 1969	Eliminated continuous AFAN solution
December 1970	Thoria campaign
January 1971	Leaks were found in all three dissolver cells. Operation was resumed with a new B3 dissolver and the A3 dissolver coil discharge routed to a backup concentrator.
September 1971	New C3 dissolver installed
December 1971	Plate coil on an old B3 dissolver--refurbished assembly installed on A3 dissolver

TABLE B-4. System Breakdown.

Flowsheet designation	Canyon position[a]	Cycle[b]	Product	Function
HA	(H2)	1	CO	Extraction/scrub
HC	(H3)	1	CO	Strip
1A	(J2	2	CO	Extraction/scrub
1BX	(J6)	2	CO	Partition
1BS	(J4)	2	Pu	Scrub
1C	(J7)	2	U	Strip
2A	(L1)	3	Pu	Extraction/scrub
2B	(L2)	3	Pu	Strip
2D	(K2)	3	U	Extraction/scrub
2E	(K3)	3	U	Strip

[a]For example, the H2 position is the second equipment position in H cell.

[b]Carbon dioxide means uranium and plutonium as a coproduct.

The following modifications were made during the first year of hot operation for improved ^{95}Zr-Nb decontamination performance:

- C-type columns were changed from organic continuous to aqueous continuous operation (this minimized carryover of interfacial materials containing high FP activity)
- Uranium feed concentration to the HA column was increased and control of the uranium saturation of the organic phase was tightened
- Feed acidities to the HA, 1A, 2D, and 2A columns were decreased
- HA column temperature was reduced
- 1BX stream acidity was decreased
- Treatment of uranium product with silica gel for additional Zr-Nb decontamination factor was initiated.

Final three-cycle operation improvements consisted of:

- Conversion of 2D column to organic continuous operation and installation of a scrub section cartridge of improved design

9413219.1208

- Treatment of plutonium product in a new, semicontinuous, anion-exchange system (installed in a portion of the regulated shop and designated as N cell)

- Improvements to the solvent treatment flowsheet.

Two-cycle operation was tested in the plant in March 1958 and the following equipment modifications were made in June:

- Bottom-interface HA column installed

- The HC column became a scrub column associated with the HA column

- The 1A column was removed

- The backcycle waste system was established using the HCP concentrator (E-H4) to evaporate aqueous waste streams from the 2D and 2A columns and N cell for recycle to the HA column

- TK-F10 was used to combine waste streams.

The installation of backcycle capability, in turn, permitted tighter (higher) saturation in the 2A and 2D columns with concomitant improvements in product quality.

Subsequent major flowsheet changes through the 1972 plant shutdown were:

- Neptunium recovery on a campaign basis using existing columns (this capability resulted because neptunium accumulates in the backcycle waste system)

- Installation of neptunium recovery equipment (another solvent extraction system plus ion exchange product treatment)

- Installation of improved scrub section cartridges and larger diameter columns

- Installation of a third plutonium solvent extraction cycle to replace the anion exchange system to provide increased reliability, capacity, and safety.

Two continuing programs that merit more detailed attention are: (1) improvements in solvent* and solvent treatment and (2) alternate reductants for partitioning.

---

*The term solvent, as used in this discussion, refers to the mixture of tributyl phosphate (TBP), the actual process solvent, and inert hydrocarbon diluent.

Most of the problems associated with FP decontamination in the PUREX solvent extraction process were attributable to solvent deterioration resulting from radiation exposure. This was one reason for placing column interfaces at the opposite end of the columns from the product stream exit. Gross quantities of FP tended to accumulate at the interface in an interface layer consisting of degraded solvent. Consequently, any entrainment or column upset could result in contamination of the stream exiting the disengaging section containing the interface.

The solvent diluent used had a profound effect on solvent quality. The original diluent was a kerosene-type mixture of saturated and unsaturated hydrocarbons which was, in essence, a specific distillation cut (Shell Spray Base E 2342). In September 1961, this was changed to Soltrol-170, a mixture of highly branched aliphatic hydrocarbons with properties similar to Shell Spray Base. Some initial improvement was noted, but this dissipated after a relatively short time. In February 1966, a switch was made to a normal paraffin hydrocarbon (NPH) mixture (containing m-decane tetradecane). Evaluation of contrasting run periods proved straight chain hydrocarbon superiority in terms of process performance.

Improved solvent treatment processes evolved as well. Initially, aqueous sodium carbonate washing was performed in the 10 and 20 pulse columns, followed by centrifugation to remove entrained aqueous carbonate and solids. Shortly after startup, carbonate washing on a batch basis was started in the 10F pump tank (TK-G1). However, activity in solvent treatment system 1 continued to be a problem. A nitric acid scrub was then used in the 10 column (in place of carbonate) with marked improvement in both 10 column stability and solvent quality. In late 1957, $KMnO_4$ was added to the aqueous wash in TK-G1, a turbo-mixer was installed in receiver TK-G7, and centrifugation of the 10 column product was discontinued. A caustic wash was used with the turbo-mixer. Later, the same changes were made to solvent treatment system 2, the only exception being no turbo-mixer installation in R cell.

Due to declining FP activity in the plant feed (longer cooled reactor fuel), improved FP decontamination in the first cycle as a result of fluoride complexing of ^{95}Zr-Nb during processing of N Reactor fuel (Zirflex Tonic effect), and the stability of the NPH, it was found prior to shutdown that the turbo-mixer operation was not needed. The usage of $KMnO_4$ was reduced by 60% by lowering the concentration in the wash solution by 33% and eliminating the use of a more concentrated $KMnO_4$ add stream to the system 1 wash solution. Further reductions in the Na_2CO_3- $KMnO_4$ usage were made by reusing more of the system 2 wash solution in system 1.

The search for an alternate to ferrous sulfamate as a plutonium reductant was motivated primarily by the waste management program. The iron and sulfates contribute to the total bulk of nonvolatile salt in the waste and adversely affected the waste fractionization effort in B Plant.

Plant tests have been performed with the following:

- Ferrous nitrate stabilized with hydrazine
- Uranium IV stabilized with hydrazine
- Hydroxylamine nitrate (HN)--stabilized with hydrazine
- HN in conjunction with lesser amounts of ferrous ion.

The HN stabilized with hydrazine is used in the 2D column feed and 2B column strip. The neptunium recovery system requires some ferrous ion and holding reductant due to the high nitric acid concentration and the radiolytically produced nitrite in the backcycle. The uranium IV, while demonstrated in the 1BX stream on a test basis, is not readily available for use. The tests with HN in the 1BX stream were successful, but proved too costly. The FeSA-SA system is preferred for nuclear criticality prevention. All other reducing systems require more administrative and engineered safeguards.

Solvent extraction chronology is shown in Table B-5.

WASTE TREATMENT

The waste treatment system at hot startup consisted of two cycles. A primary, high-level, waste concentrator (E-F11) evaporated the waste stream from the first extraction column. A second concentrator (E-F6) received aqueous waste streams from subsequent cycle extraction columns and the condensate from the E-F11-2 and vessel vent system condensers. The overheads from E-F6 were routed through an acid absorber-fractionator (T-F5). Two functions were performed by this system: nitric acid removal and recovery for recycle and waste volume reduction. Minimizing radioactivity in the concentrator overheads and recovering nitric acid were the primary justifications for double distillation of the HAW (HW column waste).

It was recognized during the final plant construction phase that an atmospheric acid absorber-fractionator would limit plant operating rates and have a short service life. An ancillary vacuum fractionator package was, therefore, procured and placed in operation in 1957.

TABLE B-5. Solvent Extraction Chronology. (Sheet 1 of 2)

Time period	Solvent extraction
April to May 1955	Extended scrub portions of HA and 1A columns based on hot semiworks test experience. This required chipping 5 ft of concrete out of the cell floor at the H2 and J2 positions
April 1956	Batch washing in TK-G1 of system 1 solvent initiated
During 1956	C-type columns converted to aqueous-continuous (top interface) operation
November 1956	2D column converted to organic-continuous (bottom interface) operation
May 1957	Semicontinuous anion exchange treatment of Pu product stream (2BP) initiated. Sulfuric acid added to 2BX to permit reduced 2BX flow
December 1957	HA column converted to organic-continuous (bottom interface) operation. Began continuous $KMnO_4$ solvent washing
March 1958	First test of two-cycle flowsheet, bypassing HC and 1A columns. Converted 10 column to dilute acid scrub
June 1958	First attempt to recover Np (campaign basis from 1WW). Implemented two-cycle system
February 1960	Solvent quality deteriorated to the point where it had to be discarded and replaced with fresh solvent
September 1961	Switched from Shell Spray Base diluent to Soltrol-170
October 1962	Began Np recovery on a continuous basis
January 1965	Process test of modified Thorex solvent extraction flowsheet successful
February 1966	Switched from Soltrol-170 to NPH diluent
May 1966	Performed Th campaign
July 1967	First N Reactor fuels processing campaign, "tonic" effect (enhanced ^{95}Zr-Nb decontamination factor) noted; (attributed to F^- in November 1969). Ferrous nitrate/hydrazine tested in 1BX
September 1969	Replaced sulfamic acid with hydrazine (1BX)
October 1969	1UD recycle (to 1CX)
January 1970	Deliberate butting of HAF with fluoride for decontamination factor enhancement

TABLE B-5. Solvent Extraction Chronology. (Sheet 2 of 2)

Time period	Solvent extraction
April 1970	Tested U(IV) as alternate 1BX reductant
May 1970	Tested HN in 2DIS
July to December 1970	Performed second Th campaign
April 1971	Third Pu cycle installed
August 1971	HAO stream eliminated for improved Np recovery. Use of TK-J2 eliminated. Second- and third-cycle organic waste streams routed directly to 1BXF tank (TK-J3)

The most significant change in the waste treatment system consisted of conversion to a single cycle. This change became feasible after converting to the two-cycle solvent extraction system with backcycle waste concentration, reducing the volume to be concentrated in the waste treatment system. In addition, sodium nitrite was added to suppress ruthenium volatilization into the overheads of concentrator E-F6 and a prototype demister was installed in the E-F6 offgas, markedly reducing activity in the overheads.

Another concern was volume reduction of high-level waste due to the high cost of underground storage tanks. Efforts toward this end involved reduction of nonvolatile components (previously discussed in the flowsheet section), increased concentration of high-level waste (eventual four-fold concentrator bottoms flow reduction), and denitration of the high-level waste. The process required alkalinization of the waste with sodium hydroxide prior to storage in the underground, carbon steel waste tanks. After extensive laboratory study, equipment for a process utilizing formaldehyde to destroy nitric acid in the waste stream (1WW) without adding nonvolatile constituents to reduce sodium hydroxide additions was installed and briefly used in the plant. The reaction proved quite violent, causing excessive entrainment of 1WW in the gaseous products, resulting in high activity in the recovered acid. Sugar (sucrose) was tested as an alternative, proving eminently suitable, and was used after October 1963.

Two other topics merit discussion: intermediate-level waste disposal and plant rework of high-level-waste-containing product. Sump wastes and drainage from miscellaneous contaminated areas (i.e., sampler cave drains) are collected in TK-F18. This material may generally be disposed of to either nonboiling or boiling waste storage. If reworked due to high product content, it is processed through TK-F8.

The TK-F8 proved very beneficial to the PUREX process. It is a standard tank equipped with a large steam coil and a condenser with capability to reflux by condensate return to F8 or concentrate by condensate disposal to TK-F7. Most material to be reworked contains some solvent. During prolonged refluxing, the solvent is degraded to nonvolatile phosphates, with the majority of the organic destroyed. Small batches of this rework material may then be blended with feed going to the first extraction column. The F18 and F8 operations have remained essentially unchanged throughout the history of the plant.

High-level waste without associated solvent may be reworked directly without F8 reflux. It has been found that F6 bottoms may be reworked with only occasional process upset, that partially sugar denitrated 1WW cannot be reworked without serious process upset, and that post-sugar/post-high-temperature digestion 1WW can be reworked with some beneficial effects on first-cycle decontamination.

High-level waste treatment and acid recovery chronology are shown in Table B-6.

TABLE B-6. High-Level Waste Treatment and Acid Recovery Chronology.

Time period	High-level waste treatment and acid recovery
February 1957	Vacuum fractionator on-line
September 1958	Tested anion exchange for product recovery from concentrated first-cycle waste
September 1962	E-F11 (evaporator) shut down; single evaporation cycle waste treatment initiated
August 1963	Test denitration of 1WW using formaldehyde
September 1963	Successful sugar denitration of PUREX 1WW
November 1966	First $SO_4^{=}$-free 1WW campaign
November 1967	First condensate recycle system, F5 condensate used as reflux (acid absorber distillate)
March 1971	E-F11 used to concentrate sump material and coil condensate and cooling water

MAJOR EQUIPMENT CHANGES, 1954 to 1972

A chronology of equipment changes is given in Table B-7. Equipment changes preceded plant construction. One of the earliest, significant decisions concerning plant equipment involved redesign of the first-generation concentrator to provide removable steam tube bundles. Based on laboratory information and offsite experience, the tube bundle failure rate due to corrosion was expected (and subsequently found) to be considerably more frequent than originally believed.

TABLE B-7. Chronology of Major Equipment Changes, 1956 to 1972. (Sheet 1 of 2)

Time period	Major equipment changes
Mid-1954	Removable tube bundles on all concentrators
April to May 1955	Went to longer scrub sections on HA and 1A columns. Required deepening cells
November 1955	Pumps placed in E6 to H1, E5 to E6 (jet dilution too great)
May 1956	Unidiameter 1A column installed, replacing L cell package (went to single stripper and a concentrator)
November 1956	T-K2 (2D) bottom interface column with "Zebra" (both stainless steel and plastic plates) scrub cartridge installed
April 1957	Ti tube bundle installed in the Pu product concentrator
November 1957	Bottom interface Zebra scrub cartridge HA installed
January 1958	N Cell Pu anion exchange system installed
February 1958	Replaced silver reactor
March 1958	G7 turbo-mixer installed
June 1958	HC and 1A columns removed; HS column and TK-J2 installed
December 1958	Second 40-ton crane installed
June 1960	Installed new HS, 1C, and 2E column
February 1961	Replaced 2A column (which was a packed column) with one with a cartridge
August 1961	Replaced concentrator tube bundle steam traps on trap pits south of 202-A with in-cell, diaphragm-operated valves
August to October 1962	Installed Np recovery and purification equipment
August 1965	Installed first annular dissolver(3). Floor in cell rebuilt. New silver reactor (two beds) installed in A cell
January 1966	Second annular dissolver installed with modified vent in B cell; some concrete along one wall removed
April 1966	Installed new N6 and N7 vessels (Ta-lined and all Ti, respectively)
September 1966	C3 annular dissolver installed

TABLE B-7. Chronology of Major Equipment Changes, 1956 to 1972. (Sheet 2 of 2)

Time period	Major equipment changes
February 1967	UNH rework concentration and reflux system installed in 203 area (TK-P6)
March 1967	Installed: geometrically favorable 2AF tank (TK-J5), improved N6 demister, greater capacity J4 cartridge (in hot spare column)
May 1967	Two-bed silver reactor in C cell; Zirflex decladding waste treatment system in E cell
June 1967	Unidiameter HA column installed. Also, new specially designed tanks installed in F7 position
April 1968	First use of Cd as neutron poison during rework Pu transfers (L11 to E6) (in solvent extraction)
June 1968	Bypassed HS column, codecontamination cycle decontamination factor dropped by factor of two to three, but was acceptable
July 1968	Replaced N cell resin "pusher" with "puller" and was unsuccessful
June 1969	E-F6 leaked and was replaced. First sectionalized concentrator installation
July 1969	Replaced barometric condenser for the acid fractionator with shell-and-tube unit
November 1969	Installed 1UD recycle system
July 1970	Installed sectionalized concentrators in F11 and H4 positions for Th campaign
December 1970 to January 1971	Replaced L cell package with third Pu cycle
January 1971	All three dissolver coils leaked
March 1971	Replaced B3 dissolver
September 1971	Installed new C3 dissolver
December 1971	Installed dissolver with plate coil in A3 position

9113219.1216

This basic concentrator was succeeded in the 1960s by the sectionalized concentrator which provided a common reboiler compatible with two differing towers: one tower designed for simple evaporation and acid boiloff service, the other for organic stripping (uranium) service. The first sectionalized concentrator to be installed replaced an acid concentrator.

The chronology cited covers most of the major equipment changes at PUREX. In general, the reasons for the changes were:

- Equipment failure with attendant modifications intended to preclude future failures
- Inadequate equipment performance and, generally, increased capacity desired
- Desired improvements in process performance.

9413219.1217

SIGNIFICANT EVENTS, 1956 to 1972

The chronology shown in Table B-8 is a record of significant events occurring from 1956 to 1972.

PUREX PLANT PROJECTS

Table B-9 shows all past projects at PUREX resulting in actual equipment and facility changes.

The nature of the projects changed with time. Initially, most projects were directed toward process refinement. Gradually, emphasis shifted toward increased production. The third phase was directed toward increased plant versatility and safety. A fourth phase was directed toward upgrading the facility to meet more restrictive requirements for environmental discharges and conduct of operations.

FIRE PROTECTION SPRINKLER PROJECTS

Fire protection facilities were originally provided only in the process cells and the laboratory area. The process cells were provided with a manually operated, water-fog, fire protection system and Fireye photoelectric detectors, and the laboratory area was provided with a fire detection system. In 1967, an automatic sprinkler system was installed in the laboratory area and provided a laboratory storage area equipped with an automatic sprinkler system (Project IAP-605). In 1968, supervised automatic sprinkler systems were installed in the following areas (Project IAP-615): west end of first floor service area up to and including the main entrance lobby and airlock on the east; west transformer substation, dock storage

area; regulated and hot shops, including airlock; N cell airlock; and the east end of the storage gallery. This project also provided fire detectors in the west switchgear and cable rooms, and the existing paint storage area fire detectors in the 2714-A building were connected to the PUREX alarm circuit. In 1974, the light-water, aqueous, film-forming-foam sprinkler system was installed in process cells G, H, J, K, and R and TK-R1A under Projects HCP-678 and B-115. The dissolver drown tanks were seismically upgraded to assure structural survival in the event of an earthquake under Project B-336.

TABLE B-8. Chronology of Significant Events. (Sheet 1 of 3)

Time period	Event
February 20, 1956	Blowback of contamination from L package-pressurized stripper to force overflow with a manometer[7]
April 7, 1956	Partition failure due to NO_2^- recycle
June 2, 1956	First U product met specifications
February 13, 1958	Silver reactor explosion[8] (returned to service August 28, 1958)
February 23, 1958	Fire in hot shop (oil or acid soaked rags); minor loss of contamination control in the hot shop area
March 24, 1959	Seepage from A sump in F cell into storage gallery roped off and covered with blocks for shielding; rebuilt cell floor
January 10, 1960	PR can pressurization (lacked vent hole in gasket). One small loss of contamination control in the PR area
February 25, 1960	Fire in N cell (welding)
April 3, 1961	30,000 gal of 57% HNO_3 drained to chemical sewer due to error in valving
April 20, 1961	1,335 lb of depleted U drained to crib due to valve leak
January 1962	F11 tube bundle; changeout resulted in major loss of contamination control in the canyon area
March 3, 1962	13,600 gal of water on floor in K cell. Loss of contamination control in the PR room
June 3, 1963	Caustic added to uncovered Al-clad fuel produced H_2 explosion. No significant equipment damage
August 1963	Charge of insufficiently aged fuel to C3. Dissolver idled for 2 mo

TABLE B-8. Chronology of Significant Events. (Sheet 2 of 3)

Time period	Event
April 9 to 19, 1964	Main exhaust filter differential pressure increased rapidly to 14 in. WG due to pluggage by ammonium nitrate. New exhaust filter installed
June 14, 1964	TK-F15 coil leaked; approximately 10,000 Ci discharged to pond. Routings changed so that discharges of most coils exposed to high-activity solutions are to steam condensate crib
February 1966	A-36-A crib overloaded with radionuclides (new dissolver); switched to 216-A-36-B
September 23, 1966	PR can tipped over on PR elevator; minor loss of contamination control to the PR elevator area
May 10, 1968	UNH truck leaked from PUREX to UO_3; minor U contamination
May 15, 1968	Overflowed cask in 212-A; no loss of contamination control in the 212-A facility
December 1969	Dumped a bucket of Zircaloy-clad N Reactor fuel on floor; fuel elements retrieved
May 1970	Approximately 14,000 Ci discharged to 216-A-36-B crib due to leaking valve in scrubber drain to catch tank
August to December 1970	Stack emission during Th processing[9] primarily due to Ru volatilization
January 3, 1971	First dissolver coil leak (C3) discovered
March 1971	All three dissolver coils subsequently found to leak
April 8 to 22, 1971	Pu polymer found in third Pu cycle due to omission of HNO_3 to 3BP during startup operability testing
October 1971	Spilled a canister of N Reactor fuel on B cell floor; picked some up, covered rest with water (water leaked out expansion joint and contaminated the storage gallery)
December 4, 1971	KOH inadvertently added to C3 dissolver. As a result, Pu was precipitated as a Pu polymer. U feed solution reacidified with HNO_3, digested at an elevated temperature for an extended time, and processed as normal feed
January 1972	High PR stack Pu emissions (L9 sampler pressurized tank)

TABLE B-8. Chronology of Significant Events. (Sheet 3 of 3)

Time period	Event
February 23, 1972	Greater than 500 g Pu accumulated in F11[10]
April 3, 1972	Transferred excessive Pu to TK-D4[11] (exceeded total Pu limit specifications)
April 11, 1972	Sugar denitrated a waste batch with greater than 811 g of Pu[12]
June 26 to July 17, 1972	14 of 19 charges exceeded criticality prevention limits for spike fuel due to change in canister load size. One deliberate excessive charge also made[13]

9413219.1220

TABLE B-9. PUREX Projects. (Sheet 1 of 5)

	Project	Description
CA-513-A	PUREX facility	Design and construction of PUREX separations plant with appurtenant tank farm and required outside facilities to process production metal fuel
CA-513-C	Three-column prototype with concentrator	Built in 300 Area to test large-size pulse columns and concentration step in the PUREX process
CA-513-E	PUREX Plant modifications	Addition of a spare 40-ton crane, and conversion to 2-cycle process
CG-598	Vacuum fractionator	Auxiliary structure to PUREX, enclosing vacuum acid fractionator and associated equipment
CG-647	Backup (radioiodine) removal	293-A facility
CG-683	Reroute PUREX cooling water to new pond	216-A-271 diversion box to pond 216-A-25 retention basin
CG-686	In-line monitoring instruments, REDOX and PUREX	Provide equipment for in-line monitoring. PUREX installation and modifictions of gamma and alpha monitors and an automatic air cutoff. Iodine--A, B, C, and F 293-A. Gamma--G5, H3, J5A, and K3. Alpha--L1, L8, and 2BP
CG-742	HA column replacement--PUREX	Provide an HA column of new design to replace the existing column. The new column to be designed for a 4.0 capacity. The project includes burial of the old column and additional instrumentation and piping
CG-766	Additional ventilation for PUREX P&O gallery	Provide additional ventilation facilities, including a new blower, new ductwork, associated controls, and other minor equipment
CAC-140	PUREX ventilation filter replacement	Provide replacement for PUREX ventilation exhaust filter
CAC-152	PUREX analytical laboratory additions	Provide alteration and an addition to existing facilities
CAC-895	Reliability improvements 3.6 capacity factor, PUREX	D3/D4, demineralizer anion column, formaldehyde (TK-40)

9113219.1221

TABLE B-9. PUREX Projects. (Sheet 2 of 5)

	Project	Description
CAC-964	New burial tunnel	Construction of new tunnel for retrievable storage of failed equipment
CGC-124	Increased processing flexibility--provide facilities for processing of natural, E metal, D metal, N Reactor fuel elements, and PRTR fuel elements	New J5, F7/F26, sump modifications for nuclear safety
CGC-172	Pu concentration and storage facilities--L cell--PUREX	Provide modifications to the existing L cell package. New L11, safe for ^{233}U
CGC-773	Interim waste crib--PUREX	Provide a new crib with associated piping and equipment for 200 East Area for disposing of condensate from PUREX boiling wastes
CGC-821	Project birch--PUREX Np	Provide in-canyon and gallery facilities in the PUREX Plant for recovery and decontamination of Np as a nitrate salt in aqueous solution from the PUREX waste streams
CGC-854	Conversion of PUREX anion exchange prototype to a manufacturing unit--N cell	Provision of a control room enclosure, access air lock and concrete radiation shielding wall, new improved feed pump system, and associated work to convert the facility to a manufacturing unit
CGC-873	FP loadout facility--PUREX	Provide an FP loadout facility on the south side of the PUREX building for transferring material from existing in-canyon vessels to suitable shipping containers (212-A Building)
CGC-891	PUREX 216-A-6 replacement (process condensate)	216-A-10 crib
CGC-929	NPR fuels processing annular dissolvers	Provide separation facilities for processing fuels from the new production reactor
CA-930	Secondary containment of dissolver offgases	Provide backup filters for each dissolver system in PUREX and REDOX facilities

9413219.1222

TABLE B-9. PUREX Projects. (Sheet 3 of 5)

	Project	Description
IAP-600	PUREX 203-A UNH rework storage facility	TK-P6 installation in 203-A
IAP-601	Coproduct demonstration processing facilities, 200 Area	Coproduct insert
IAP-606	PUREX process condensate crib, 216-A-38	Replace A-10
IAP-608	PUREX D cell in-line sample station	D5 cave and canyon recirculation system
IAP-610	PUREX steam condensate crib, 216-A-37-1	A-30 replacement
IAP-615	Fire protection facilities, PUREX	Hot shop, regulated shop, service, and storage gallery
ICE-628	Third plutonium solvent extraction cycle, PUREX	Replace existing stripper concentrator with a third SX cycle, stripper-concentrator, and associated tanks
HCP-677	PUREX condensate bypass and pump system, 200 East Area	A pump tank and new run of piping (including a section over the railroad tunnel)
HCP-678	Improved fire protection, PUREX	Upgraded in-plant protection system
HCP-685	PUREX NH_3 scrubber, Waste Concentration Facilities	F11 receives NH_3 scrubber wastes
B-107	PUREX condensate recycle	Recycle AAD and AFD with excess to TK-F10, 3WD combination with 1UD
B-112	Waste management effluent control	Second-stage filtration--PR stack; filtration of R cell and U cell exhaust; upgrade sampling capability, retention basin for cooling water. Install third PUREX filter
B-115	Additional fire protection, PUREX Plant	J, K, L, and R cells on fire-foam system. New sprinkler system in ventilation tunnel
B-128	PUREX waste routing to 214-A-102 and -103 tanks	New waste routing
B-129	Piping from PUREX 241-AX-151 Encasement to 241-A valve pits	Route alkalinized and dilute wastes to 242-A system

TABLE B-9. PUREX Projects. (Sheet 4 of 5)

	Project	Description
B-132	PUREX NH_3 offgas system	Ties E cell scrubber offgas to F11
B-151	PUREX safeguards	Guardhouse and vehicle inspection equipment
B-153	PUREX Plant fire protection	Upgrades portions of facility to national fire safety codes
B-172 B-172A	Hanford plant security inspections, 200 Areas	Tie-in PUREX alarm systems with central computer, improve communications network. New perimeter fence
B-175	Plutonium oxide conversion facility	Provides facility in N cell to convert Pu nitrate to PuO_2 product
B-175A	PuO_2 rework	Provides batch equipment to dissolve off-specification PuO_2 and other recoverable material for rework in the PUREX process
B-188	NH_3 scrubber waste sampling system	Adds sampling system to NH_3 scrubber waste discharge for environmental monitoring
B-189	Primary stack systems	Adds sampler monitor and flow totalizing systems to the primary PUREX stack (216-A-1)
B-190	Chemical sewer systems	Adds monitor and flow diversion systems for effluent control
B-191	PR room stack systems	Adds sampling, monitoring, and flow totalizing systems for effluent monitoring
B-196	Pollution abatement	Adds H_2O_2 injection system to reduce NO_x emissions from the primary ventilation stack (291-A-1)
B-201	PUREX safeguards improvements	Building perimeter fencing, lighting, intrusion detection
B-203	Process condensate systems	Adds improved sampling, monitoring, and flow totalizing capability to the process condensate waste discharge
B-205	PUREX P&O gallery services upgrade	New cable trays and cable separation in P&O gallery
B-220	Waste transfer facilities, 200 Areas	Additional PUREX acid waste and neutralized high-level waste lines routed to PUREX

9413219.1224

TABLE B-9. PUREX Projects. (Sheet 5 of 5)

	Project	Description
B-254	UNH storage area effluents	Adds collection, sampling, and selective disposal routing for environmental control
B-268	Pond 216-B-3 expansion	Expands pond 216-B-3 to receive increased flow of cooling water from PUREX
B-280	Additional waste lines 202-A to 244-AR	Install PUREX high-level waste lines
B-281	Additional waste lines 202-A to 241-AW	Install PUREX high-level waste lines
B-311	HVAC centralized control	Adds control room instruments for centralized control of ventilation systems 1 and 2
B-313	PUREX perimeter intrusion detection system addition	Provides a second intrusion detection and alarm system for PUREX
B-336	PUREX seismic upgrade	Reinforces dissolver drown tanks to assure structural survival in the event of an earthquake. Adds backup light water foam fire protection system for Cells H and J
B-346	HVAC pressure zone	Provides airlocks to segregate 202-A facility into four distinct pressure zones and provides the differential pressure sensors/alarms to monitor those pressure zones
B-379	White room filtration system	Adds HEPA filters and control instruments to divert airflow in the event of a leak in the P&O gallery
B-382	PUREX backup generators	Adds motor-generator sets and electrical switchgear to provide backup power
B-410	Lag storage tank	Adds additional tank for Pu nitrate storage

9413219.1225

REFERENCES

1. GE, PUREX Technical Manual, HW-31000, Table B-2, Volumes I and II, General Electric Hanford Atomic Products Operation, Richland, Washington (1955).

2. GE, Description of PUREX Plant Process, HW-60116, General Electric Hanford Atomic Products Operation, Richland, Washington (April 30, 1959).

3. Rockwell, PUREX Technical Manual, RHO-MA-116, Rockwell Hanford Operations, Richland, Washington (1980).

4. "Reactor Handbook," Volume II, Chapter IV, 2nd Edition, Interscience Publishers (1961).

5. Rockwell, PUREX Flowsheet-Reprocessing N Reactor Fuel, PFD-P-020-00001, Rockwell Hanford Operations, Richland, Washington (September 20, 1982).

6. Isochem, PUREX Annular Dissolver Modifications, ISO-147, Table B-3, Isochem, Richland, Washington (March 7, 1966).

7. GE, Blowback of Contamination from L Cell, HW-41701, General Electric Hanford Atomic Products Operations, Richland, Washington (1956).

8. GE, PUREX Plant Silver Reactor Incident, HW-55223, General Electric Hanford Atomic Products Operations, Richland, Washington (1958).

9. ARHCO, PUREX Process Operation and Performance 1970 Thoria Campaign, ARH-2127, Atlantic Richfield, Richland, Washington (1977).

10. ARHCO, Criticality Prevention Infractions - Low Acidity in Plutonium Containing Waste Concentrator, ARH-2428, Table B-7, Atlantic Richfield Hanford Company, Richland, Washington (March 31, 1972).

11. ARHCO, Criticality Specification Violation - Exceeding 10,000 grams Plutonium Limit in TK-D4, ARH-2466, Atlantic Richfield Hanford Company, Richland, Washington (April 25, 1972).

12. ARHCO, Sugar Denitration of Zirflex High-Level Waste, ARH-2340, Rev. 1, Table B-7, Atlantic Richfield Hanford Company, Richland, Washington (April 25, 1972).

13. ARHCO, Criticality Specification Violation - Exceeding 12,000 lbs Uranium Charge Limit to PUREX Dissolvers, ARH-2575, Table B-7, Atlantic Richfield Hanford Company, Richland, Washington (April 25, 1972).

9413219.1226

9413219.1227

PUREX FSAR

APPENDIX C PUREX OPERATIONAL FLOW DIAGRAM

9413219.1228

This page intentionally left blank.

APPENDIX C

PUREX OPERATIONAL FLOW DIAGRAM

The objectives of a preliminary hazards analysis (PHA) are to identify potential hazardous conditions inherent within the system and to determine the significance of potential accidents that might arise.

The first step in a PHA is to identify potentially hazardous elements or components within the system. This process is facilitated by an organized approach to examining system components, engineering experience and the exercise of engineering judgment.

This first step was initiated by the development of a system model which allowed organized system/component evaluation, e.g., a process description, drawn as a Plutonium Uranium Extraction Plant (PUREX) operational flow diagram. The purpose of this flow diagram was to allow methodical analysis of the accident potential of the PUREX process steps. This flow diagram is presented in Figures C-1 through C-19. Figure C-1 is the flow diagram for the overall process. Figures C-2 through C-19 are each amplifications of the blocks in Figure C-1. The notation is such that the leading digit refers to the major process category and subsequent digits further identify the specific operations. Chemicals associated with the process steps are indicated above the steps in Figures C-2 through C-19.

The second step in the preparation of the PHA was the identification of those events that could transform specific hazardous conditions into potential accidents. Then the seriousness of the potential accident was assessed to determine the adequacy of plant control features. This was accomplished in the PUREX PHA in a series of round table discussions involving several senior Rockwell Hanford Operations personnel. Each section of the flow diagram was evaluated by the group for potential accidents by examining for energy sources such as flammable solvents, steam, hydrogen, radiation, etc.; the presence of adequate safety systems such as High Efficiency Particulate Air filters, engineered and adminis-trative barriers; and potential consequences.

The format of this PHA followed closely that used in previous analyses, including a PHA done by Pacific Northwest Laboratories[9-1] and an unissued probabilistic risk assessment performed by Science Applications, Inc. Both of these studies were reviewed by Rockwell and incorporated in the final PHA as appropriate.

Each block in the operational flow diagram with accident potential was selected for more careful consideration in the PHA presented in Chapter 9.0. The number in the extreme left column of the PHA identifies the process block in this appendix from which the potential hazard was identified.

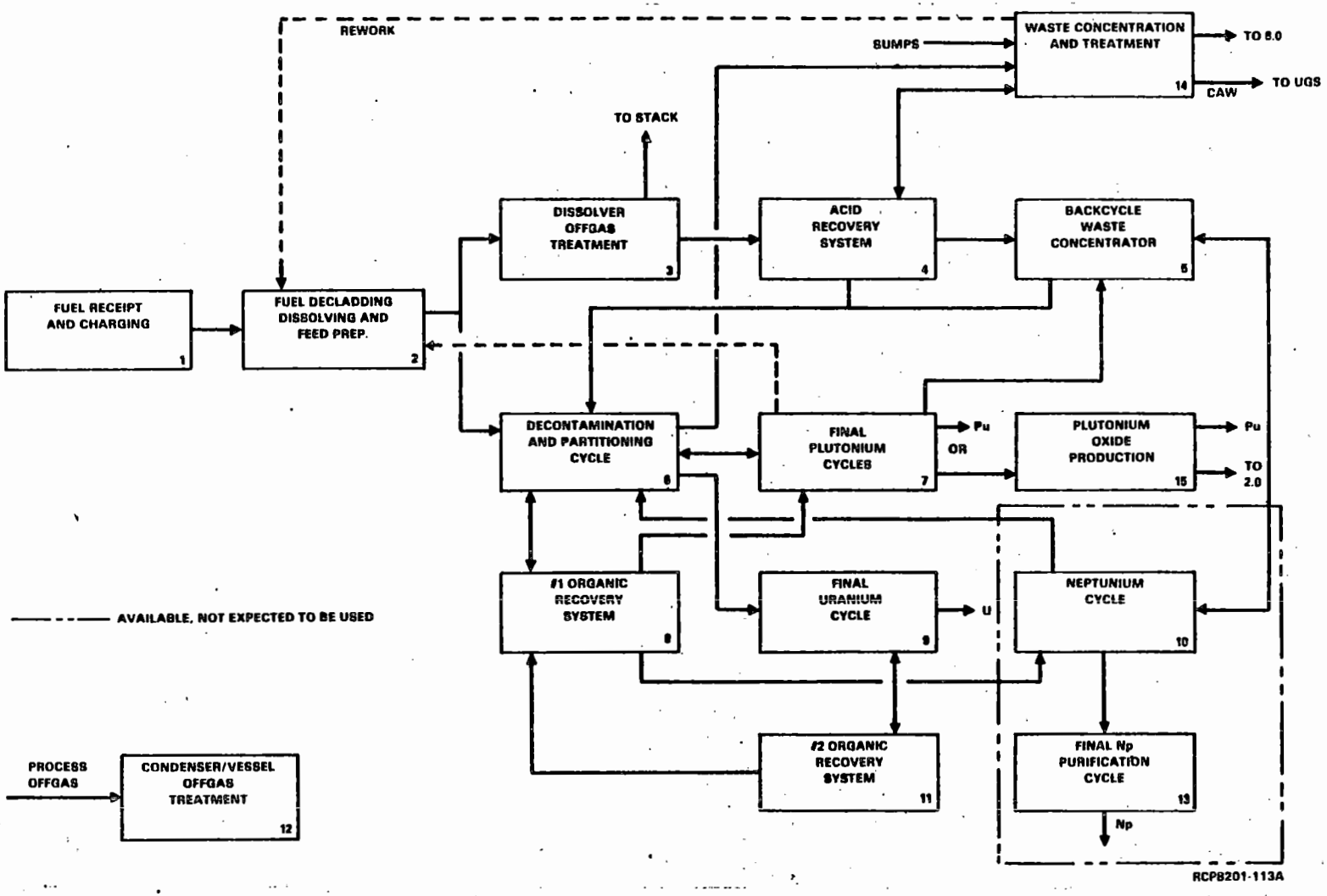


FIGURE C-1. PUREX Process Overview.

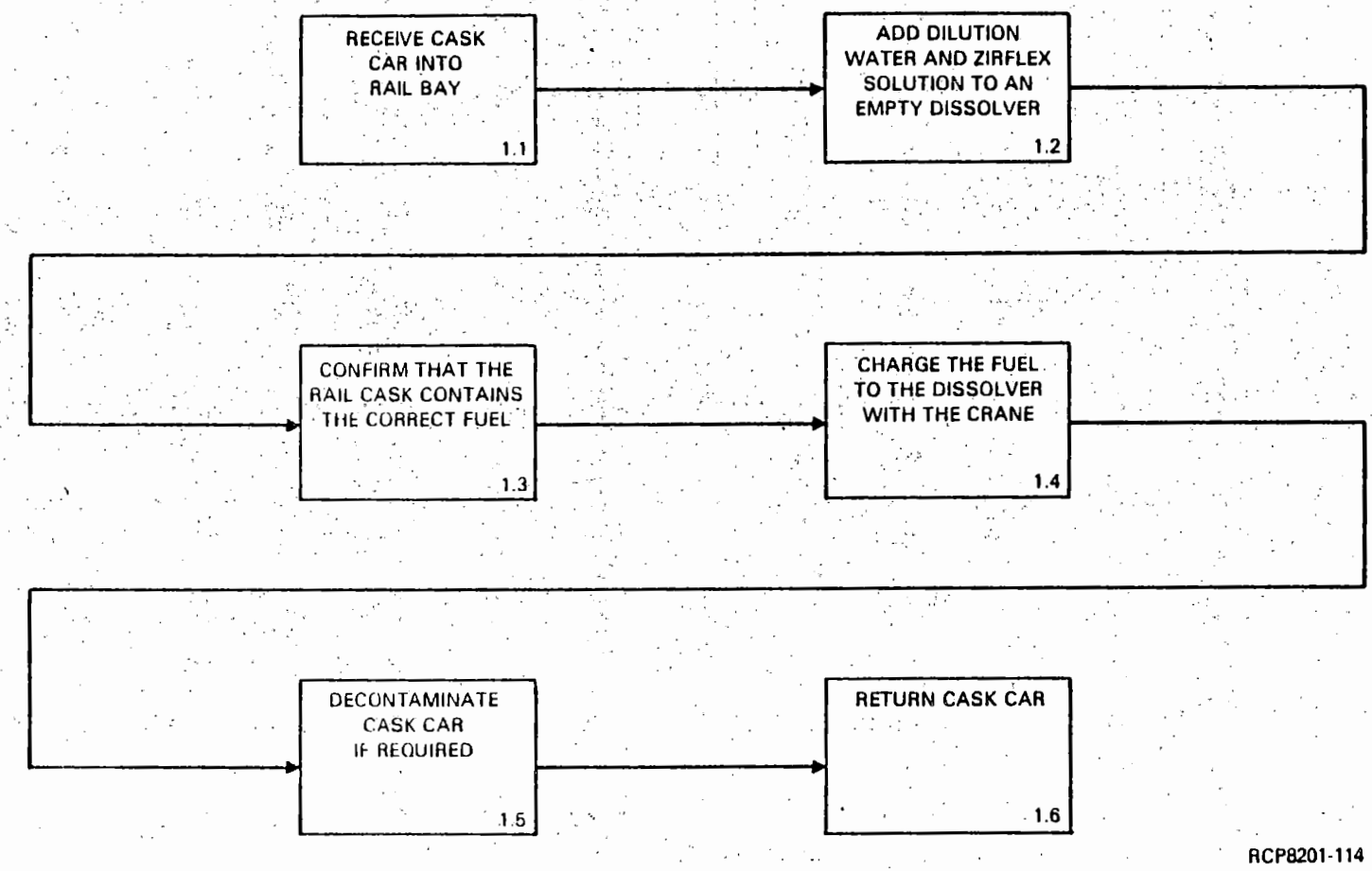


FIGURE C-2. Fuel Receipt and Charging.

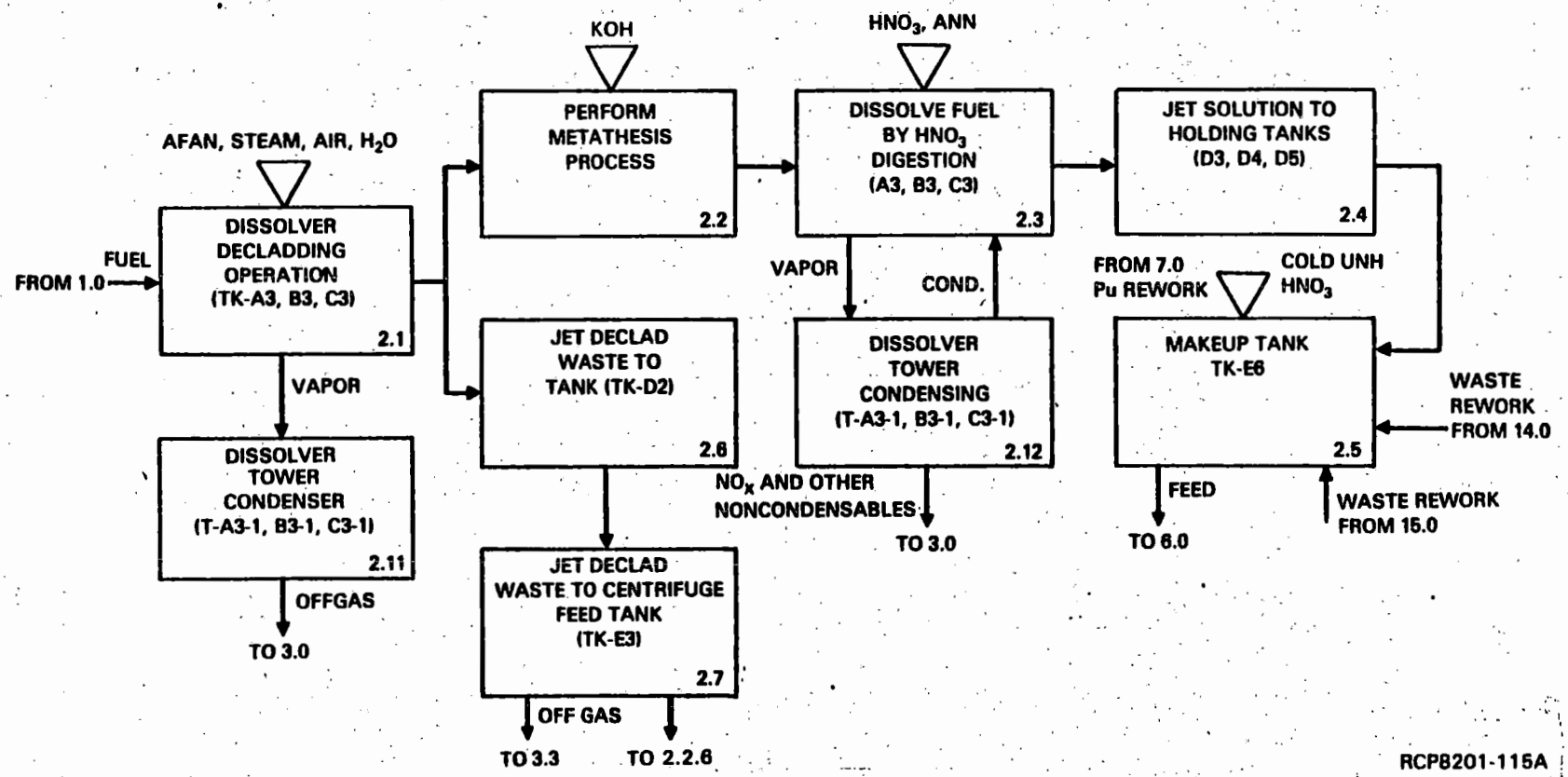


FIGURE C-3. Fuel Decladding and Dissolution. (Page 1 of 2)

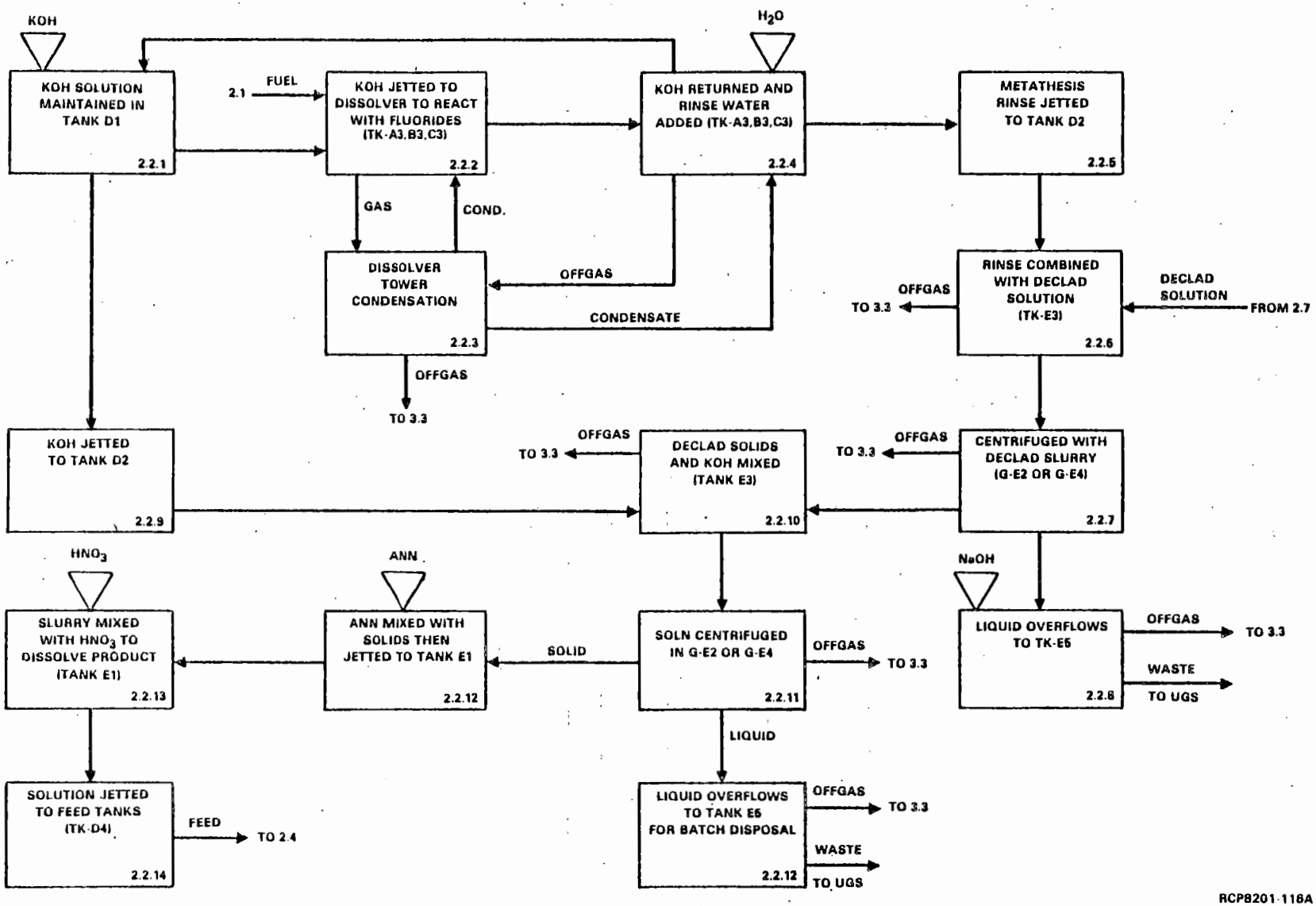


FIGURE C-4. Fuel Decladding and Dissolution. (Page 2 of 2)

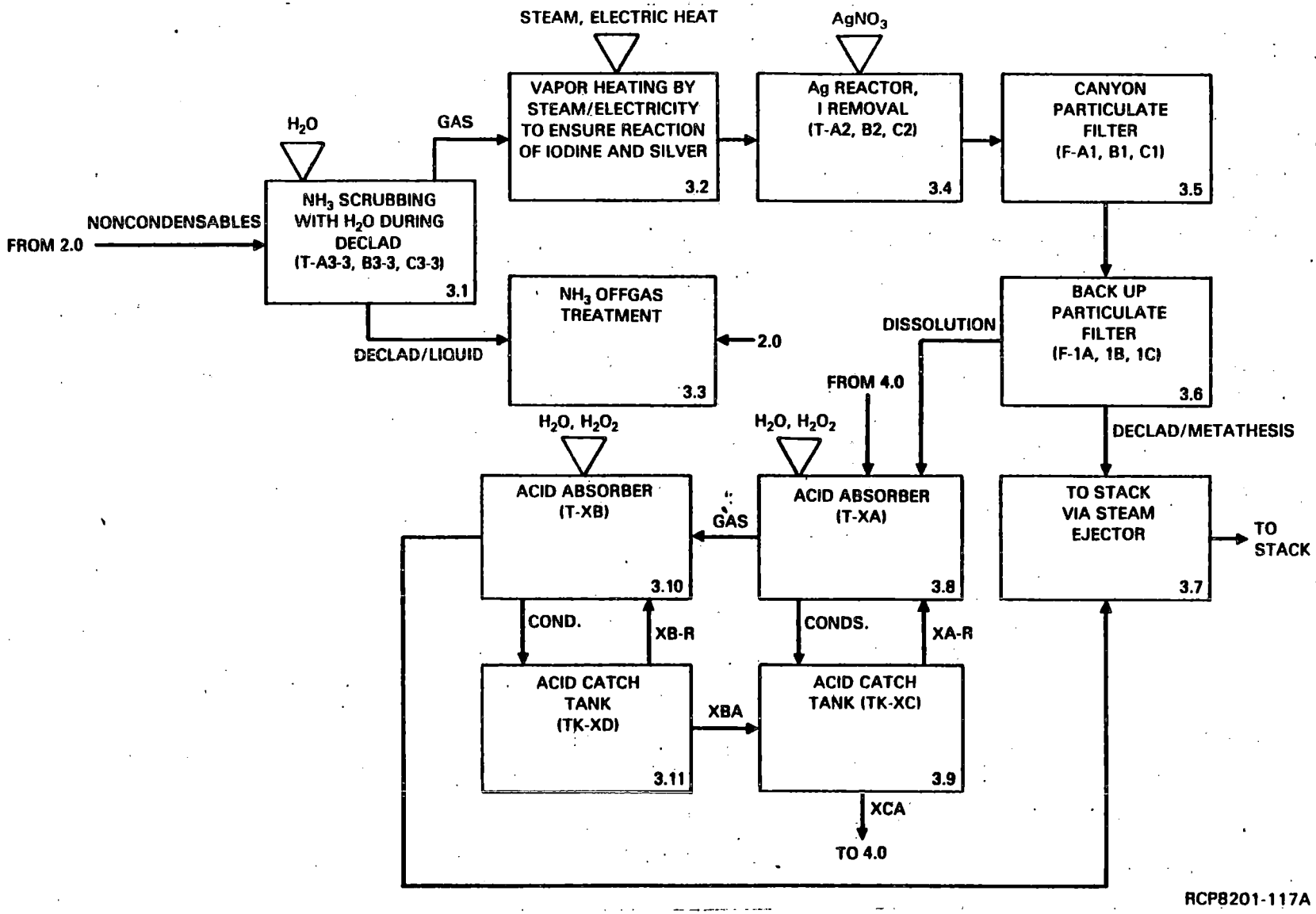


FIGURE C-5. Dissolver Offgas Treatment.

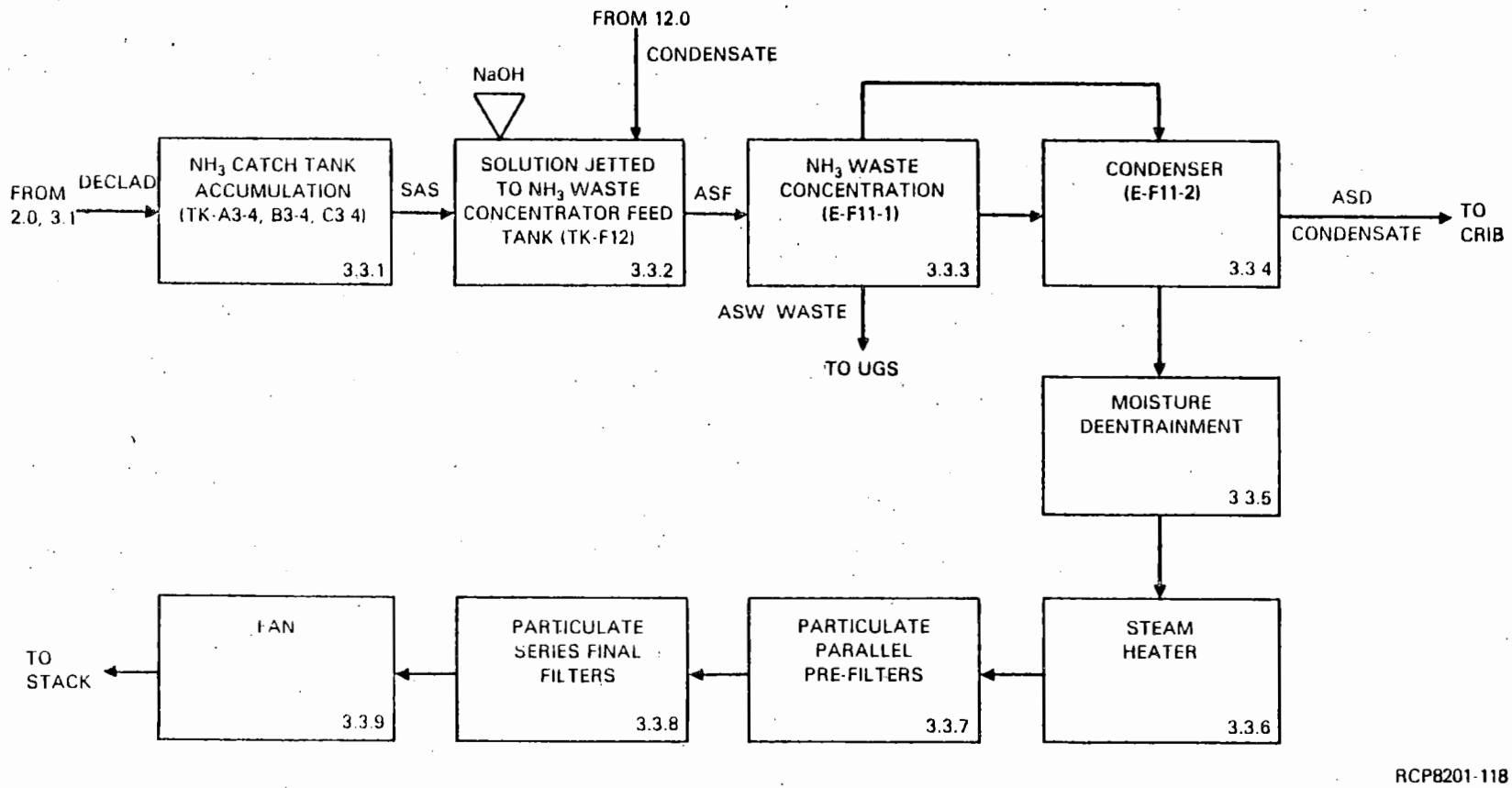


FIGURE C-6. NH_3 Offgas Treatment.

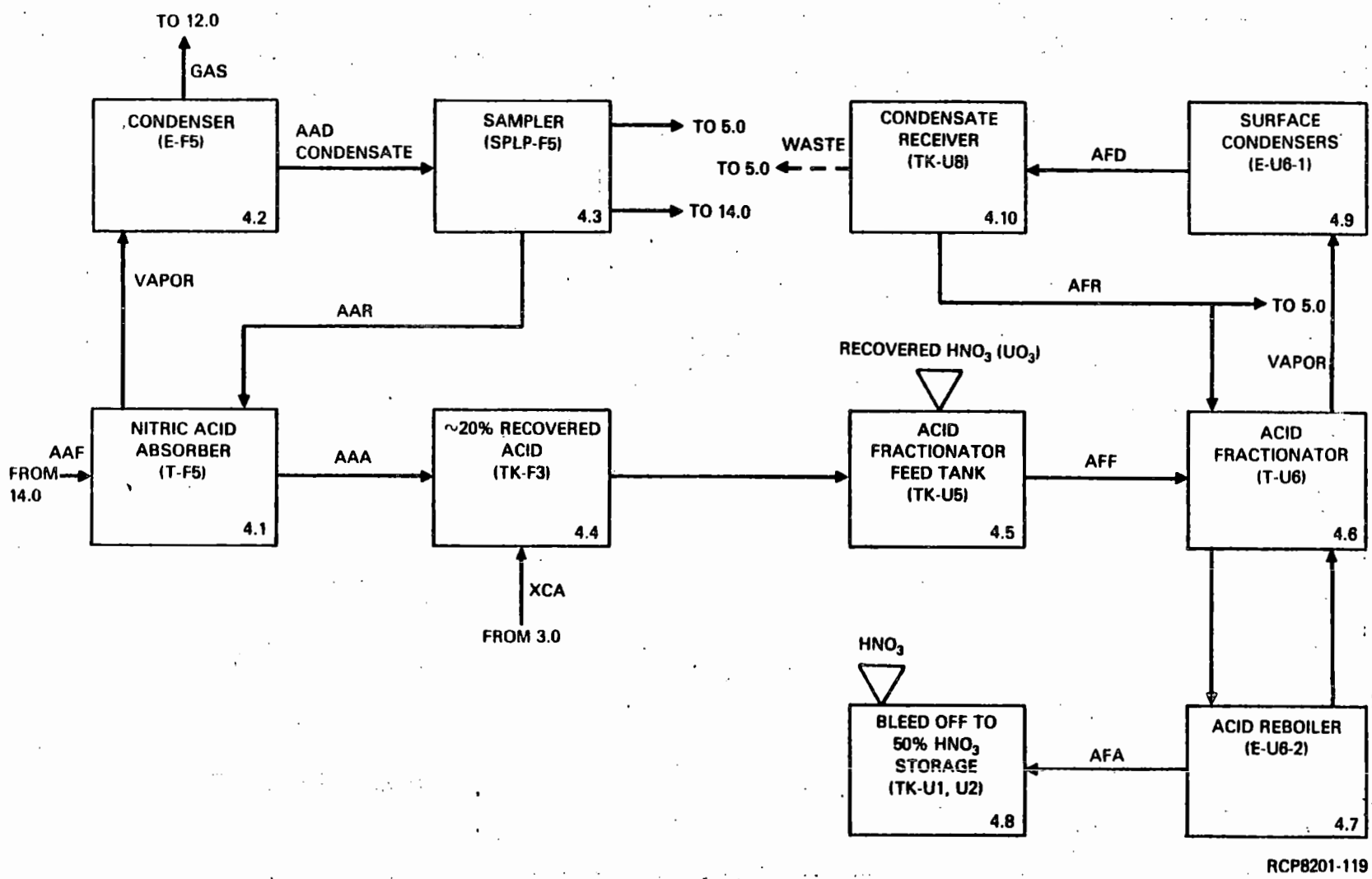


FIGURE C-7. Acid Recovery System.

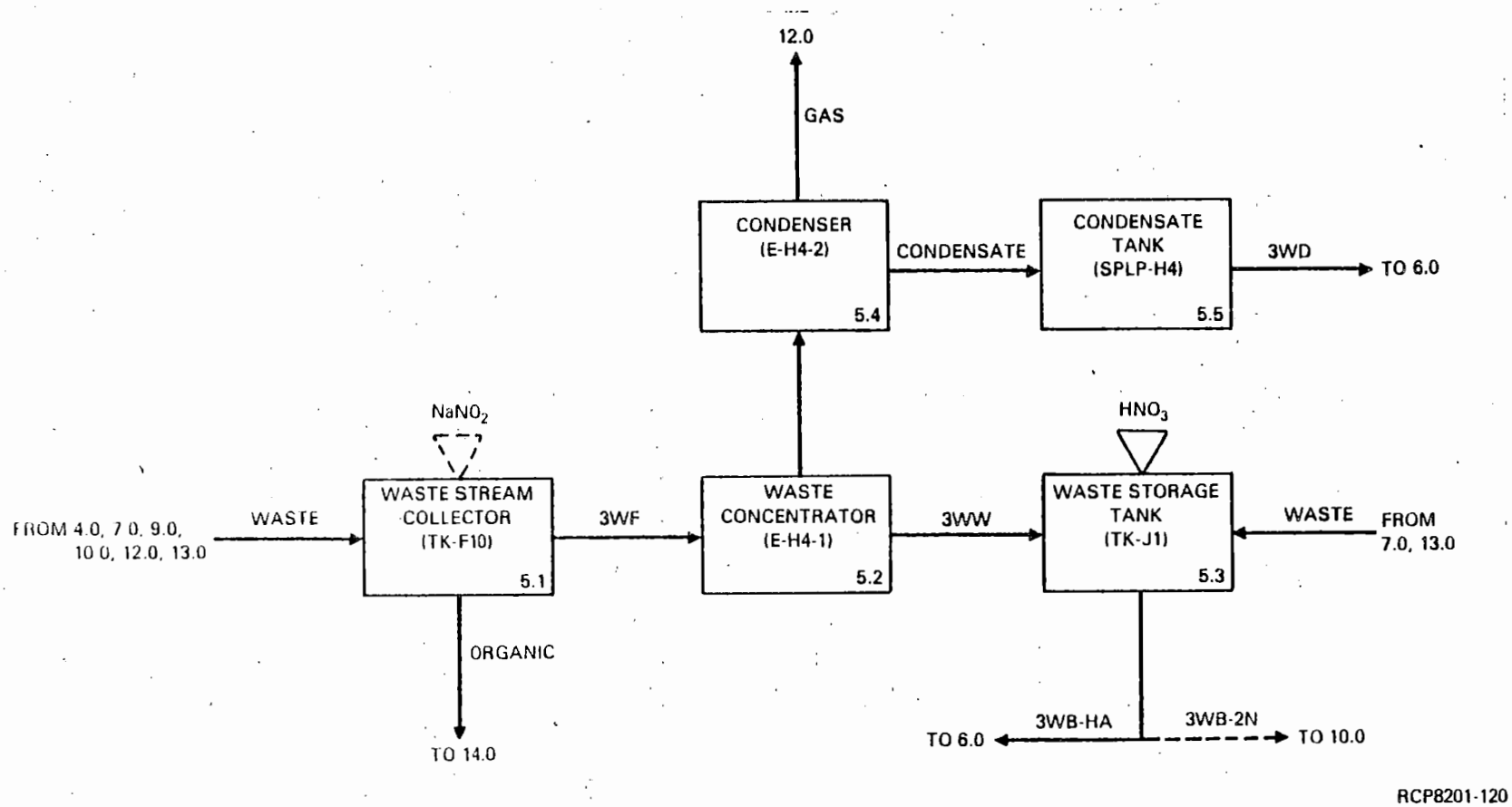


FIGURE C-8. Backcycle Waste Concentrator.

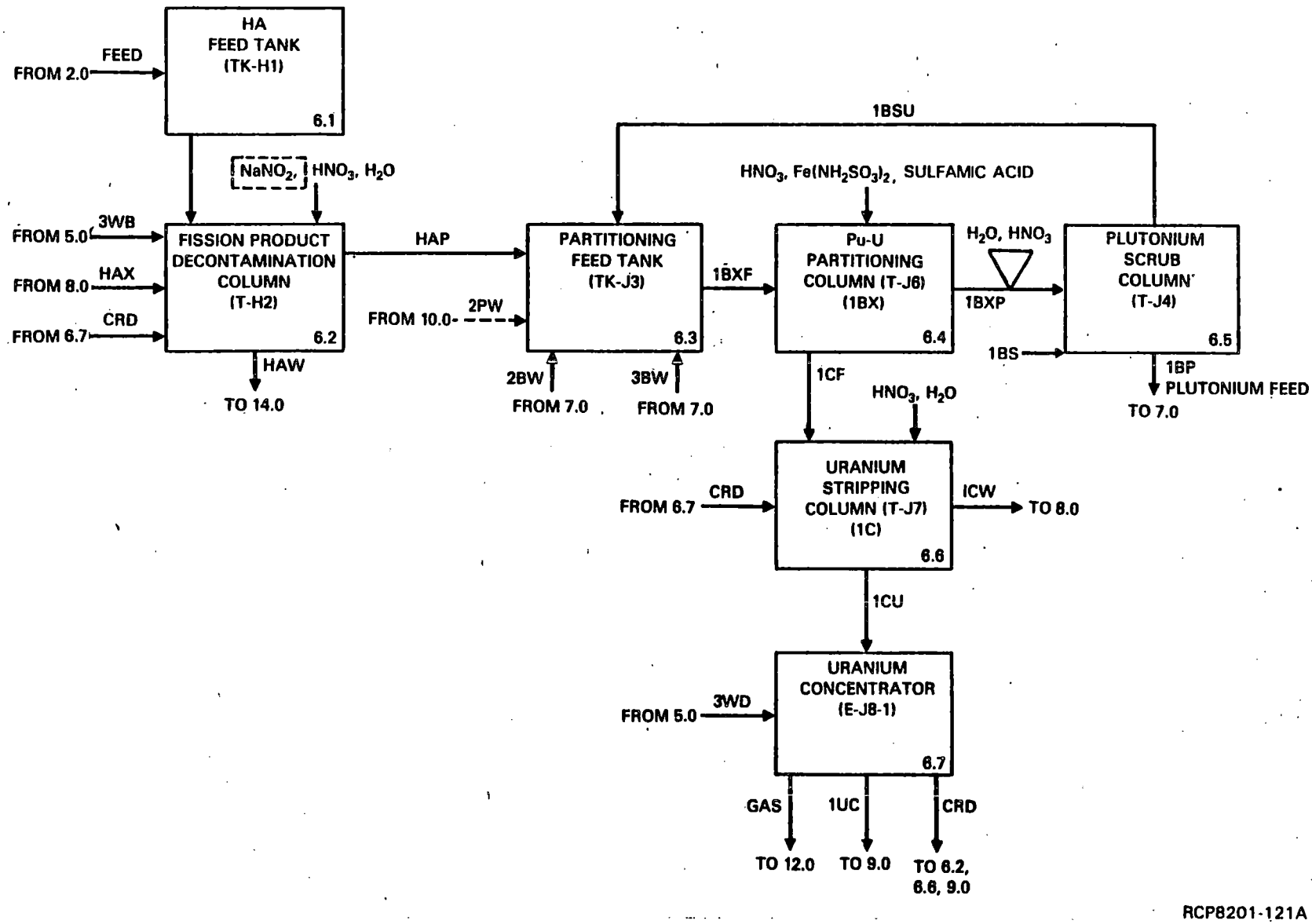


FIGURE C-9. Decontamination and Partitioning Cycle.

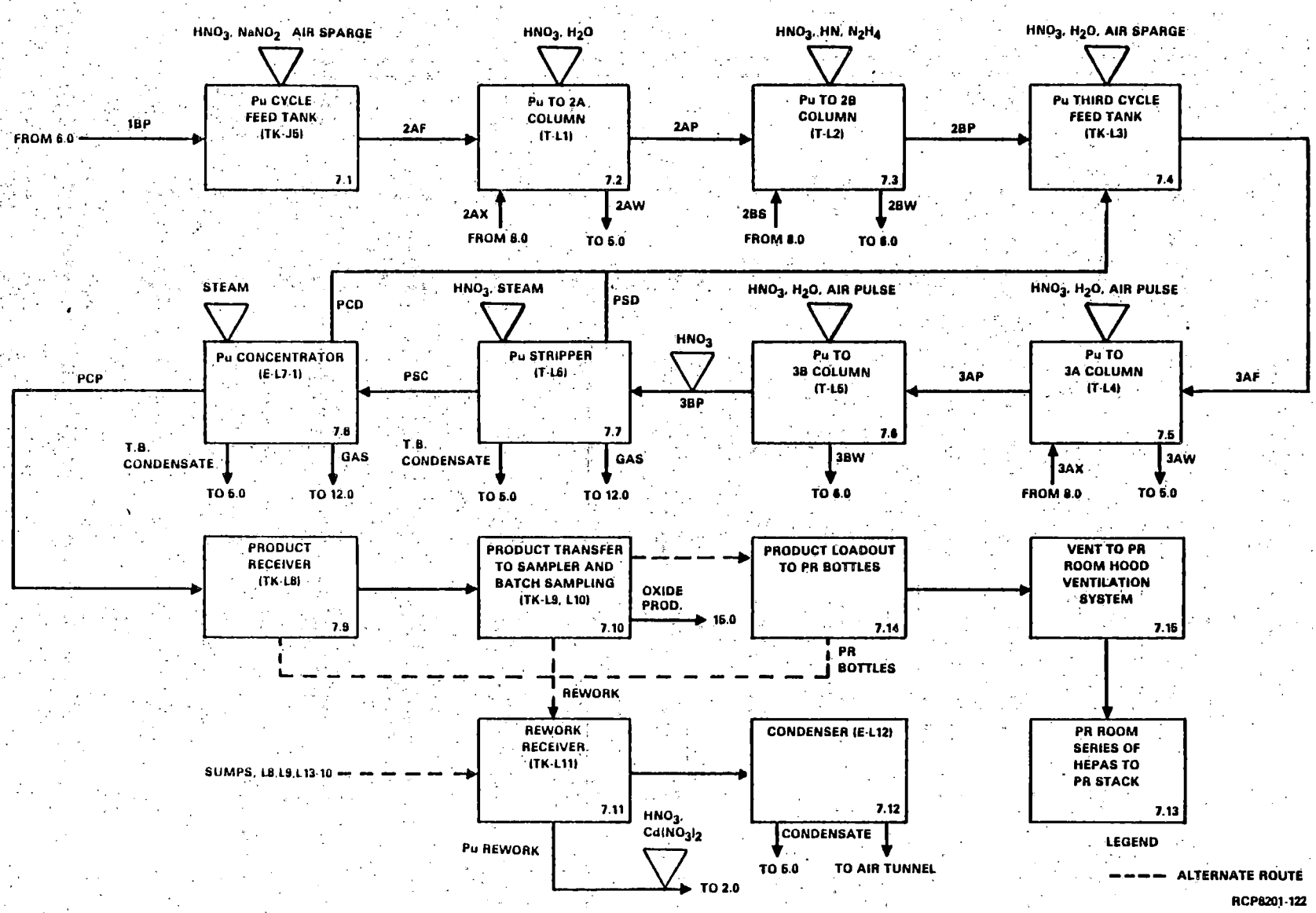


FIGURE C-10. Final Plutonium Cycle.

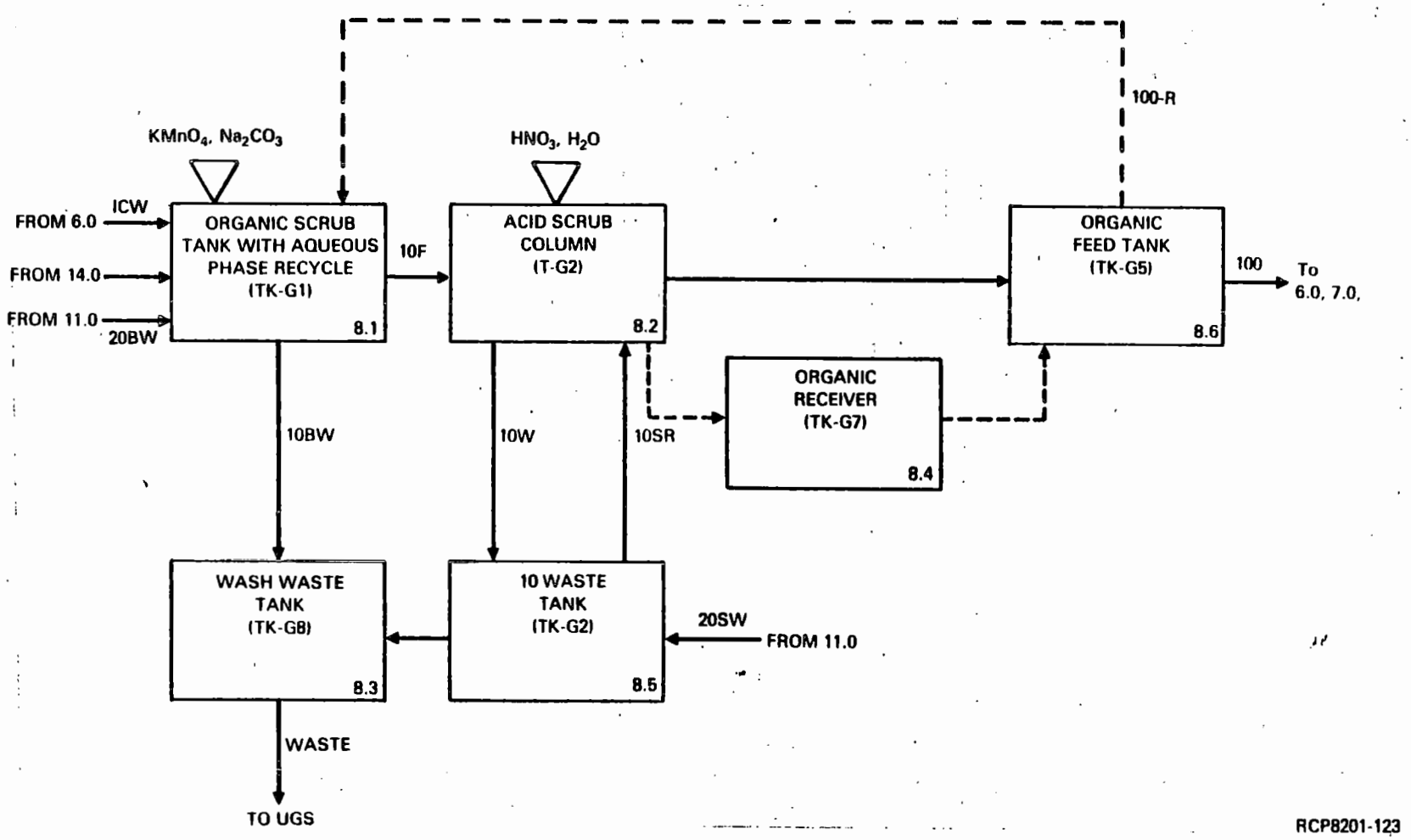


FIGURE C-11. #1 Organic Recovery System.

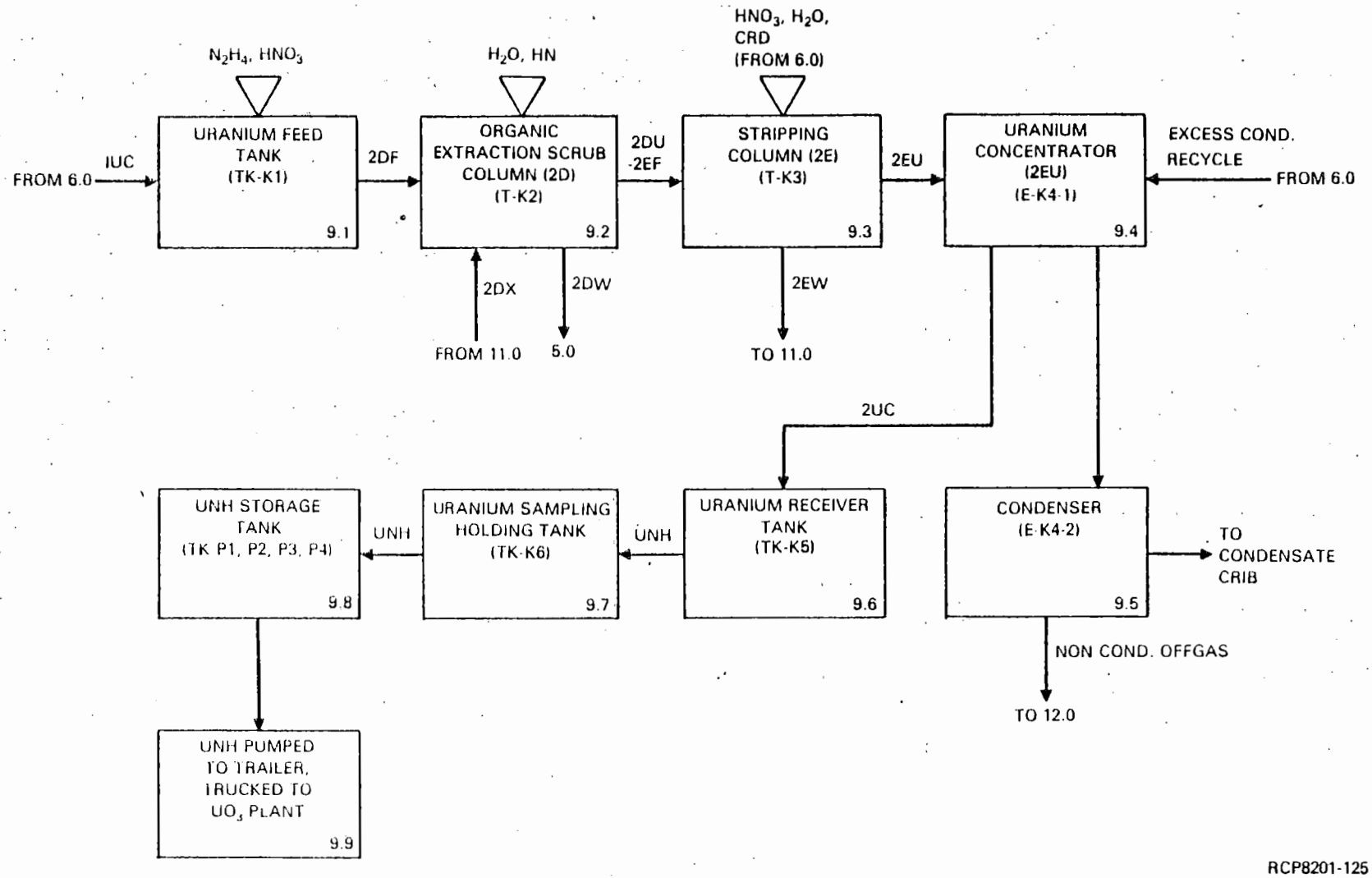


FIGURE C-12. Final Uranium Cycle.

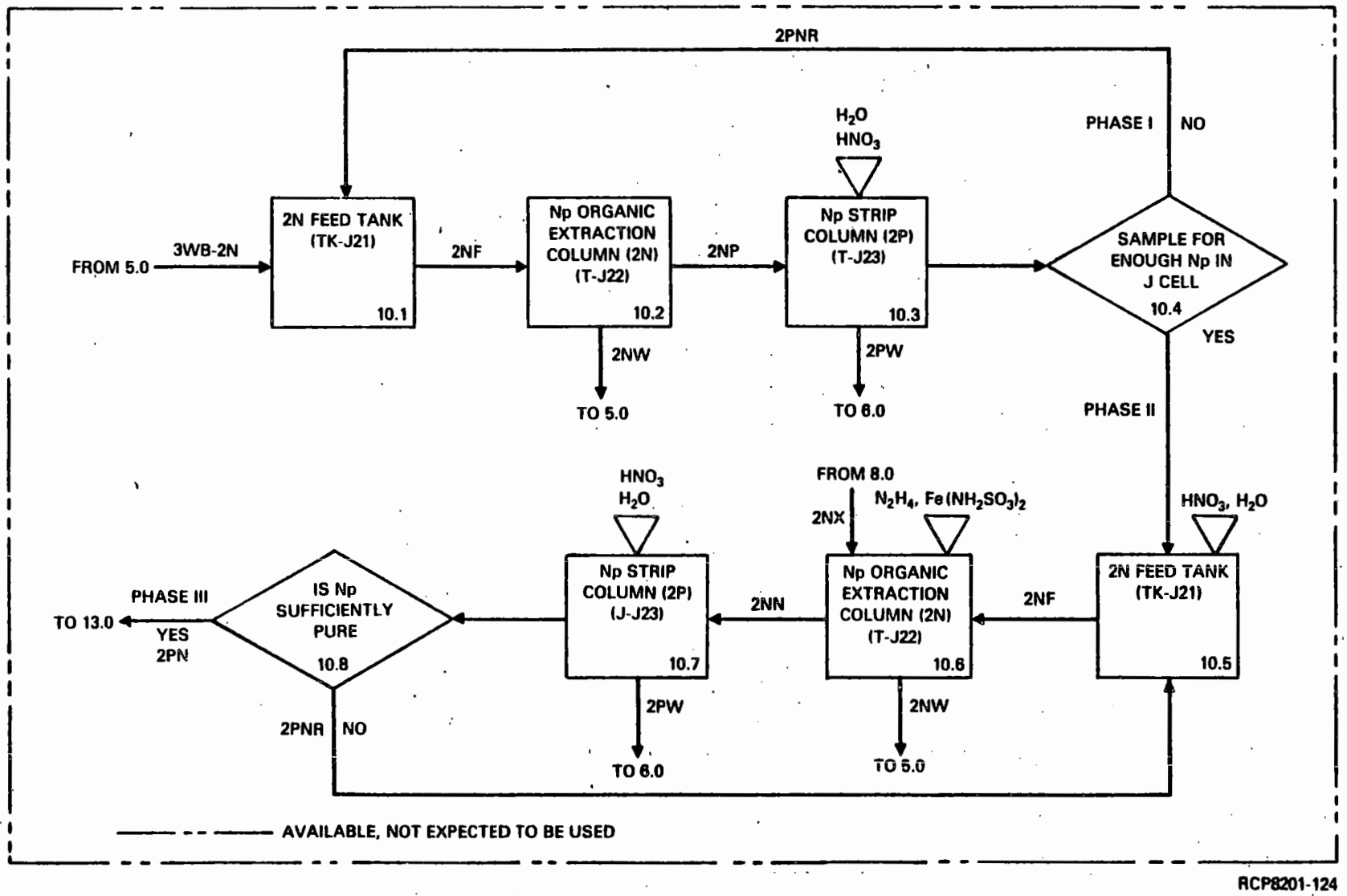


FIGURE C-13. Second Neptunium Cycle.

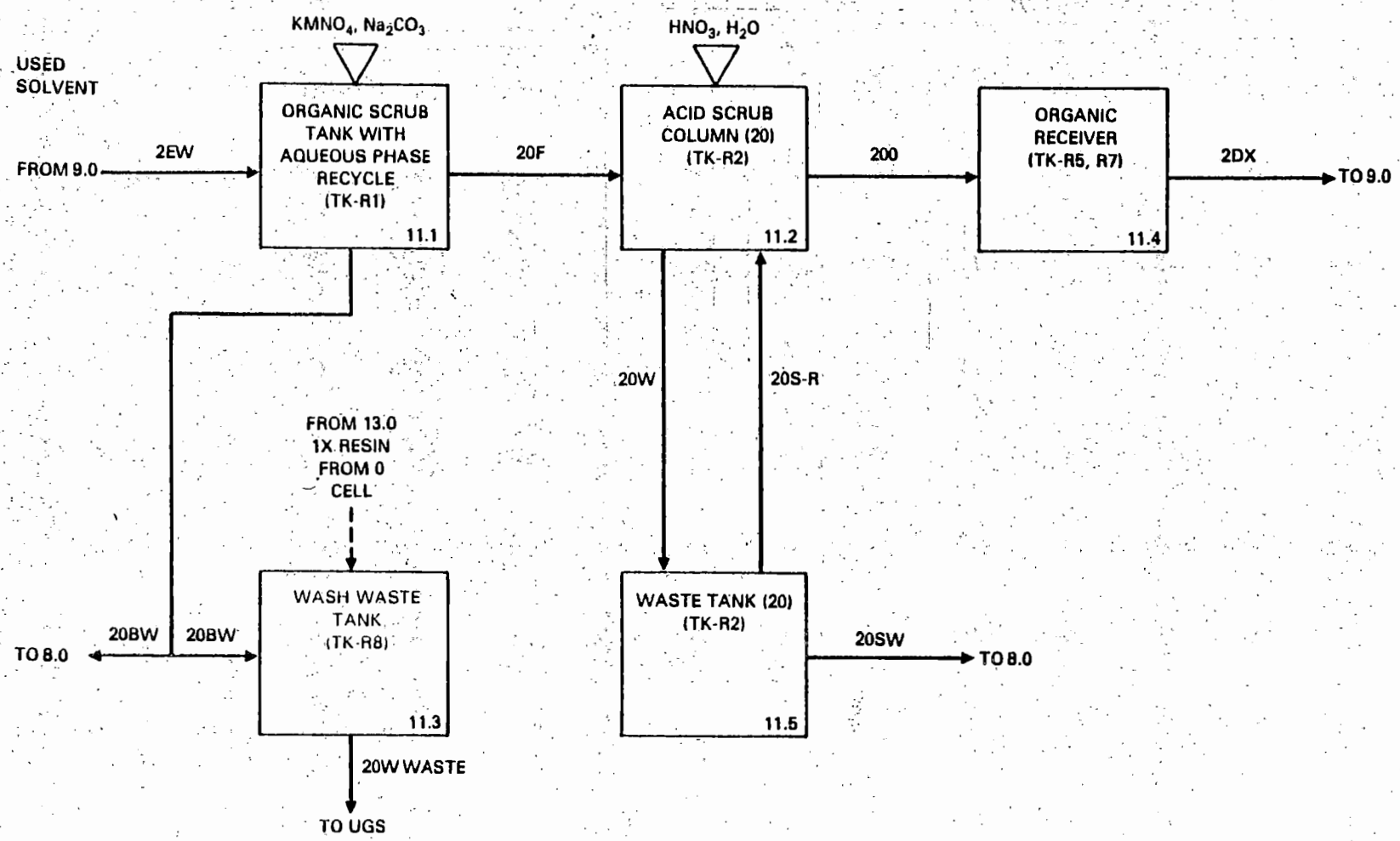


FIGURE C-14. Organic Recovery System 2.

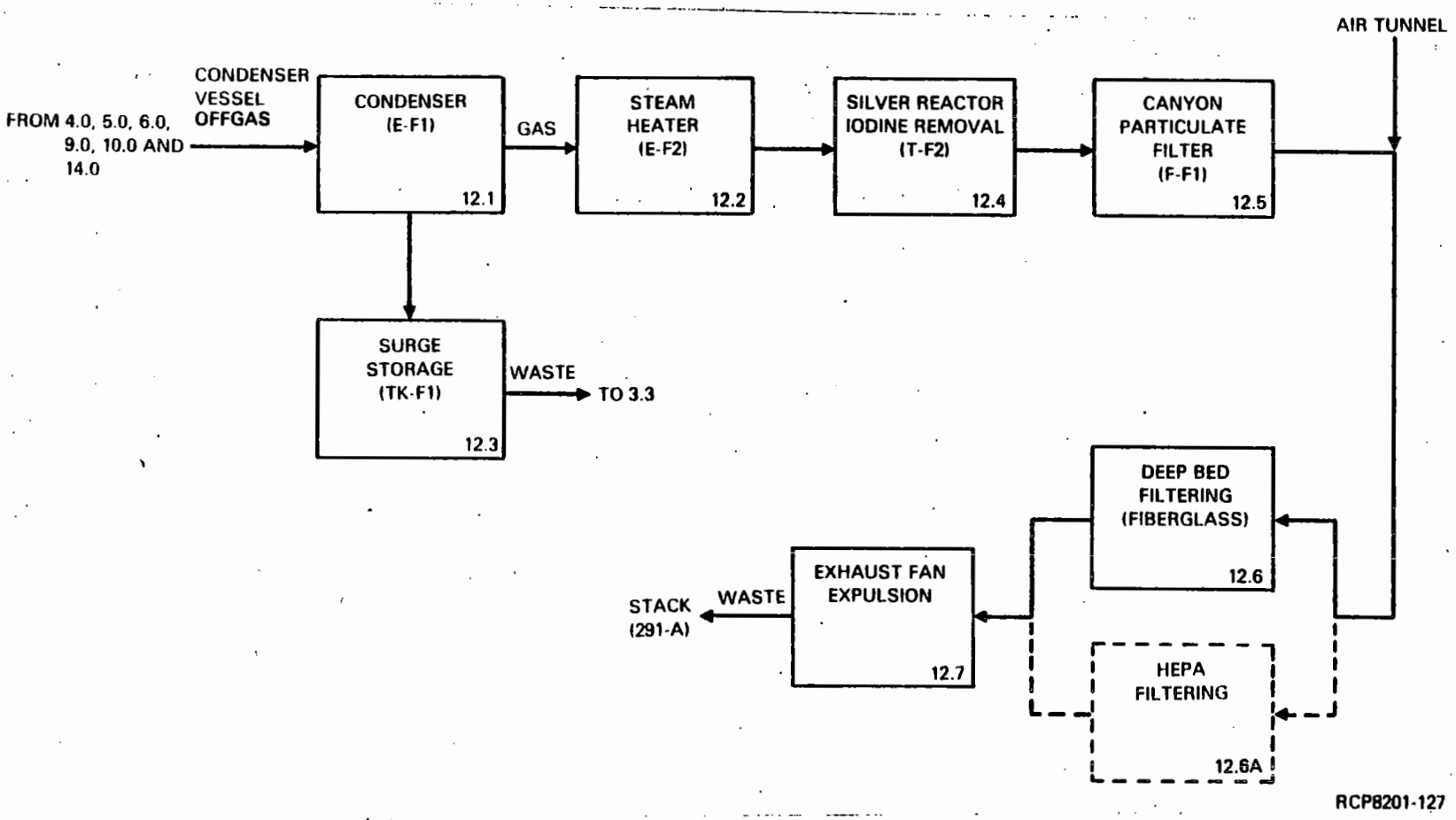


FIGURE C-15. Condenser Vessel Vent Offgas Treatment.

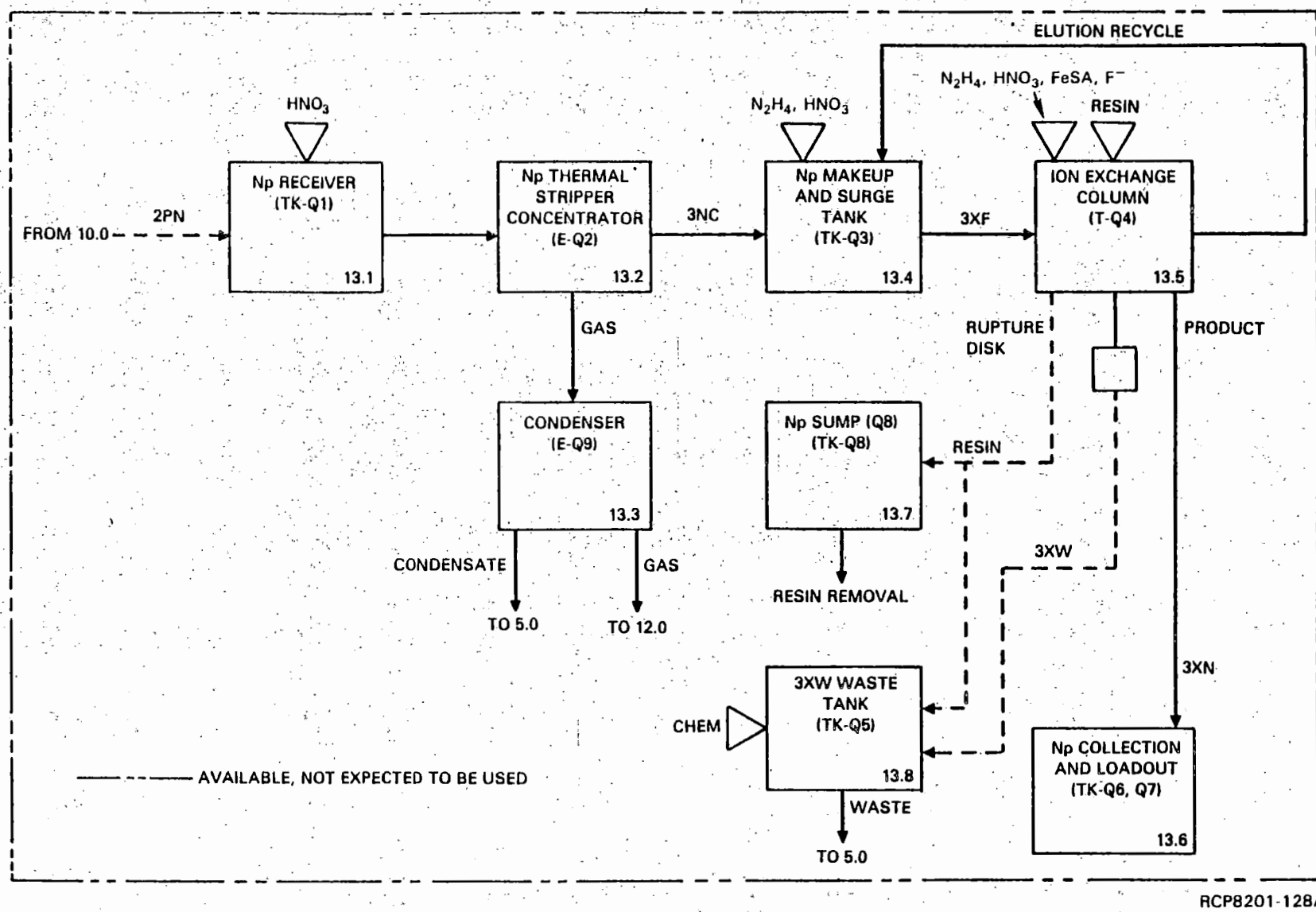


FIGURE C-16. Final Neptunium Cycle.

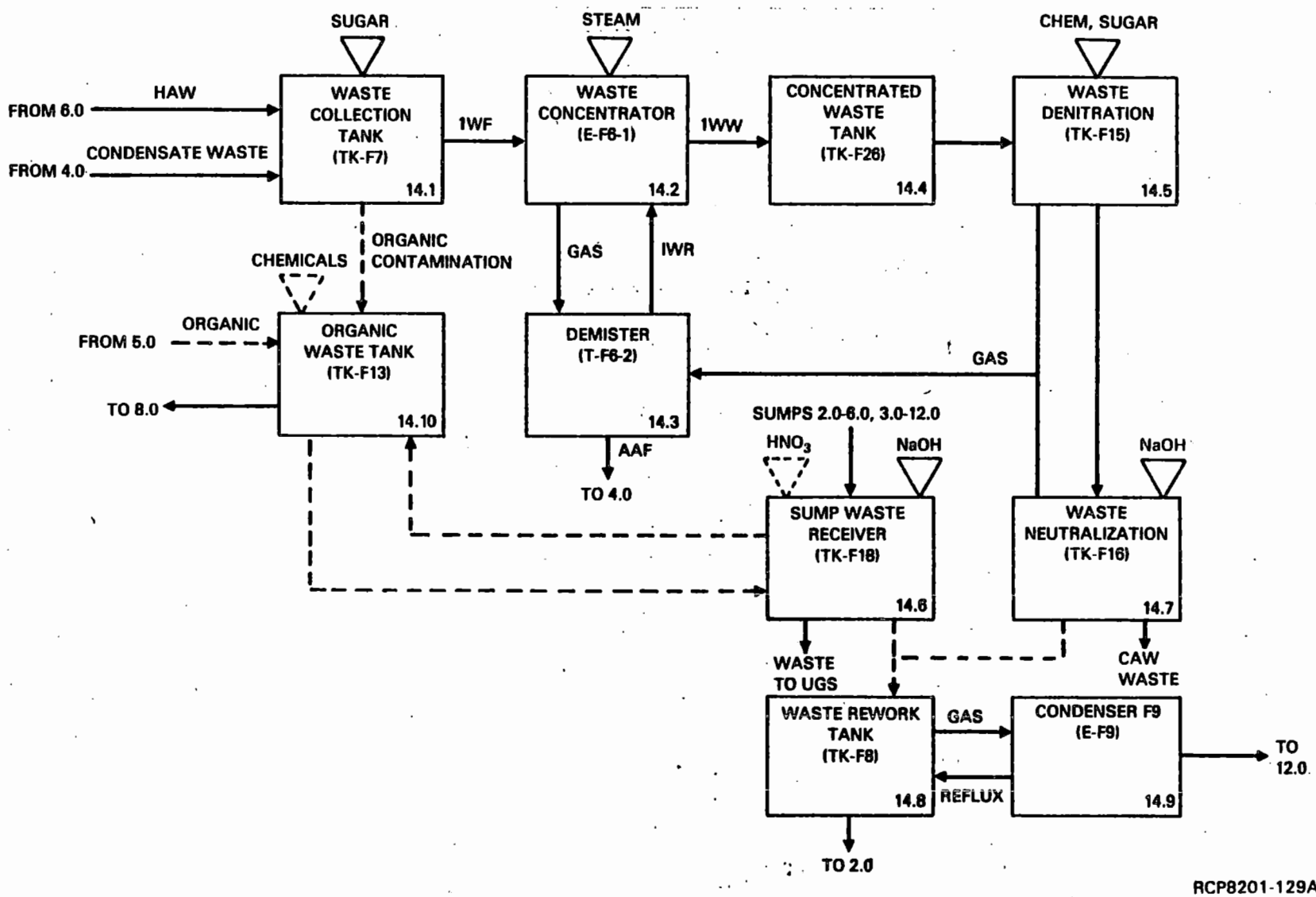


FIGURE C-17. Waste Concentration and Treatment.

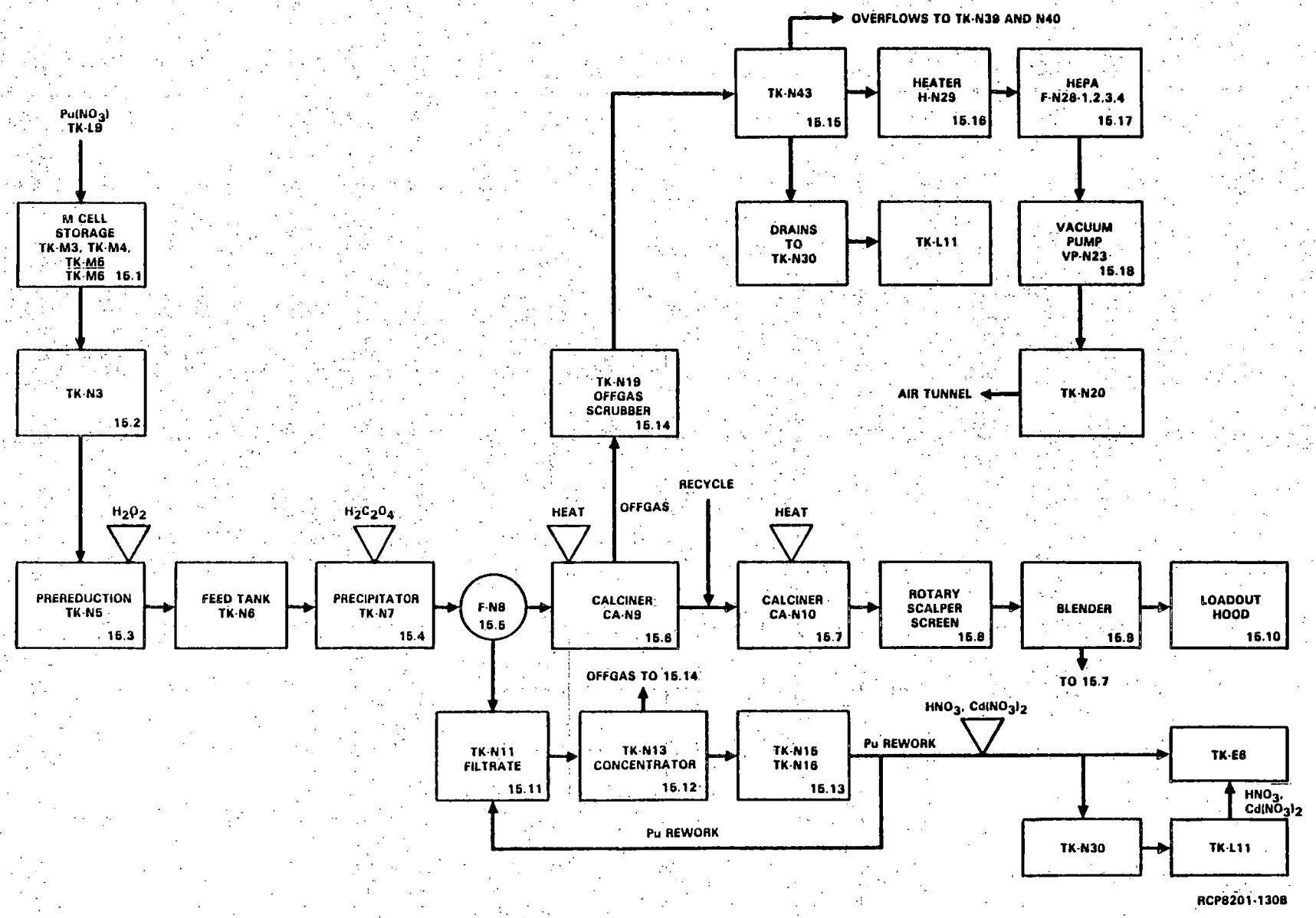


FIGURE C-18. Oxide Conversion.

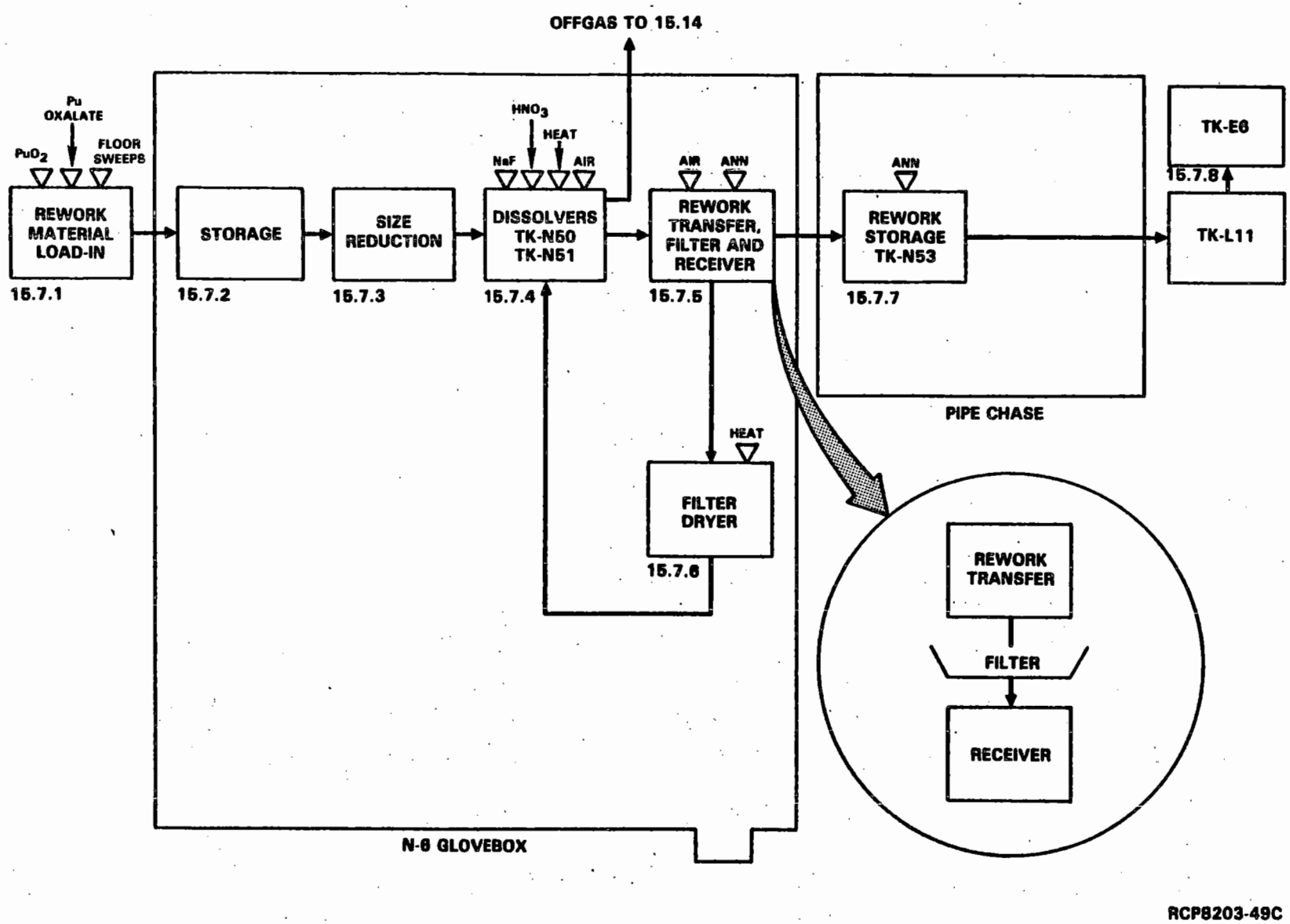


FIGURE C-19. Oxide Conversion Rework.

9413210.1249

PUREX FSAR

APPENDIX D PUREX FAULT TREE ANALYSIS

9142D.1250

This page intentionally left blank.

Appendix D

PUREX FAULT TREE ANALYSIS

THE FAULT TREE MODEL

Fault tree analysis is an analytical technique, whereby an undesired state of the system is specified (usually a state that is critical from a safety standpoint) and the system is then analyzed in the context of its environment and operation to find all credible ways in which the undesired event can occur. The fault tree itself is a graphic representation of the various parallel and sequential combinations of faults that will result in the occurrence of the undesired event. The faults can be events that are associated with component hardware failures, human errors, or any other pertinent events which can lead to the undesired event. A fault tree thus depicts the logical interrelationships of basic events that lead to the undesired event which is the top event of the fault tree.

It is important to note that a fault tree is not a model of all possible system failures or all possible causes for system failure. A fault tree is tailored to its top event which corresponds to some particular system failure mode, and the fault tree thus includes only those faults that contribute to this top event. Moreover, these faults are not exhaustive--they cover only the most credible faults as assessed by the analyst.

It is also important to point out that a fault tree is not in itself a quantitative model. It is a qualitative model that can be evaluated quantitatively and often is.

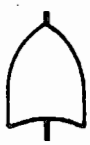
A fault tree utilizes symbols known as gates which serve to permit or inhibit the passage of fault logic up the tree. The gates show the relationships of events needed for the occurrence of a higher event. The higher event is the output of the gate; the lower events are the inputs to the gate. The gate symbol denotes the type of a relationship of the input events required for the output event. Thus, gates are somewhat analogous to switches in an electrical circuit or to valves in a piping layout.

SYMBOLOGY

A typical fault tree is composed of a number of symbols which are described in detail below; and depicted and summarized in Figure D-1.

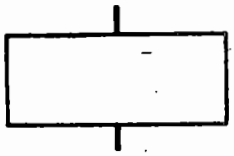
AND-Gate Symbol

Coexistence of all inputs required to produce output.



OR-Gate Symbol (Nonexclusive)

Output will exist if at least one input is present.



RECTANGLE

An event resulting from the combination of more basic events acting through logic gates.

CIRCLE

An event described by a basic component or part failure. The event is independent of other events.

DIAMOND

An event not developed to its cause. Sequence is terminated for lack of information or lack of consequences.

TRIANGLE

A connecting or transfer symbol. All tree construction below the "out" triangle is transferred in at "in" triangle location(s).

FIGURE D-1. Fault Tree Symbology.

9413219.1252

USE OF GRAPHIC SYMBOLS

Graphic symbols used in fault tree construction are of two general categories: logic symbols and event symbols.

Two basic logic symbols (logic gates) are used to interconnect the events that contribute to the specified main event (the TOP event). The AND-gate provides an output event only if all input events occur simultaneously. The OR-gate provides an output event if one or more of the input events are present. When quantifying a fault tree, the probability of the output of an AND-gate is the product of the probabilities of the inputs to the gate. The probability of the output of an OR-gate is approximated by the sum of the inputs to the gate.

The three event symbols used are the RECTANGLE, CIRCLE, and DIAMOND.

The RECTANGLE represents a fault event resulting from the combination of contributory fault events acting through a logic gate. The CIRCLE designates a basic system component failure or input fault event independent of all other events. The DIAMOND symbol is used to depict an input fault event that is considered basic to the specific fault tree being constructed, however, the event described is not basic in the sense that laboratory or field failure rate data are available. Rather, the fault tree is simply not developed further either because: (1) the event is of insufficient consequence or (2) the necessary information to extend the development is not available to the analyst. Events that appear on the tree as circles or diamonds are treated as primary events.

The TRIANGLE is not a true event symbol, strictly speaking. Two triangles are used to accomplish a transfer of one part of the fault tree from another location. A line to the side of the triangle denotes all the tree construction below and including the event transfers out (in addition to its progression up the tree). A line from the triangle apex denotes an event is transferred into the section of the tree as input to the associated logic gate. The transfer-in and transfer-out triangles are uniquely identified to prevent possible confusion between transferred segments. The technique is used to reduce tree construction time and space.

QUANTITATIVE EVALUATION OF FAULT TREES

Quantitative evaluation of fault trees is performed in sequential manner, first determining the component failure probabilities, and finally the system, or top event probability, The term "component" is used to refer to any basic primary event shown on the fault tree (circle, diamond, etc.). For each component, a failure probability model with a constant failure rate per unit time is established. Any time dependent effects such as component wear-in or wear-out are ignored. Where precision better than a factor of 10 is required or time dependent effects are important, then more elaborate models are required.

Consider a component whose failure rate is modeled as a constant failure rate per unit time. This failure rate per hour is denoted by λ. When λ is used, the assumption is made that failure probabilities are directly related to component exposure time. The longer the exposure time, the higher the probability of failure.

For the λ model, the resulting first failure probability distributions are exponential distributions. The probability that a component suffers its first failure in a given time period, t, given it is initially working, is

$$F(t)=1-e-\lambda T$$

Estimates of values for λ are given for a variety of components in various data sources. The analyst needs to obtain a value of λ for each component failure on the fault tree for which the constant failure rate model is applied. To more accurately reflect the cyclic nature of PUREX operations λ is evaluated for primary components as a failure rate or frequency per day (d^{-1}).

Because extreme precision is not required in a fault tree evaluation, the exponential distribution can be approximated by its first order term to simplify the calculations. The cumulative exponential distribution is thus approximated as

$$F(t)=\lambda t$$

This approximation is accurate to within 5% for failure probabilities [F(t)] less than 0.1, and the error is conservative. Furthermore, the approximation error is small compared to uncertainties in λ. Thus, if λ is an operating failure rate, the t is the actual operating time period and f (t) is the probabilty that failure will occur in operation.

PRIMARY FAILURE RATES

The failure rates and demand probabilities used for the quantification of the PUREX fault trees were derived from reference handbooks, reports, operating experience, and experience from similar fuels reprocessing facilities. A major source of failure data, particularly human performance data, was the Reactor Safety Study WASH-1400, Appendix III, Failure Data.(D-1)

Table D-1 presents the summary of the system, component and human failure rates used in the following analyses, along with appropriate references.

TABLE D-1. Primary Failure Rates for PUREX Fault Tree Analysis. (Page 1 of 4)

Component	Failure rate or probability	Reference
Pressure gauges	1.91/yr or 2×10^{-4}/h for systems in contact with dirty fluids	2, 3
	1×10^{-5}/h	4
Temperature measurement systems	0.35/yr or 4×10^{-5}/h	2, 3
Temperature sensor/detector	3.3×10^{-6}/h	3
Thermocouples	1.0×10^{-5}/h	4
Level measurement (using differential pressure, in contact with dirty fluids)	2×10^{-4}/h	3
	1.55/yr	2
Controllers, general purpose, pneumatic electric and temperature	3.3×10^{-5}/h	3
	0.29/yr	2
Solid state devices fail to function, for applications 1 amp and/or 28 V for instrumentation, includes transmitter, amplifier and output device	3×10^{-6}/h	1
Geiger Mueller detector	1.4×10^{-5}/h	3
Ion chamber	3.0×10^{-6}/h	2
Pipes, rupture or plug of a section		
3 in. in diameter	1×10^{-9}/h	1
3 in. in diameter	1×10^{-10}/h	1
General piping failure rate	2×10^{-7}/h	1
	2×10^{-7}/h	4
Gasket failure	1×10^{-4}/h	1
Tubing Leaks	5×10^{-6}/h (based upon highest frequency reported in Ref. 4 for failure rate of piping <3 in. in diam)	1

9413219.1255

TABLE D-1. Primary Failure Rates for PUREX Fault Tree Analysis. (Page 2 of 4)

Component	Failure rate or probability	Reference
Welds, flanges, expansion joints and gaskets	3×10^{-6}/h	1
Tubing Leaks	5×10^{-6}/h (based upon highest frequency reported in Ref. 4 for failure rate of piping 3 in. in dia.)	1
Manual valve, failure to remain open (plug)	1×10^{-4}/demand	1
Relief valves, fail to open	1×10^{-5}/demand	1
Sprinkler head	6×10^{-7}/h	3
Sprinkler systems, failure to perform	5×10^{-2}/demand	5
Heat exchangers, vertical shell and tube type	2.5×10^{-5}/h	3
Heaters, general electrical	2.3×10^{-6}/h	3
HEPA[a] filters	1×10^{-4}/demand	6
Active components:	1×10^{-4}/demand	7
In general, active components modify the system, such as valves which open and close to modify a system's fluid flow or a switch which has a similar effect in an electrical circuit. Also relays pumps, etc.	3×10^{-7}/h	

TABLE D-1. Primary Failure Rates for PUREX Fault Tree Analysis. (Page 3 of 4)

Component	Failure rate or probability	Reference
Passive components:	1×10^{-6} to 1×10^{-7}/demand	7
In general, passive components contribute in a static way to the functioning of a system, such as pipes, journals, welds, wiring. They have a general failure rate 2 to 3 orders of magnitude below active componnents		
Primary failures of containing equipment releasing flammable material capable of explosion and/or fire on ignition. In general, welded and flanged process systems in areas of light vibration from mechanical equipment	1×10^{-4}	8
Primary ignition of liquid mass with a flash point of 110°F to 200°F with no obvious source of ignition and without explosion proof electrical installations.	1×10^{-2}	8
Ignition of massive release of gas or similar release in a process area with no obvious source of ignition	1×10^{-1}	8
External events:		
• DBT strike on Hanford Site[b]	6×10^{-7}/yr 1×10^{-7}/yr	9 10
• HRHE[c]	3.2×10^{-3}/yr	11
• SSE[d]	1.2×10^{-4}/yr	11
Human Performance:		
• Nuclear power plant personnel in high stress situation--extreme accident	0.2/demand	1

TABLE D-1. Primary Failure Rates for PUREX Fault Tree Analysis. (Page 4 of 4)

Component	Failure rate or probability	Reference
• Passive inspection error rate; value used for passive redundancy	0.5/demand	1
• General human error of omission where items omitted are contained in procedures or administrative controls	1×10^{-3}/demand	1
• General human error of omission, where no feedback is provided	1×10^{-2}/demand	1
• Human actions where active redundancy is practiced, i.e., independent checks, duplication of effort, etc.	1×10^{-5}/demand	1

[a]High efficiency particulate air.
[b]Design basis tornado.
[c]Hanford regional historical earthquake.
[d]Safe shutdown earthquake.

9413219.1258

Representative fault trees are developed for each path, although only those found to result in significant releases in the PUREX PHA are further developed.

For those accidents involving significant releases, the fault tree is quantified to develop an estimated frequency of occurrence of the accident. Release paths not felt to be significant are not quantified, however, the fault trees are presented for consistency.

In the quantitative probability estimates made in this analysis, component behavior data in the form of failure rates are required as inputs to the fault trees. Since the goal of this safety analysis is risk assessment, as opposed to reliability analysis, relatively larger errors (order of magnitude type accuracy) can be tolerated in the quantified results. Inherent in these results are uncertainties evolving from variations and uncertainties in the failure rate data used, due to different reporting sources used, variations in application, fluctuation in daily operations and environmental conditions, and variations that can exist from component to component.

The following PUREX Fault Trees (Fig. D-2 through D-9), graphically represent the system or component failures required to cause the top event in Fault Tree 1, an "above normal release of radioactivity from PUREX operations." Fault Tree 1 is the top tier of the PUREX Fault Trees, and organizes the analysis for the selection of significant accidents. Fault tree analysis of a complex facility can easily become unwieldy, therefore development of the analysis must be concentrated in the areas felt to present the most significant hazard.

The approach taken for the fault tree analysis of the PUREX Plant operations has been to select the specific top event, and to so organize the top tiers, that a format is developed that selects failures or accidents with significant consequences.

For the PUREX Plant the configuration presenting the greatest source term and most significant potential for driving force is that of normal operations, therefore this is the configuration selected for further development on tier "a" of Fault Tree 1 (see Fig. D-2). Within the context of an operating plant, events lacking sufficient source term or driving force to present significant potential hazard are excluded from analysis. This exclusion is shown on tier "b" of Fault Tree 1.

Further resolution is accomplished by confining the accidents to be analyzed to those resulting in significant airborne releases. Airborne release is specified as it is felt to result in the most widespread distribution of radionuclides, resulting in immersion, inhalation, and ingestion doses. The most logical analysis therefore, centers on examination of aerosol release paths. This results in a focus on accidents that result in releases into the exhaust systems, or releases directly to the environment. An additional category investigated is that of releases due to external phenomena, which provide their own energy for dispersion, and may create additional release paths via openings in the buildings.

Tier "c" of Fault Tree 1 specifies these release paths of interest. Lower tiers, "d" through "g" specify potential accidents which could result in radionuclide releases via these release paths. Fault Tree 1 stops at the level of identifying potential accidents of significance. Development of each accident is carried out in subsequent lower tier fault trees, indicated by "transfer in" symbols. For example, block d1, "URR due to N cell Fire" shows a "transfer in" symbol, a triangle from Fault Tree 3, indicating that a detailed fault tree analysis was performed in Fault Tree 3.

9413219.1260

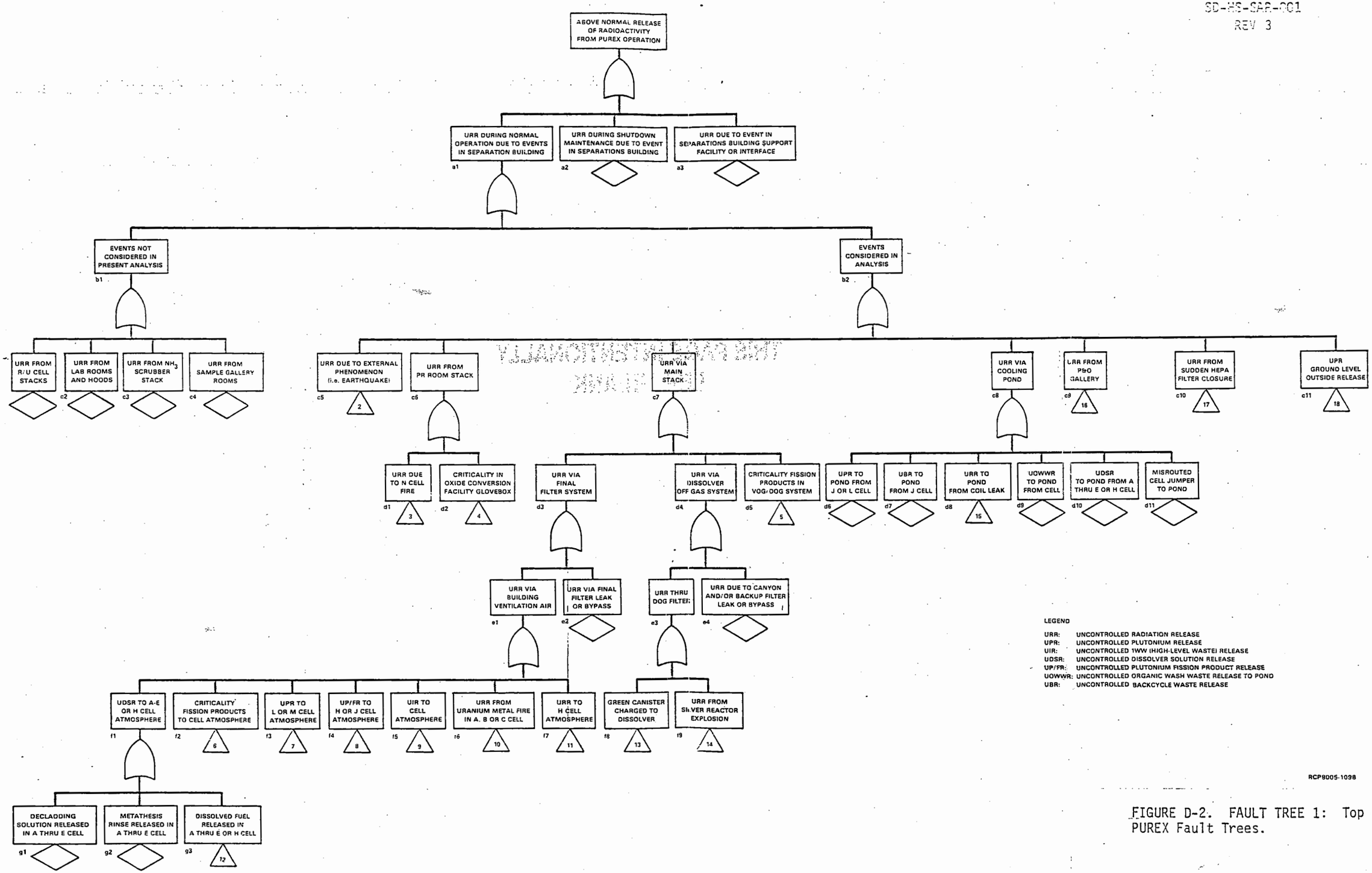


FIGURE D-2. FAULT TREE 1: Top of PUREX Fault Trees.

THIS PAGE INTENTIONALLY
LEFT BLANK

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK

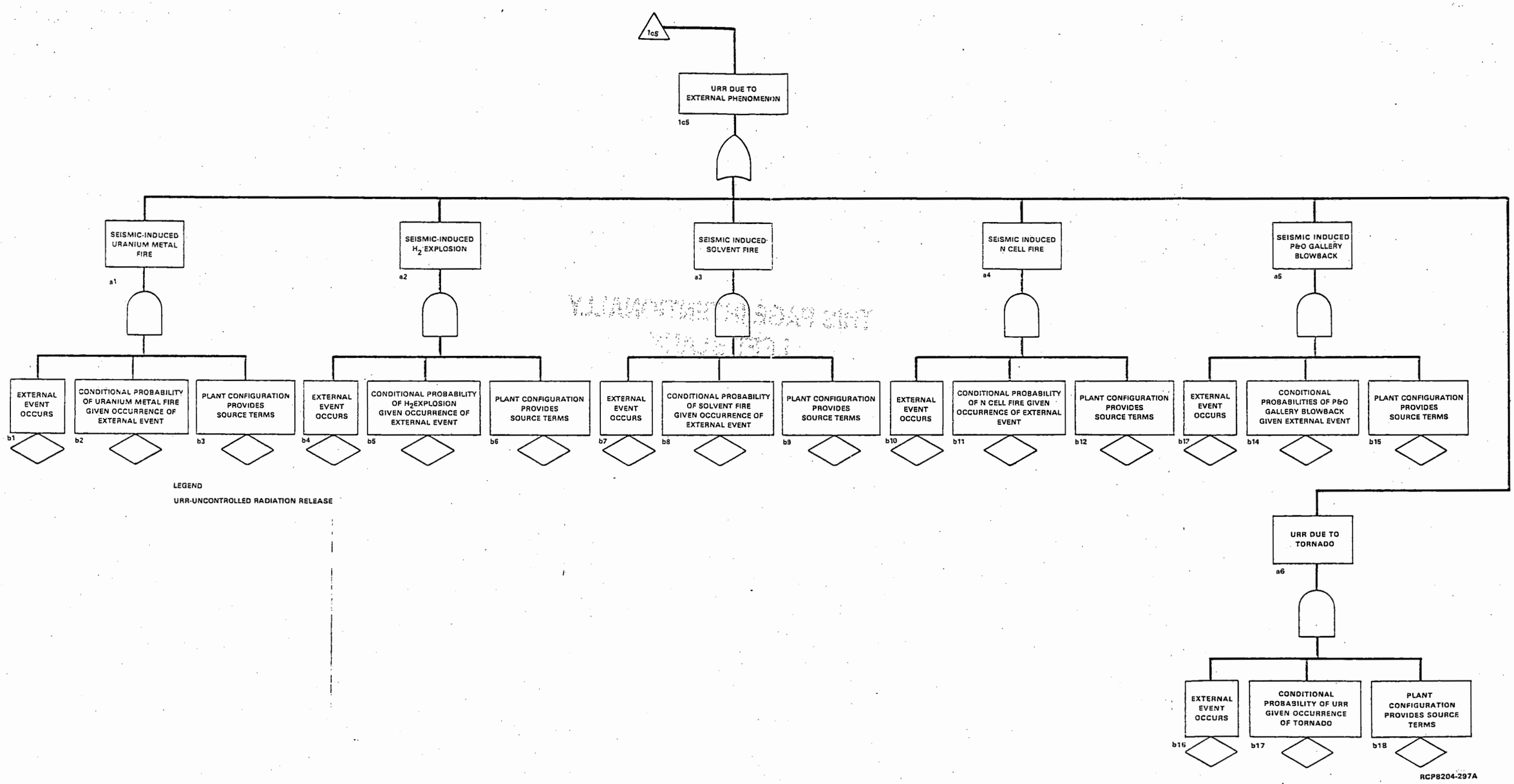


FIGURE D-3. FAULT TREE 2: URR Due to External Phenomenon.

THIS PAGE INTENTIONALLY
LEFT BLANK

This page intentionally left blank.

Quantification of 1c5

URR Due to External Phenomenon

Event	Remarks	Conditional probability
b1 - External event occurs (yr^{-1})	DBT	
	HRHE (0.1 g lateral acceleration	3.2×10^{-3}
	SSE (0.25 g lateral acceleration	1.2×10^{-4}
b2 - Conditional probability of uranium metal fire	Judgmental evaluation of the seismic resistance of the drown tanks and their associated piping indicated a significant conditional probability of failure under HRHE and SSE conditions. Loss of drown tank capability could potentially have resulted in the inability to keep the dissolver fuel covered, possibly resulting in a uranium metal fire. Engineering upgrades have been provided to seismically harden these components and thus preclude the possibility of loss of drown tank capability. The probability of a seismic-induced uranium metal fire is judged to be insignificant.	
b3 - Plant configuration provides source terms	Based on a TOE of 70%	.7
b4 - External event occurs	See b1	
b5 - Conditional probability of H_2 explosion	Evaluated in Ref. 12	HRHE - 1.1×10^{-2} SSE - 3.7×10^{-2}
b6 - Plant configuration provides source terms	See b3	

9413219.1265

Event	Remarks	Conditional Probability
b7 - External event occurs	See b1	
b8 - Conditional probability of solvent fire in H-J cells	Judgmental evaluation of the seismic resistance of the H-J cell fire suppression system indicated a significant conditional probability of failure under HRHE and SSE conditions. Engineering upgrades have been provided to include a seismically hardened fire suppression system to reduce the probability of a seismic induced solvent fire by a factor of 5×10^{-2}.	HRHE - 1×10^{-3} SSE - 1.8×10^{-3}
b9 - Plant configuration provides source terms	See b3	
b10 - External event occurs	See b1	
b11 - Conditional probability of N cell fire	Evaluated in Ref. 12	HRHE - 1.0 SSE - 1.0
b12 - Plant configuration provides source terms	See b3	
b13 - External event occurs	See b1	
b14 - Conditional probability of pipe and operating (P&O) gallery blowback	Evaluated in Ref. 12	HRHE - 5.2×10^{-3} SSE - 3.3×10^{-2}
b15 - Plant configuration provides source terms	See b3	
b16 - External event occurs	DBT	6×10^{-7}

b17 - Conditional probability of release given occurrence of DBT	1.0
b-18 Plant configuration provides source terms	1.0

$P(1c5) = a1 + a2 + a3 + a4 + a5 + a6$

$P(a1) = (b1)(b2)(b3)$

$P(a1)$ for HRHE $= (3.2 \times 10^{-3})(\varepsilon)(.7)$

$= \varepsilon$

$P(a1)$ for SSE $= (1.2 \times 10^{-4})(\varepsilon)(.7)$

$= \varepsilon$

$P(a2) = (b4)(b5)(b6)$

$P(a2)$ for HRHE $= (3.2 \times 10^{-3})(1.1 \times 10^{-2})(.7)$

$= 2.5 \times 10^{-5}/yr$

$P(a2)$ for SSE $= (1.2 \times 10^{-4})(3.7 \times 10^{-2})(.7)$

$= 3.1 \times 10^{-6}/yr$

$P(a3) - (b7)(b8)(b9)$

$P(a3)$ for HRHE $= (3.2 \times 10^{-3})(1 \times 10^{-3})(.7)$

$= 2.2 \times 10^{-6}/yr$

$P(a3)$ for SSE $= (1.2 \times 10^{-4})(1.8 \times 10^{-3})(.7)$

$= 1.5 \times 10^{-7}/yr$

$P(a4) = (b10)(b11)(b12)$

$P(a4)$ for HRHE $= (3.2 \times 10^{-3})(1.0)(.7)$

$= 2.2 \times 10^{-3}/yr$

$P(a4)$ for SSE $= (1.2 \times 10^{-4})(1.0)(.7)$

$= 8.4 \times 10^{-5}/yr$

$P(a5) = (b13)(b14)(b15)$

9413249.1267

P(a5) for HRHE $= (3.2 \times 10^{-3})(5.2 \times 10^{-3})(.7)$

$= 1.2 \times 10^{-5}/yr$

P(a5) for SSE $= (1.2 \times 10^{-4})(3.3 \times 10^{-2})(.7)$

$= 2.8 \times 10^{-6}/yr$

P(a6) for DBT $= (6 \times 10^{-7})\ (1.0)\ (1.0)$

$= 6 \times 10^{-7}/yr$

P(1c5) for HRHE $= (\ ) + (2.5 \times 10^{-5}) + 2.2 \times 10^{-6}) + 2.2 \times 10^{-3})$

$+ 1.2 \times 10^{-5}) + 6 \times 10^{-7})$

$= 2.2 \times 10^{-3}/yr$

P(1c5) for SSE $= (\ ) + (3.1 \times 10^{-6} + 1.5 \times 10^{-7}) + (8.4 \times 10^{-5})$

$+ (2.8 \times 10^{-6}) + (6 \times 10^{-7})$

$= 9.0 \times 10^{-5}/yr$

9921 0126146

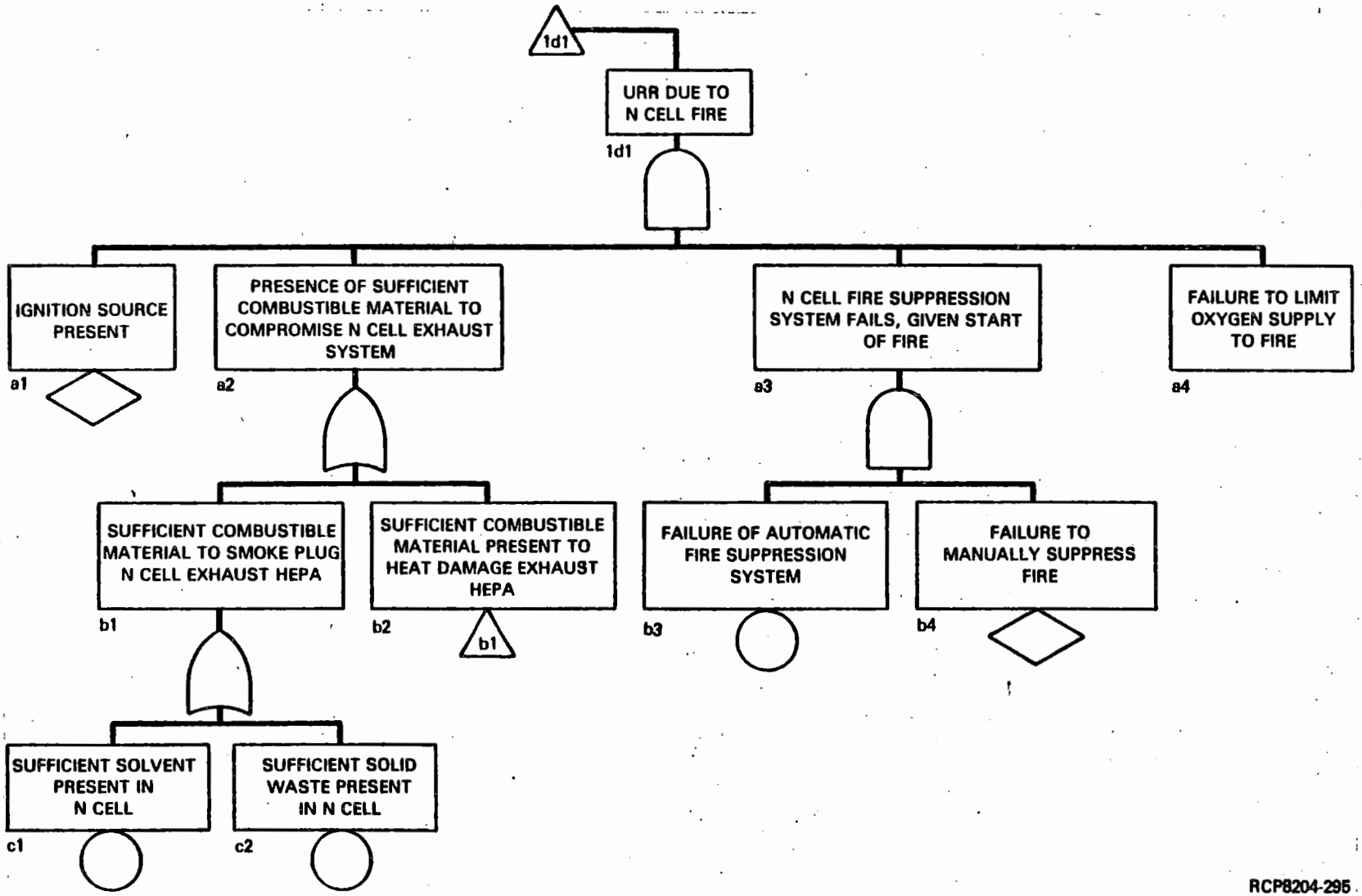


FIGURE D-4. FAULT TREE 3: URR Due to N Cell Fire.

Quantification of 1d1

UPR Due to N Cell Fire

Event	Remarks	Failure frequency (d^{-1})
a1 - Ignition source present	No obvious source of ignition; probability of 1×10^{-2}	1×10^{-2}
c1 - Presence of sufficient solvent in N cell	Equated with probability of pipe rupture at 1×10^{-9}/h; $(1 \times 10^{-9}/h)(24\ h/d)$	2.4×10^{-8}
c2 - Presence of sufficient solid waste in N cell	Procedural error (1×10^{-3}/demand) over two shifts, passive redundancy failure rate of 0.5 applies; $(1 \times 10^{-3})(.5)=5 \times 10^{-4}$	5×10^{-4}
b2 - Sufficient combustible material present to heat damage exhaust HEPA	Quantified as b1	
b3 - Failure of automatic fire suppression system	Probability of failure 5×10^{-2}	5×10^{-2}
b4 - Failure to manually suppress	0.2 general error rate under very high stress levels with dangerous activities	0.2
a4 - Failure to limit oxygen to fire	1.0/demand	1.0

$$P(1d1) = (a1)(a2)(a3)(a4)$$

$$= [a1][b1 + b2][(b3)(b4)][a4]$$

$$= [a1][(c1 + c2) + (c1 + c2)][(b3)(b4)][a4]$$

$$= [1 \times 10^{-2}][(2.4 \times 10^{-8} + 5 \times 10^{-4}) + (2.4 \times 10^{-8} + 5 \times 10^{-4})]$$

$$[(5 \times 10^{-2})(.2)][1.0]$$

$$= [1 \times 10^{-2}][1 \times 10^{-3}][1.0 \times 10^{-2}][1.0]$$

$$P(1d1) = 1 \times 10^{-7}/d \times 365\ d = 3.7 \times 10^{-5}/yr$$

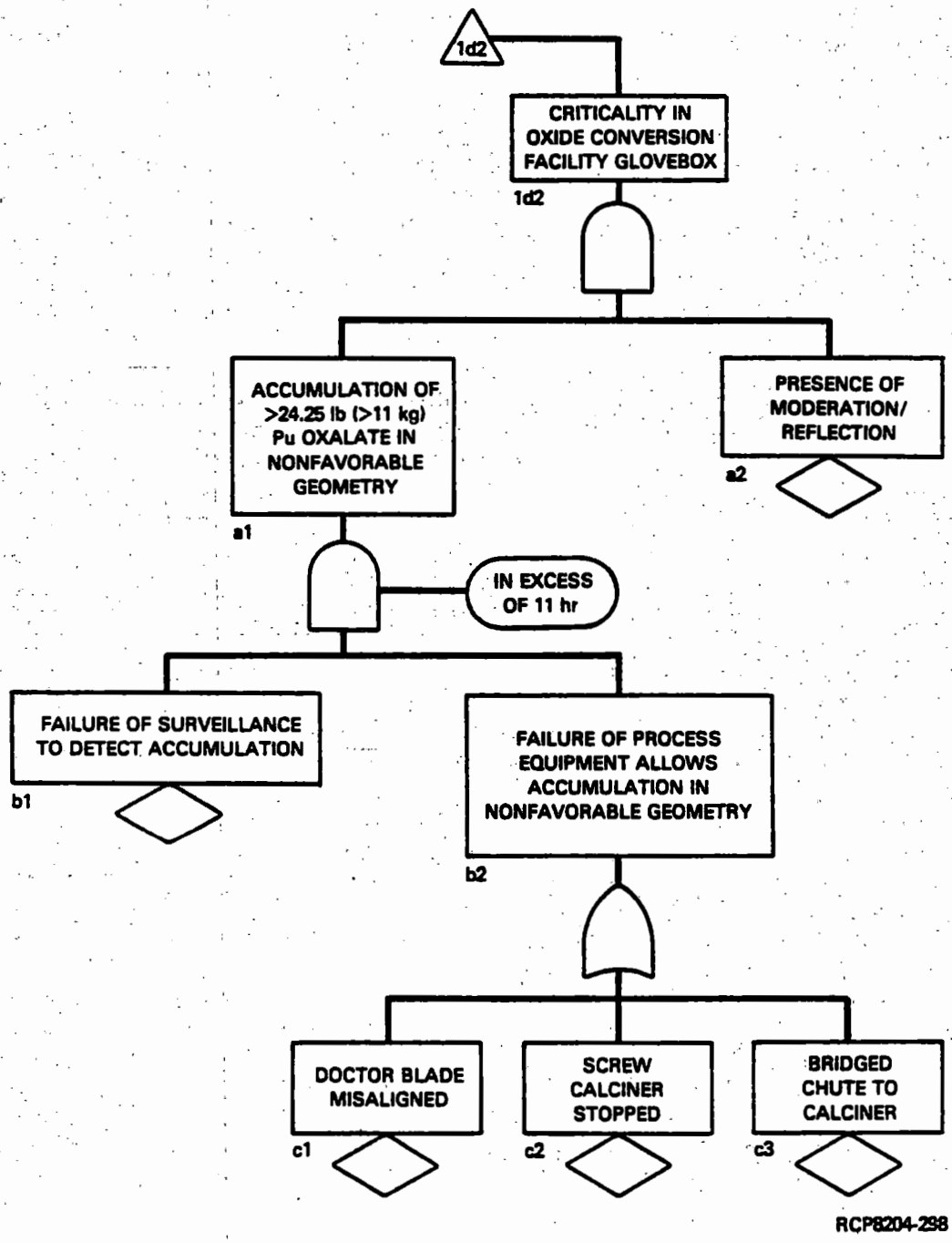


FIGURE D-5. FAULT TREE 4: Criticality in Oxide Conversion Facility Glovebox.

Quantification of 1d2

Criticality in Oxide Conversion Facility Glovebox

Event	Remarks	Failure Frequency (d^{-1})
b1 - Failure of surveillance to detect accumulation	Active redundancy over a period of time greater than 11 h (1×10^{-5})	1×10^{-5}
c1 - Doctor blade misaligned	1/shift or 3/d	3
c2 - Calciner screw stopped	1/shift or 3/d	3
c3 - Bridged chute to calciner	1/shift or 3/d	3
a2 - Presence of moderation/ reflection	Procedural violation 1×10^{-3}/ demand; 3 demands/d	3×10^{-3}

$$P(1d2) = (a1)(a2)$$

$$= [(b1)(b2)][a2]$$

$$= [b1][c1 + c2 + c3]\ [a2]$$

$$= [1 \times 10^{-5}][9][3 \times 10^{-3}]$$

$$= [1 \times 10^{-4}][3 \times 10^{-3}]$$

$$P(1d2) = 3 \times 10^{-7}/d \times 250/d = 7.5 \times 10^{-5}/yr$$

THIS PAGE INTENTIONALLY
LEFT BLANK

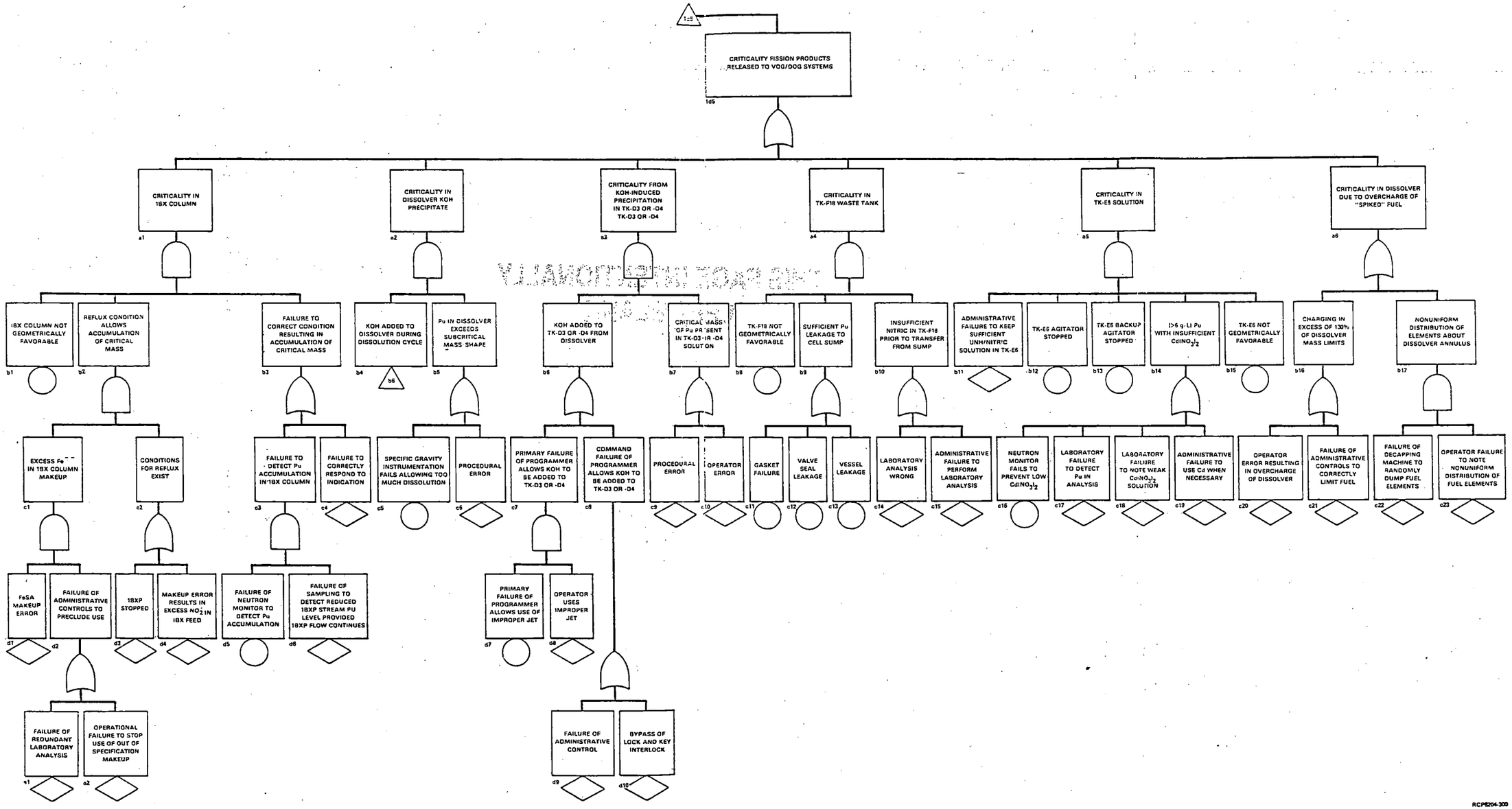


FIGURE D-6. FAULT TREE 5: Criticality Fission Products in VOG/DOG System.

THIS PAGE INTENTIONALLY
LEFT BLANK

9413219.1271

This page intentionally left blank.

Quantification of ld5

Criticality Fission Products to VOG/DOG Systems

Event	Remarks	Failure frequency (d^{-1})
b1 - 1BX column not geometrically favorable	Assumed	1.0
d1 - FeSA makeup error	Passive redundancy failure rate of 5×10^{-3}/demand; (5×10^{-3}/demand) 2 demand/d)	1×10^{-2}
e1 - Failure of redundant lab analysis	Active redundancy failure rate of 1×10^{-5}/demand; (1×10^{-5}/demand) (2 demand/d)	2×10^{-5}
e2 - Failure to stop use of out of specification makeup	Failure to follow procedure; failure rate of 1×10^{-3}/demand; (1×10^{-3}/demand)(1 demand/60 d)	2×10^{-5}
d3 - 1BXP stopped	Passive redundancy failure rate of 5×10^{-3}/demand; (5×10^{-3}/demand) (3 demands/d)	1.5×10^{-2}
d4 - Makeup error results in excess NO_2 in 1BX feed	Error rate of 1×10^{-2}/demand; approximate 1 demand/d	1×10^{-2}
d5 - Failure of neutron monitor to detect Pu accumulation	Failure rate for solid state devices including ion chambers 3×10^{-6}/h; (3×10^{-6}/h)(24 h/d)	7.2×10^{-5}
d6 - Sampling failure to detect reduced Pu in 1BXP stream	Failure rate of 1×10^{-2}/demand; (1×10^{-2}/demand)(3 demands/d)	3×10^{-2}
c4 - Failure to properly respond to indication	Passive redundancy failure rate of 5×10^{-3}/demand; (5×10^{-3}/demand) (3 demands/d)	1.5×10^{-2}
b4 - KOH added to dissolver during dissolution cycle	Quantified for b6	6×10^{-3}
c5 - Specific gravity instrument fails	Failure rate for pressure measurement instrument in contact with dirty fluids is 2×10^{-4}/h; (2×10^{-4}/h)(24 h/d)	4.8×10^{-3}

9413219.1275

Event	Remarks	Failure frequency (d^{-1})
c6 - Procedural error	Active redundancy failure rate of 1×10^{-5}/demand; (1×10^{-5}/demand) (3 demands/d)	3×10^{-5}
d7 - Primary failure of programmer	Failure rate for general purpose controllers of 3.3×10^{-5}/h; (3.3×10^{-5}/h)(24 h/d)	8×10^{-4}
d8 - Operator uses improper jet	Failure rate of 1×10^{-2}/demand; (1×10^{-2}/demand)(3 demands/d)	3×10^{-2}
d9 - Failure of administrative control	Procedural failure rate of 1×10^{-3}/demand; (1×10^{-3}/demand) (3 demand/d)	3×10^{-3}
d10 - Bypass of lock and key interlock	See d9	3×10^{-3}
c9 - Procedural error	Active redundancy failure rate of 1×10^{-5}/demand; (1×10^{-5}/demand) (3 demand/d)	3×10^{-5}
c10 - Operator error	Active redundancy failure rate of 1×10^{-5}/demand; (1×10^{-5}/demand) (3 demand/d)	3×10^{-5}
b8 - TK-F18 not geometrically favorable	Assumed probability of 1.0	1.0
c11 - Gasket failure	Gasket failure rate of 1×10^{-4}/h; (1×10^{-4}/h)(24 h/d)	2.4×10^{-3}
c12 - Valve seal leakage	General piping failure rate of 1×10^{-9}/h; (1×10^{-9}/demand) (24 h/d)	2.4×10^{-8}
c13 - Vessel leakage	See c12	2.4×10^{-8}
c14 - Lab analysis error	Passive redundancy failure rate of 5×10^{-3}/demand; (5×10^{-3}/demand) 1 demand/d)	5×10^{-3}
c15 - Failure to perform lab analysis	Procedural error rate of 1×10^{-3}/demand; (1×10^{-3}/demand)(1 demand/d)	1×10^{-3}

Event	Remarks	Failure frequency (d^{-1})
b11 - Administrative failure	Procedural error rate of 1×10^{-3}/demand; (1×10^{-3}/demand)(1 demand/d)	1×10^{-3}
b12 - TK-E6 agitator stopped	Human error rate of 1×10^{-2}/demand; (1×10^{-2}/demand) (3 demand/d)	3×10^{-2}
b13 - TK-E6 backup agitator stopped	Procedural error rate of 1×10^{-3}/demand; (1×10^{-3}/demand) (3 demand/d)	3×10^{-3}
c16 - Neutron monitor fails	See d5	7.2×10^{-5}
c17 - Lab failure to detect Pu	Human error rate of 1×10^{-2}/demand; (1×10^{-2}/demand) (3 demand/d)	3×10^{-2}
c18 - Lab failure to detect weak $Cd(NO_3)_2$	Human error rate of 1×10^{-2}/demand; (1×10^{-2}/demand) (1 demand/30/d)	3×10^{-4}
c19 - Administrative failure	Procedural error rate of 1×10^{-3}/demand; (1×10^{-3}/demand) (3 demand/d)	3×10^{-3}
b15 - TK-E6 not geometrically favorable	Assumed probability of 1.0	1.0
c20 - Operator error results in overcharge of dissolver	Active redundancy failure rate of 1×10^{-5}/demand; (1×10^{-5}/demand) (3 demand/d)	3×10^{-5}
c21 - Failure of administrative controls to limit fuel to dissolver	Active redundancy failure rate of 1×10^{-5}/demand; (1×10^{-5}/demand) (3 demand/d)	3×10^{-5}
c22 - Failure of decapping machine to randomly dump	Sufficient nonrandomness is considered highly unlikely to be achievable	ε
c23 - Operator failure to note nonuniform distribution of fuel	Human error failure rate of 1×10^{-2}/demand; (1×10^{-2}/demand) (3 demand/d)	3×10^{-2}

AL3219.1277

$P(1d5) = a1 + a2 + a3 + a4 + a5 + a6$

$P(a1) = (b1)(b2)(b3)$

$= [(b1)][(c1)(c2)][(c3) + (c4)]$

$= \{b1\}\{[(d1)(d2)][d3 + d4]\}\{[(d5)(d6)] + c4\}$

$= \{b1\}\{[(d1)(e1 + e2)][d3 + d4]\}\{[(d5)(d6)] + c4\}$

$= \{1.0\}\{[(1 \times 10^{-2})(2 \times 10^{-5} + 2 \times 10^{-5})][1.5 \times 10^{-2} + 1 \times 10^{-2}]\}$
$\{[(7.2 \times 10^{-5})(3 \times 10^{-2})] + 1.5 \times 10^{-2}\}$

$= \{1.0\}\{[4 \times 10^{-7}][2.5 \times 10^{-2}]\}\{2.2 \times 10^{-7} + 1.5 \times 10^{-2}\}$

$= [1.0][1 \times 10^{-8}][1.5 \times 10^{-2}]$

$= 1.5 \times 10^{-10}/d$

$P(a1) = 1.5 \times 10^{-10}/d \times 250\ d = 3.8 \times 10^{-8}/yr$

$P(a2) = [b4][b5]$

$= [b4][c5 + c6]$

$= [6 \times 10^{-3}][4.8 \times 10^{-3} + 3 \times 10^{-5}]$

$= [6 \times 10^{-3}][4.8 \times 10^{-3}]$

$P(a2) = 3 \times 10^{-5} \times 250\ d = 7.5 \times 10^{-3}/yr$

$P(a3) = [b6][b7]$

$= [c7 + c8][c9 + c10]$

$= \{[(d7)(d8)] + (d9 + d10)\}\{c9 + c10\}$

$= \{[(8 \times 10^{-4})(3 \times 10^{-2})] + (3 \times 10^{-3} + 3 \times 10^{-3})\}\{3 \times 10^{-5} + 3 \times 10^{-5}\}$

$= \{2.4 \times 10^{-5} + 6 \times 10^{-3}\}\{6 \times 10^{-5}\}$

$= \{6 \times 10^{-3}\}\{6 \times 10^{-5}\}$

$P(a3) = 3.6 \times 10^{-7} \times 250\ d = 9.0 \times 10^{-5}/yr$

9413219.1278

$P(a4) = (b8)(b9)(b10)$

$= [b8][c11 + c12 + c13][c14 + c15]$

$= [1][2.4 \times 10^{-3} + 2.4 \times 10^{-8} + 2.4 \times 10^{-8}][5 \times 10^{-3} + 1 \times 10^{-3}]$

$= [1][2.4 \times 10^{-3}][6 \times 10^{-3}]$

$P(a4) = 1.4 \times 10^{-5}/d \times 250\ d = 3.6 \times 10^{-3}/yr$

$P(a5) = (b11)(b12)(b13)(b14)(b15)$

$= (b11)(b12)(b13)[(c16 + c17 + c18 + c19)](b15)$

$= (1 \times 10^{-3})(3 \times 10^{-2})(3 \times 10^{-3})[(7.2 \times 10^{-5} + 3 \times 10^{-2} +$
$3 \times 10^{-4} + 3 \times 10^{-3})](1)$

$= (1 \times 10^{-3})(3 \times 10^{-2})(3 \times 10^{-3})(3 \times 10^{-2})(1)$

$P(a5) = 3 \times 10^{-9}/d \times 250\ d = 7.5 \times 10^{-7}/yr$

$P(a6) = (b16) + (b17)$

$= c20 + c21 + [(c22)(c23)]$

$= 3 \times 10^{-5} + 3 \times 10^{-5} + [(\ )(3 \times 10^{-2})]$

$= 3 \times 10^{-5} + 3 \times 10^{-5} +$

$P(a6) = 6 \times 10^{-5} \times 250\ d = 1.5 \times 10^{-2}/yr$

9413219.1229

9413219.1280

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK

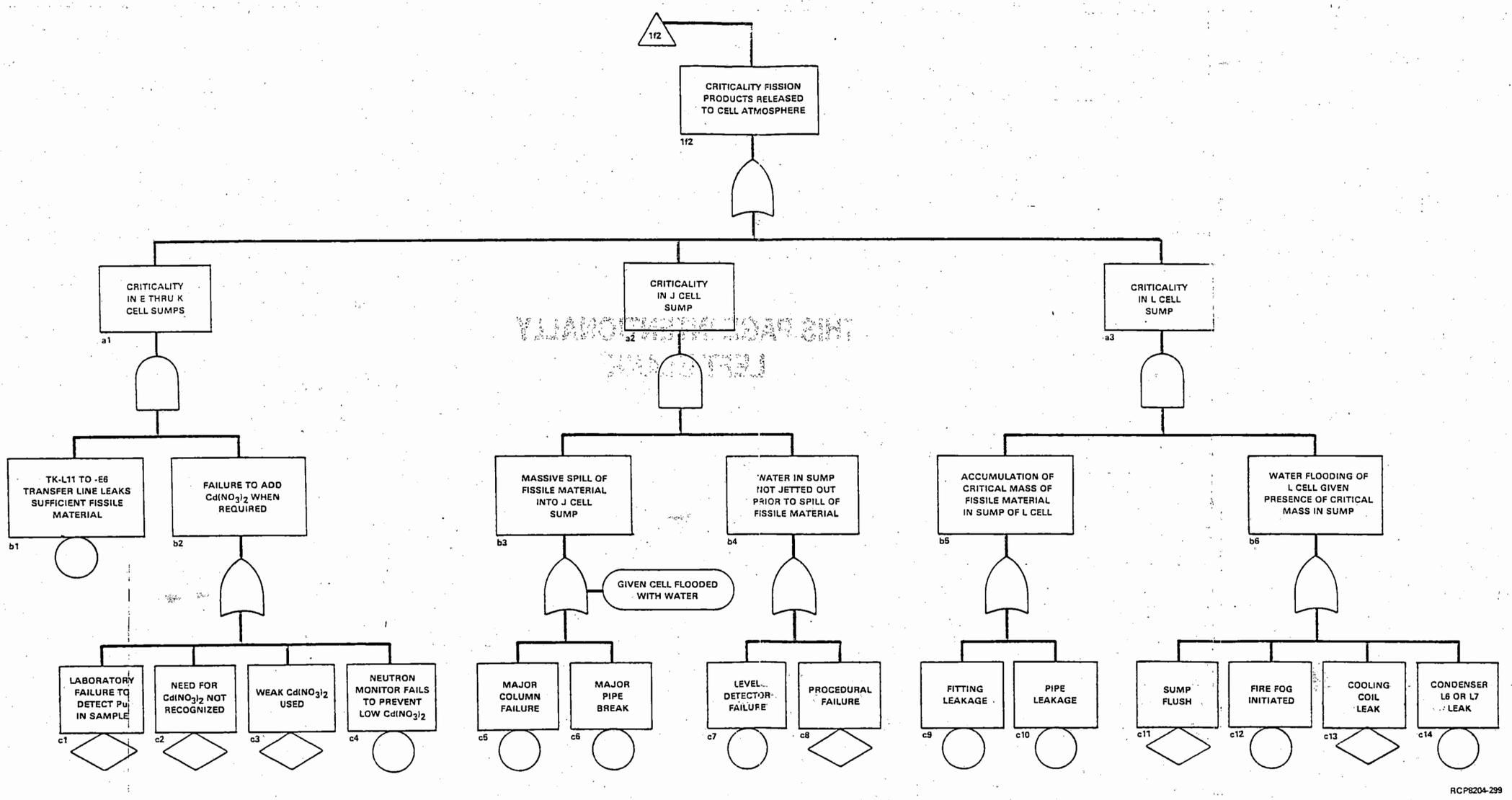


FIGURE D-7. FAULT TREE 6: Criticality Fission Products to Cell Atmosphere.

THIS PAGE INTENTIONALLY
LEFT BLANK

9413219.1282

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK

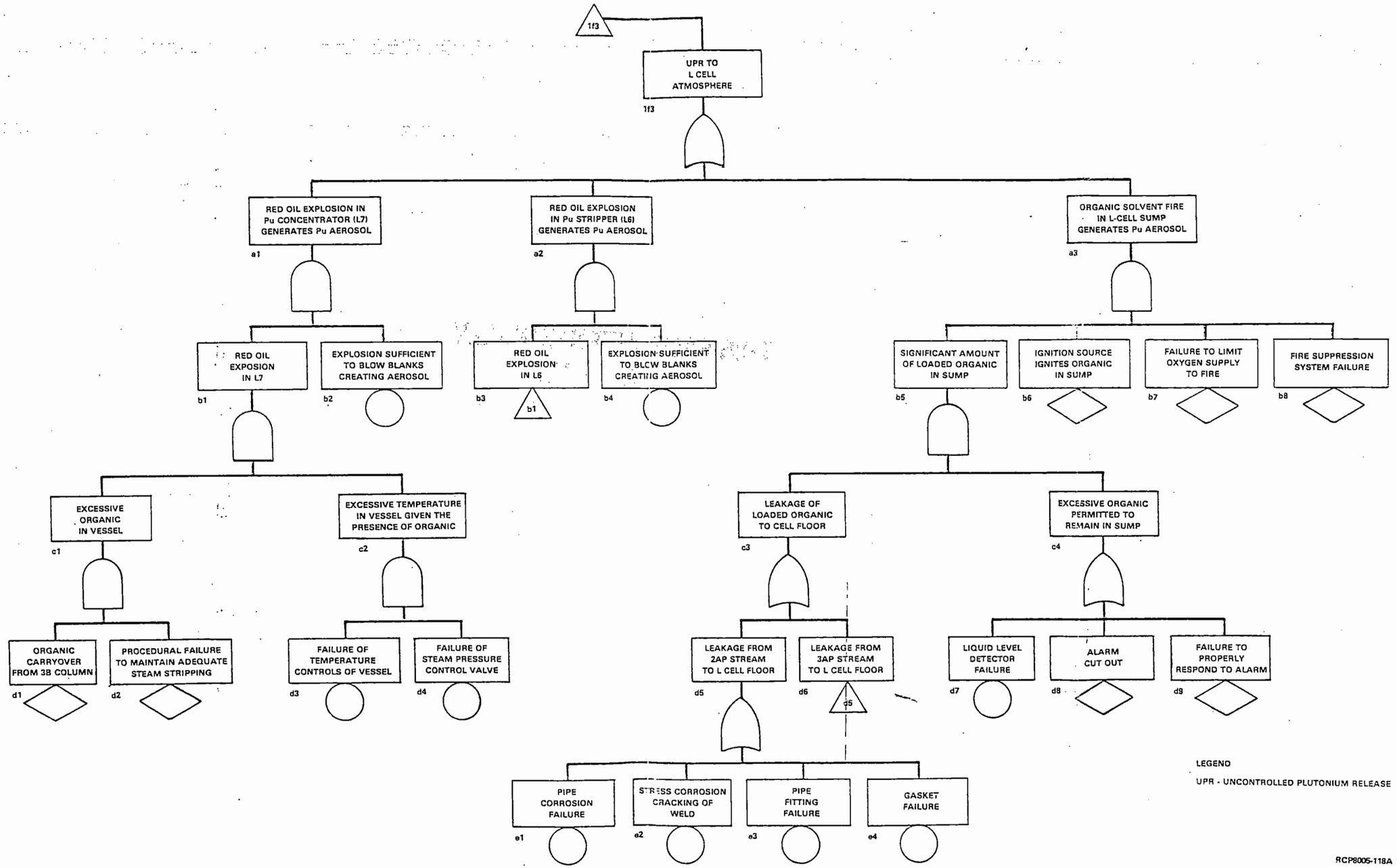


FIGURE D-8. FAULT TREE 7: UPR to L or M Cell Atmosphere.

THIS PAGE INTENTIONALLY
LEFT BLANK

9413219.1284

This page intentionally left blank.

Quantification of 1f3

UPR to L or M Cell Atmosphere

Event	Remarks	Failure frequency (d^{-1})
d1 - Organic carryover from 3B column	Assumed probability of 1.0	1.0
d2 - Procedural failure to maintain adequate steam stripping	Human error rate, failure to follow procedure; (1×10^{-3}/demand) (3 demand/d)	3×10^{-3}
d3 - Failure of temperature controls	General controller failure rate of 3.3×10^{-5}/h; (3.3×10^{-5}/h) (24 h/d)	8.0×10^{-4}
d4 - Failure of steam pressure control valve	Relief valve fails to open; failure rate of 1×10^{-5}/demand; (1×10^{-5}/demand)(3 demand/d)	3×10^{-5}
b2 - Explosion sufficient to blow blanks	Assumed probability of 1.0	1.0
b3 - Red oil explosion in L6	Quantified for b1	
b4 - Explosion sufficient to blow blanks	Assumed probability of 1.0	1.0
e1 - Pipe corrosion failure	General piping failure rate of 1×10^{-9}/h; (1×10^{-9}/h) (24 h/d)	2.4×10^{-8}
e2 - Stress corrosion cracking of weld	See e1	2.4×10^{-8}
e3 - Pipe fitting failure	See e1	2.4×10^{-8}
e4 - Gasket failure	Gasket failure rate of 1×10^{-4}/h; (1×10^{-4}/h)(24 h/d)	2.4×10^{-3}
d6 - Leakage from 3AP stream to L cell floor	Quantified for d5	2.4×10^{-3}

Event	Remarks	Failure frequency (d^{-1})
d7 - Liquid level detector failure	Liquid level detector failure rate of 2×10^{-4}/h for instruments in contact with dirty fluids; (2×10^{-4}/h)(24 h/d)	4.8×10^{-3}
d8 - Alarm cut out	Alarm cutout human error of 1×10^{-2}/demand; (1×10^{-2}/demand) (3 demand/d)	3×10^{-2}
d9 - Failure to properly respond to alarm	See d8	3×10^{-2}
b6 - Ignition source ignites organic in sump	No obvious source of ignition; probability is 1×10^{-2}	1×10^{-2}
b7 - Failure to limit oxygen supply to sump	Assumed to have a probability of 1.0	1.0
b8 - Fire suppression system failure	Fire suppression system failure rate of 5×10^{-2}/demand	5×10^{-2}

$P(1f3) = a1 + a2 + a3$

$P(a1) = (b1)(b2)$

$= (c1)(c2)(b2)$

$= (d1)(d2)(d3)(d4)(b2)$

$= (1.0)(3 \times 10\text{-}3)(8.0 \times 10^{-4})(3 \times 10^{-5})(1.0)$

$P(a1) = 7.2 \times 10\text{-}^{11}/d \times 250\ d = 1.8 \times 10^{-8}/yr$

$P(a2) = P(a1) = 5.4 \times 10^{-9}/yr$

$P(a3) = (b5)(b6)(b7)(b8)$

$= (c3)(c4)(b6)(b7)(b8)$

$= [(d5 + d6)(d7 + d8 + d9)](b6)(b7)(b8)$

$= (e1 + e2 + e3 + e4 + d6)(d7 + d8 + d9)(b6)(b7)(b8)$

$= 2(e1 + e2 + e3 + e4)(d7 + d8 + d9)(b6)(b7)(b8)$

$= 2(2.4 \times 10^{-8} + 2.4 \times 10^{-8} + 2.4 \times 10^{-8} + 2.4 \times 10^{-3})$

$(4.8 \times 10^{-3} + 3 \times 10^{-2} + 3 \times 10^{-2})\ (1 \times 10^{-2})(1.0)(5 \times 10^{-2})$

$= (5 \times 10^{-3})(6.5 \times 10^{-2})(1 \times 10^{-2})(5 \times 10^{-2})$

$P(a3) = 1.6 \times 10^{-7}/d \times 365 = 5.9 \times 10^{-5}/yr$

$P(1f3) = 1.8 \times 10^{-8} + 1.8 \times 10^{-8} + 5.9 \times 10^{-5} = 5.9 \times 10^{-5}/yr$

9413219.1287

0427 6729 1290

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK

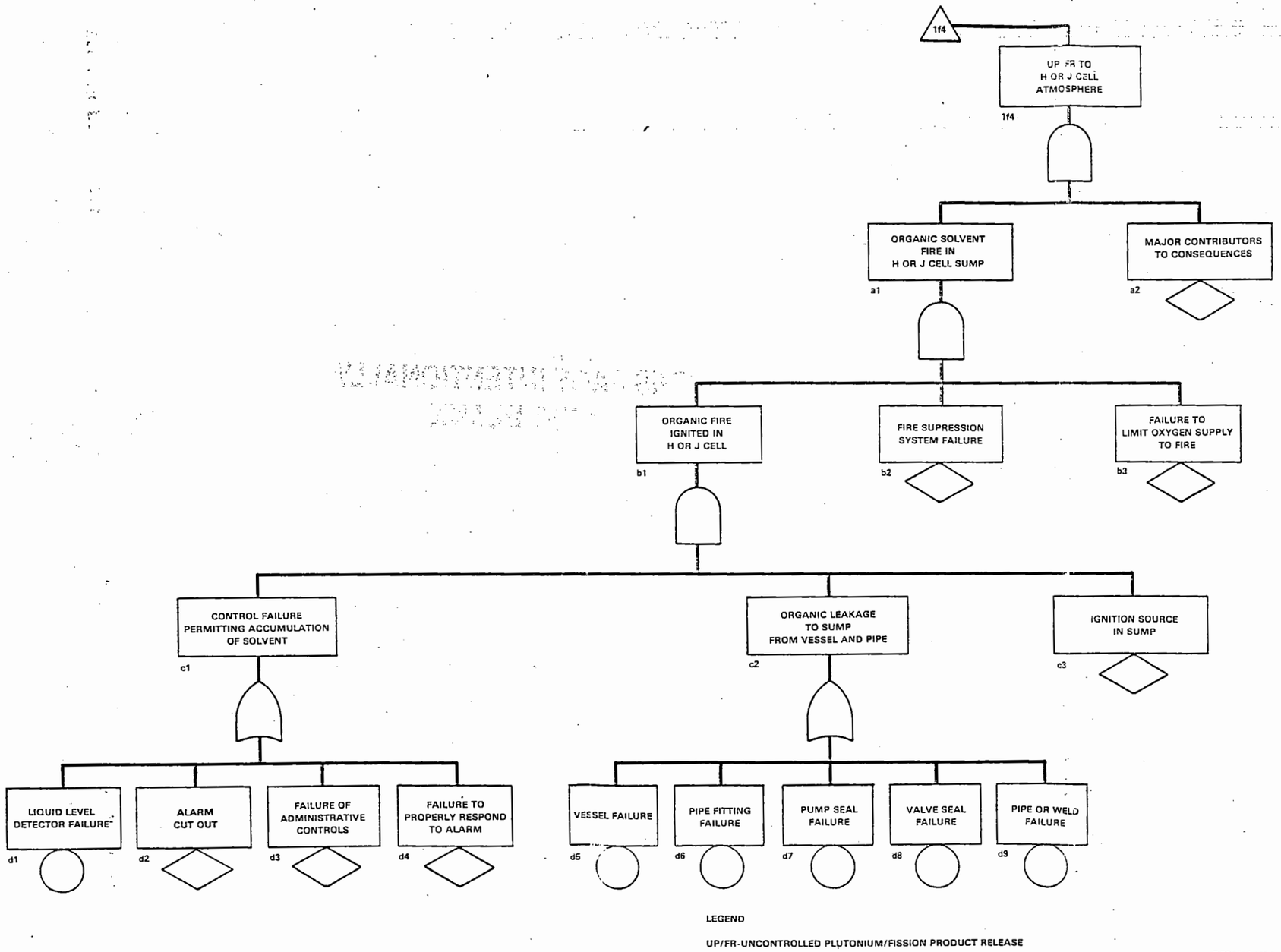


FIGURE D-9. FAULT TREE 8: UP/FR to H or J Cell Atmosphere.

THIS PAGE INTENTIONALLY
LEFT BLANK

9413219.1290

This page intentionally left blank.

Quantification of 1f4

UP/FR to H or J Cell Atmosphere

Event	Remarks	Failure frequency (d^{-1})
d1 - Liquid level detector failure	Failure rate for liquid level detector in contact with dirty fluids is 2×10^{-4}/h; (2×10^{-4}/h) (24 h/d)	4.8×10^{-3}
d2 - Alarm cut out	Alarm cut out; human error rate of 1×10^{-2}/demand; (1×10^{-2}/demand) (3 demand/d)	3×10^{-2}
d3 - Failure of administrative controls	Human error rate for failure to follow procedures is 1×10^{-3}/demand; (1×10^{-3}/demand)(3 demand/d)	3×10^{-3}
d4 - Failure to respond properly to alarm	See d3	3×10^{-3}
d5 - Gasket failure	Gasket failure rate of 1×10^{-4}/h; (1×10^{-4}/h)(24 h/d)(10 gaskets)	2.4×10^{-2}
d6 - Pipe fitting failure	Pipe failure rate of 1×10^{-9}/h; (1×10^{-9}/h)(24 h/d)	2.4×10^{-8}
d7 - Pump seal failure	See d6	2.4×10^{-8}
d8 - Valve seal failure	See d6	2.4×10^{-8}
d9 - Pipe or weld failure	See d6	2.4×10^{-8}
c3 - Ignition source in sump	No obvious source of ignition; probability of 1×10^{-2}	1×10^{-2}
b2 - Fire suppression system	Requires simultaneous failure of two systems; (5×10^{-2})(5×10^{-2})	2.5×10^{-3}
b3 - Failure to limit oxygen supply to fire	Assumed to have a probability to 1.0	1.0
a2 - Major contributor to consequences	Assumed to be present	1.0

$$P(f)1f4 = (a1)(a2)$$

$$= (b1)(b2)(b3)(a2)$$

$$= [(c1)(c2)(c3)](b2)(b3)(a2)$$

$$= [d1 + d2 + d3 + d4][d5 + d6 + d7 + d8 + d9][c3][b2][b3][a2]$$

$$= [4.8 \times 10^{-3} + 3 \times 10^{-2} + 3 \times 10^{-3} + 3 \times 10^{-3}][2.4 \times 10^{-2} + 2.4 \times 10^{-8} + 2.4 \times 10^{-8} + 2.4 \times 10^{-8} + 2.4 \times 10^{-8}][1 \times 10^{-2}][2.5 \times 10^{-3}][1.0][1.0]$$

$$= [4 \times 10^{-2}][2.4 \times 10^{-2}][1 \times 10^{-2}][2.5 \times 10^{-3}][1.0][1.0]$$

$$P(f)1f4 = 2.4 \times 10^{-8} \times 365\ d = 8.6 \times 10^{-6}/yr$$

BL32109.1292

THIS PAGE INTENTIONALLY
LEFT BLANK

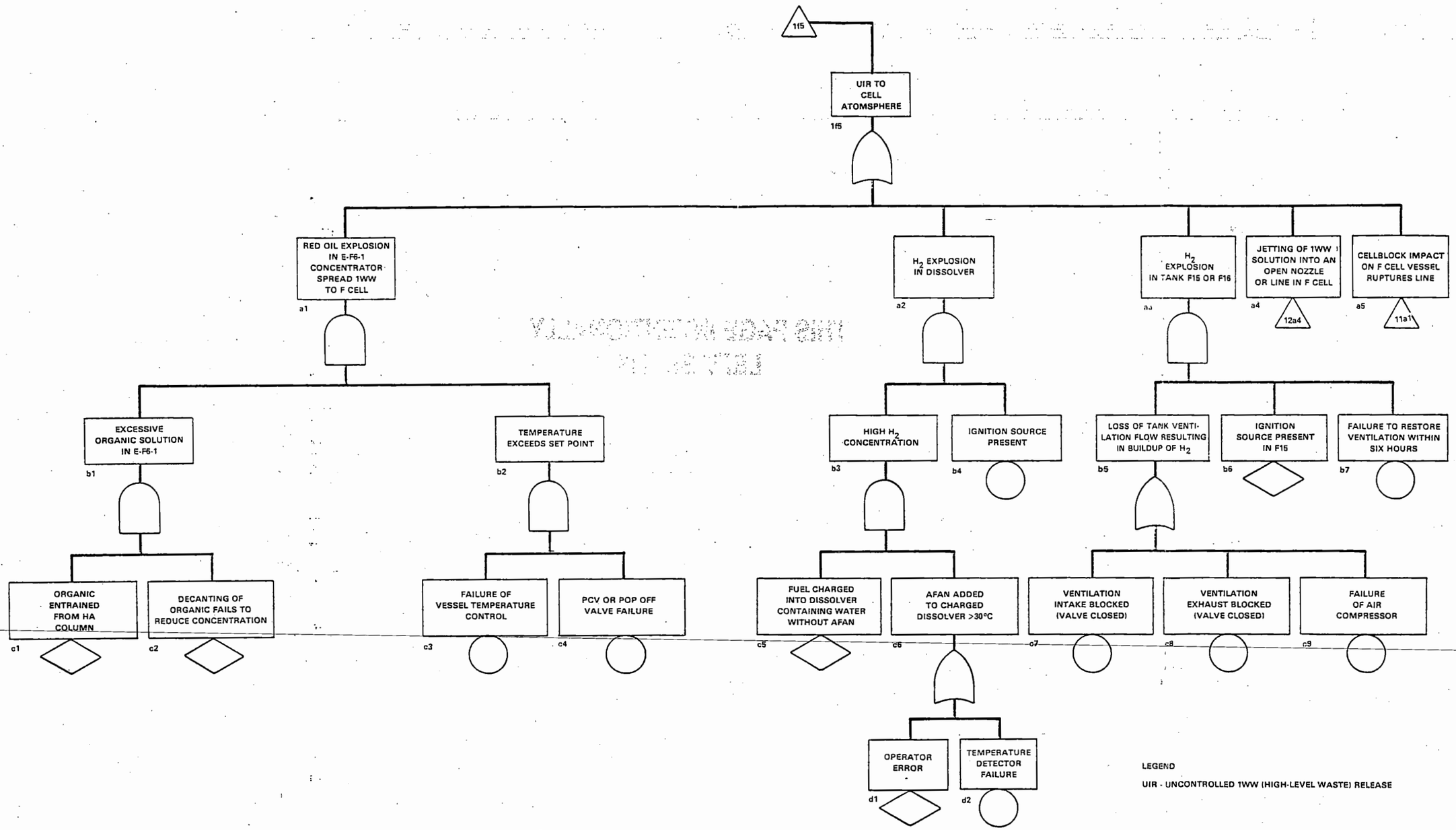


FIGURE D-10. FAULT TREE 9: UIR to Cell Atmosphere.

THIS PAGE INTENTIONALLY
LEFT BLANK

9412162.1204

This page intentionally left blank.

Quantification of 1f5

UIR to Cell Atmosphere

Event	Remarks	Failure frequency (d^{-1})
c1 - Organic entrained from HA column	Assumed to occur	1.0
c2 - Decanting of organic fails to reduce concentration	Human error in controlling decant tank (TK-F7) level; probability of 1×10^{-2}	1×10^{-2}
c3 - Failure of vessel temperature control	General failure rate for controllers of 3.3×10^{-5}/h; (3.3×10^{-5}/h) (24 h/d)	8×10^{-4}
c4 - PCV or pop-off valve failure	Relief valve failure to open 1×10^{-5}/demand; (1×10^{-5}/demand)(10 demands/d)	1×10^{-4}
c5 - Fuel charged into dissolver containing water without AFAN	Failure to follow procedure; human failure rate of 1×10^{-3}/demand; (1×10^{-3}/demand(3 demand/d)	3×10^{-3}
d1 - Operator error	Passive redundancy human error of 5×10^{-3}/demand; (5×10^{-3}/demand) (3 demand/d)	1.5×10^{-2}
d2 - Temperature detector failure	Temperature measurement failure rate of 4×10^{-5}/h. (4×10^{-5}/h) (24 h/d)	1×10^{-3}
b4 - Ignition source	No obvious source of ignition, relative probability of 1×10^{-1}.	1×10^{-1}
c7 - Vent intake blocked (valve shut)	Failure rate of manual valve to remain open. (1×10^{-4}/demand) (3 demand/d)	3×10^{-4}
c8 - Vent exhaust blocked (valve shut)	See c7	3×10^{-4}
c9 - Failure of air compressor	Failure rate for active components. (3×10^{-7}/h)(24 h/d)	7.2×10^{-6}
b6 - Ignition source	See b4	1×10^{-1}

Event	Remarks	Failure frequency (d^{-1})
b7 - Failure to restore vent within 6 h (Time) required to build-up explosive accumulation)	Failure to note loss of tank ventilation; passive redundancy; human error failure rate of 5×10^{-3}/demand; (5×10^{-3}/demand)(1 demand/d)	5×10^{-3}

$p(1f5) = a1 + a2 + a3 + a4 + a5$

$P(a1) = (b1)(b2)$

$= [(c1)(c2)][(c3)(c4)]$

$= [(1.0)(1 \times 10^{-2})][8 \times 10^{-4})(1 \times 10^{-4})]$

$= [1 \times 10^{-2}][8 \times 10^{-8}]$

$p(a1) = 8 \times 10\text{-}10/d \times 250\ d/yr = 2 \times 10^{-7}/yr$

$p(a2) = (b3)(b4)$

$= (c5)(c6)(b4)$

$= (c5)(d1 + d2)(b4)$

$= (3 \times 10^{-3})(1.5 \times 10^{-2} + 1 \times 10^{-3})(1 \times 10^{-1})$

$P(a2) = 4.8 \times 10\text{-}6/d \times 250\ d/yr = 1.2 \times 10^{-3}/yr$

$p(a3) = (b5)(b6)(b7)$

$= (c7 + c8 + c9)(b6)(b7)$

$= (3 \times 10^{-4} + 3 \times 10^{-4} + 7.2 \times 10^{-6})(1 \times 10^{-1})(5 \times 10^{-3})$

$= (6 \times 10^{-4})(1 \times 10^{-1})(5 \times 10^{-3})$

$p(a3) = 3.0 \times 10\text{-}7/d \times 365\ d/yr = 1.1 \times 10^{-4}/yr$

9413219.1296

THIS PAGE INTENTIONALLY
LEFT BLANK

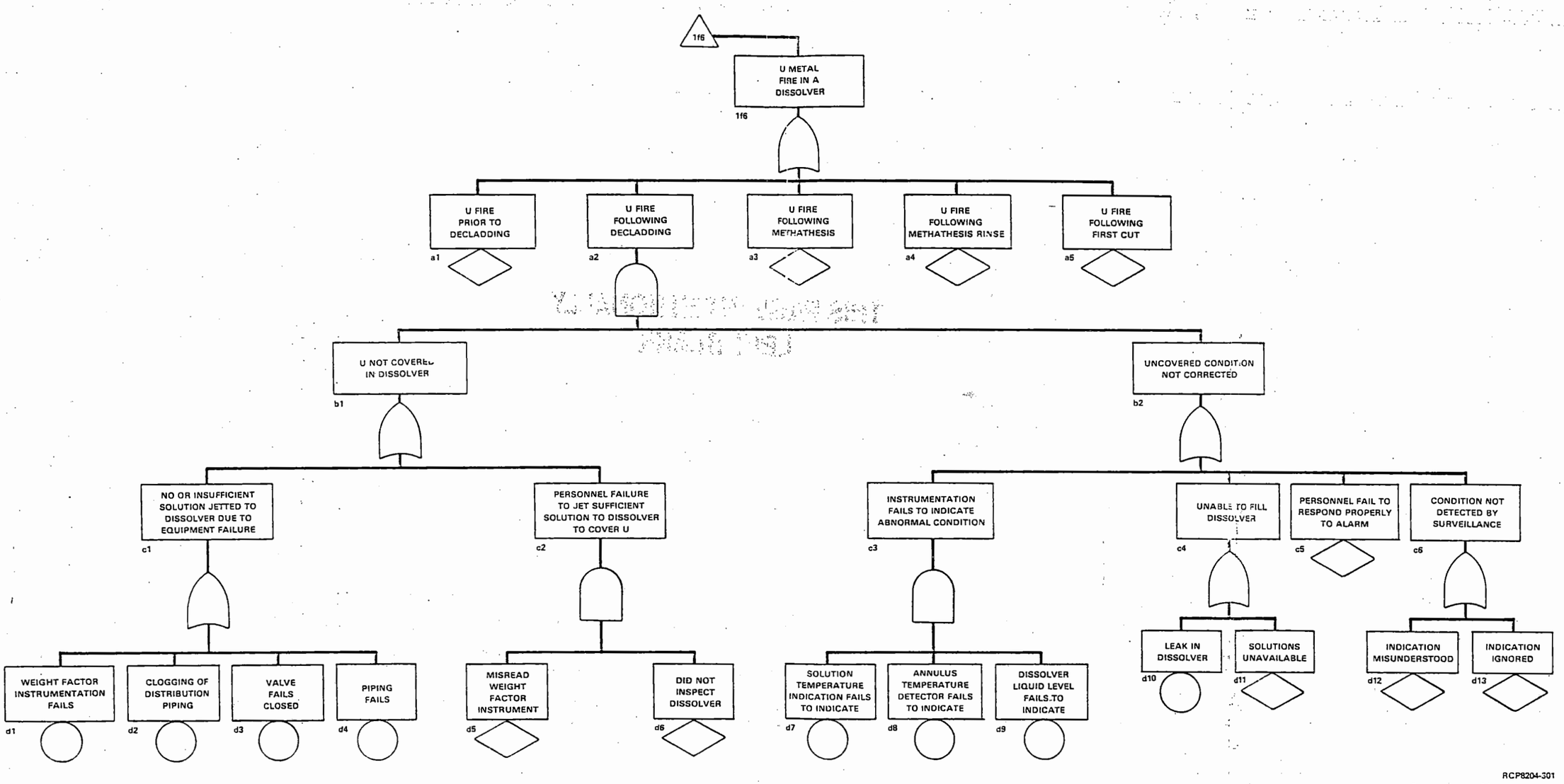


FIGURE D-11. FAULT TREE 10: URR from Uranium Metal Fire in A, B or C Cell.

THIS PAGE INTENTIONALLY
LEFT BLANK

9103219.1298

This page intentionally left blank.

Quantification of 1f6

URR from U Metal Fire in A, B, or C Cell

Event	Remarks	Failure frequency (charge^{-1})
d1 - Weight factor instrument fails	Instrument failure rate of 1.9/yr (1.9/yr) $\frac{1\ yr}{200\ charges}$	1×10^{-2}
d2 - Clogging of distribution piping	10^{-9}/h; (ε)	
d3 - Valve fails closed	1×10^{-4}/demand; (1 demand/charge)	1×10^{-4}
d4 - Piping fails	10^{-9}/h; (ε)	
d5 - Misread weight factor	Passive redundancy 5×10^{-3}/demand; (1 demand/charge)	5×10^{-3}
d6 - Did not inspect dissolver	Passive redundancy 5×10^{-3}/demand; (1 demand/charge)	5×10^{-3}
d7 - Solution temperature detector fails to indicate	Instrument failure rate of 0.35/yr (0.35/yr) $\frac{1\ yr}{200\ charges}$	1.8×10^{-3}
d8 - Annulus temperature detector fails to indicate	See d7	1.8×10^{-3}
d9 - Liquid level fails to indicate	Instrument failure rate of 1.55/yr (1.55/yr) $\frac{1\ yr}{200\ charges}$	7.8×10^{-3}
d10 - Leak in dissolver	ε	
d11 - Solution unavailable	ε	
c5 - Personnel fail to respond properly to alarm	Human error rate failure to follow procedures (10^{-3}/demand)	10^{-3}

9413219.1299

Event	Remarks	Failure frequency ($charge^{-1}$)
d12 - Indication misunderstood	Active redundancy 1×10^{-5}/demand; (1 demand/charge)	1×10^{-5}
d13 - Indication ignored	Active redundancy 1×10^{-5}/demand; (1 demand/charge)	1×10^{-5}
a1 - U Fire prior to decladding	ε	ε
a3 - U Fire following methathesis	ε	ε
a4 - U Fire following methathesis rinse	ε	ε
a5 - U Fire following first cut	ε	ε

$$P(1f6) = a1 + a2 + a3 + a4 + a5$$

$$= a2$$

$$= (b1)(b2)$$

$$= (c1 + c2)(c3 + c4 + c5 + c6)$$

$$= [(d1 + d2 + d3 + d4) + (d5)(d6)]\ \{[(d7)(d8)(d9)] + (d10 + d11) + c5 + (d12 + d13)]\}$$

$$= \{(1 \times 10^{-2} + \varepsilon + 1 \times 10^{-4} + \varepsilon) + [(5 \times 10^{-3})(5 \times 10^{-3})]\}\ [2.5 \times 10^{-8} + (\varepsilon + \varepsilon) + 1 \times 10^{-3} + (1 \times 10^{-5} + 1 \times 10^{-5})]$$

$$= [(1 \times 10^{-2}) + 2.5 \times 10^{-5}][1 \times 10^{-3} + (2 \times 10^{-5})]$$

$$= [1 \times 10^{-2}][1 \times 10^{-3}]$$

$$= 1 \times 10^{-5}$$

$$P(1f6) = 1 \times 10^{-5}/\text{charge} \times 200\ \text{charges/yr} = 2 \times 10^{-3}/\text{yr}$$

THIS PAGE INTENTIONALLY
LEFT BLANK

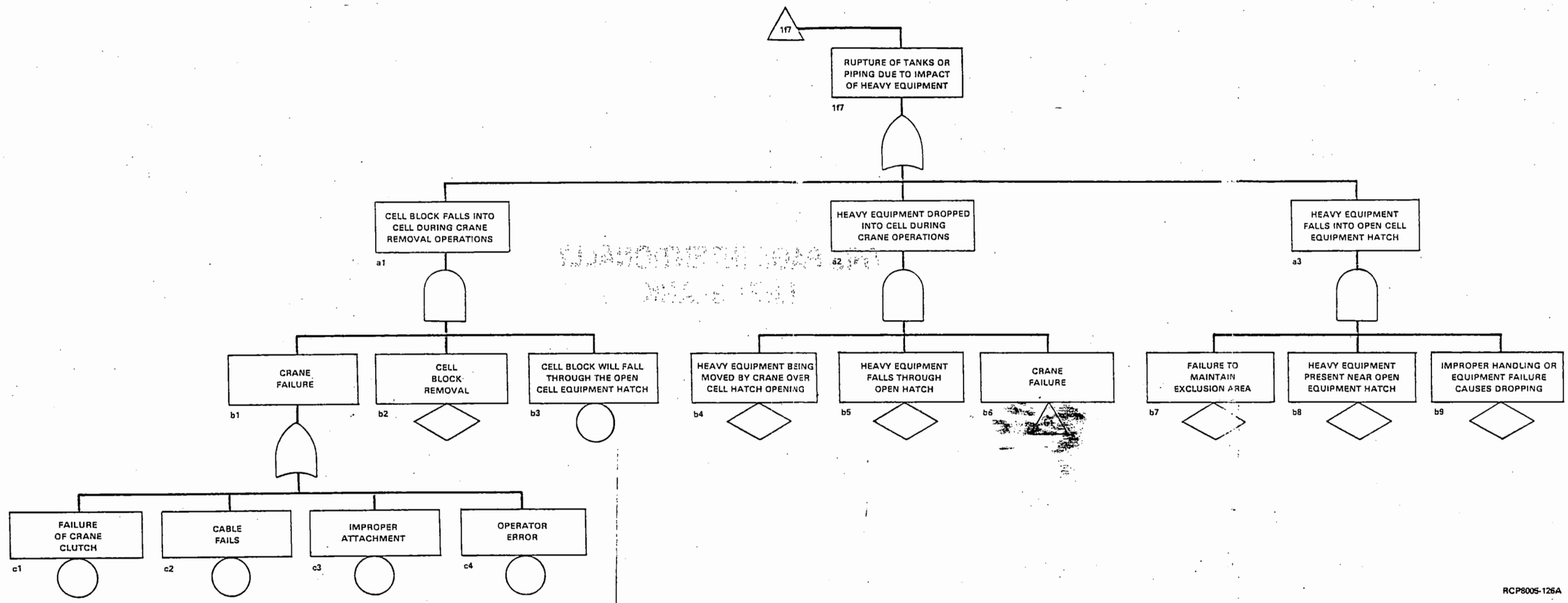


FIGURE D-12. FAULT TREE 11: URR to H Cell Atmosphere Due to Rupture of Tanks or Piping.

THIS PAGE INTENTIONALLY
LEFT BLANK

9413219.1302

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK

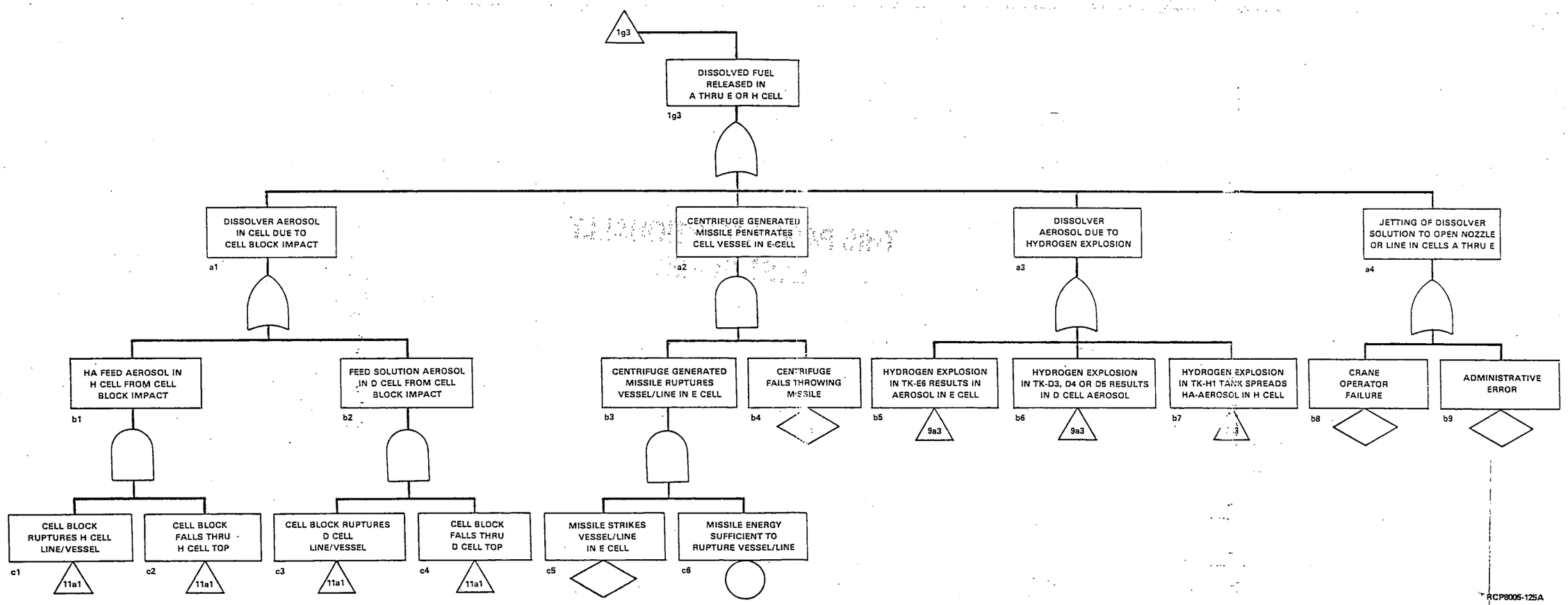


FIGURE D-13. FAULT TREE 12: Dissolved Fuel Released in A Through E Cell.

THIS PAGE INTENTIONALLY
LEFT BLANK

9413219.1304

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK

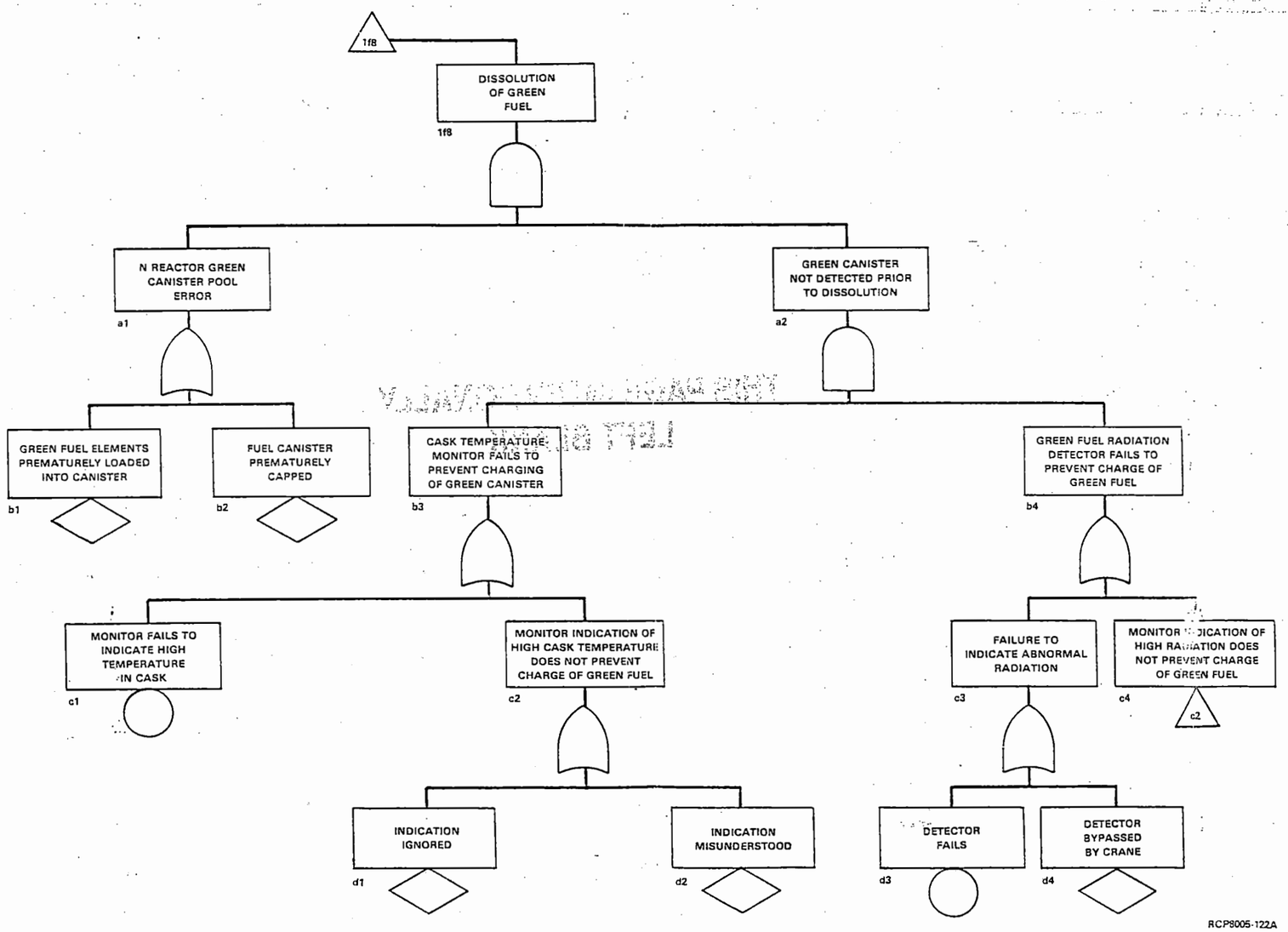


FIGURE D-14. FAULT TREE 13: Dissolution of Green Fuel.

THIS PAGE INTENTIONALLY
LEFT BLANK

This page intentionally left blank.

Quantification of 1f8

Dissolution of Green Fuel

Event	Remarks	Failure frequency (rail car^{-1})
b1 - Green fuel elements prematurely loaded into canister	Active redundancy failure rate of 1×10^{-5}/demand; (1×10^{-5}/demand) (9 demands/rail car)	9×10^{-5}
b2 - Fuel canister prematurely capped	See b1	9×10^{-5}
c1 - Monitor fails to indicate high temperature in cask	Failure rate for temperature measurement is 0.35/yr; (.35/yr/732 demand/yr	5×10^{-4}
d1 - Indicator ignored	Human error failure rate of 1×10^{-2}/demand; (1×10^{-2}/demand) (1 demand/rail car)	1×10^{-2}
d2 - Indication misunderstood	See d1	1×10^{-2}
d3 - Green fuel detector fails	Failure rate for general solid state devices including ion chambers is 3×10^{-6}/h; (3×10^{-6}/h) (2 h/rail car)	6×10^{-6}
d4 - Detector bypassed by crane	Design precludes possibility;	
c4 - Monitor indication does not prevent charge of green fuel	Green fuel monitor alarms in three locations: the head end control room, dispatcher's office, and crane cab; each location is manned while fuel is being charged; active redundancy failure rate of 1×10^{-5}/demand is used; (1×10^{-5}/demand)(9 demands/rail car)	9×10^{-5}

9413219.1307

$$P(1f8) = (a1)(a2)$$

$$= [b1 + b2][b3][b4]$$

$$= [b1 + b2][(c1 + c2)(c3 + c4)]$$

$$= [b1 + b2][(c1 + d1 + d2)(d3 + d4 + c4)]$$

$$= (9 \times 10^{-5} + 9 \times 10^{-5})(5 \times 10^{-4} + 1 \times 10^{2} + 1 \times 10^{-2})$$

$$(6 \times 10^{-6} + \varepsilon + 9 \times 10^{-5})$$

$$P(1f8) = 3.5 \times 10^{-10}/\text{rail car} \times 732 \text{ rail car/yr} = 2.5 \times 10^{-7}/\text{yr}$$

BDC1 9/26/16

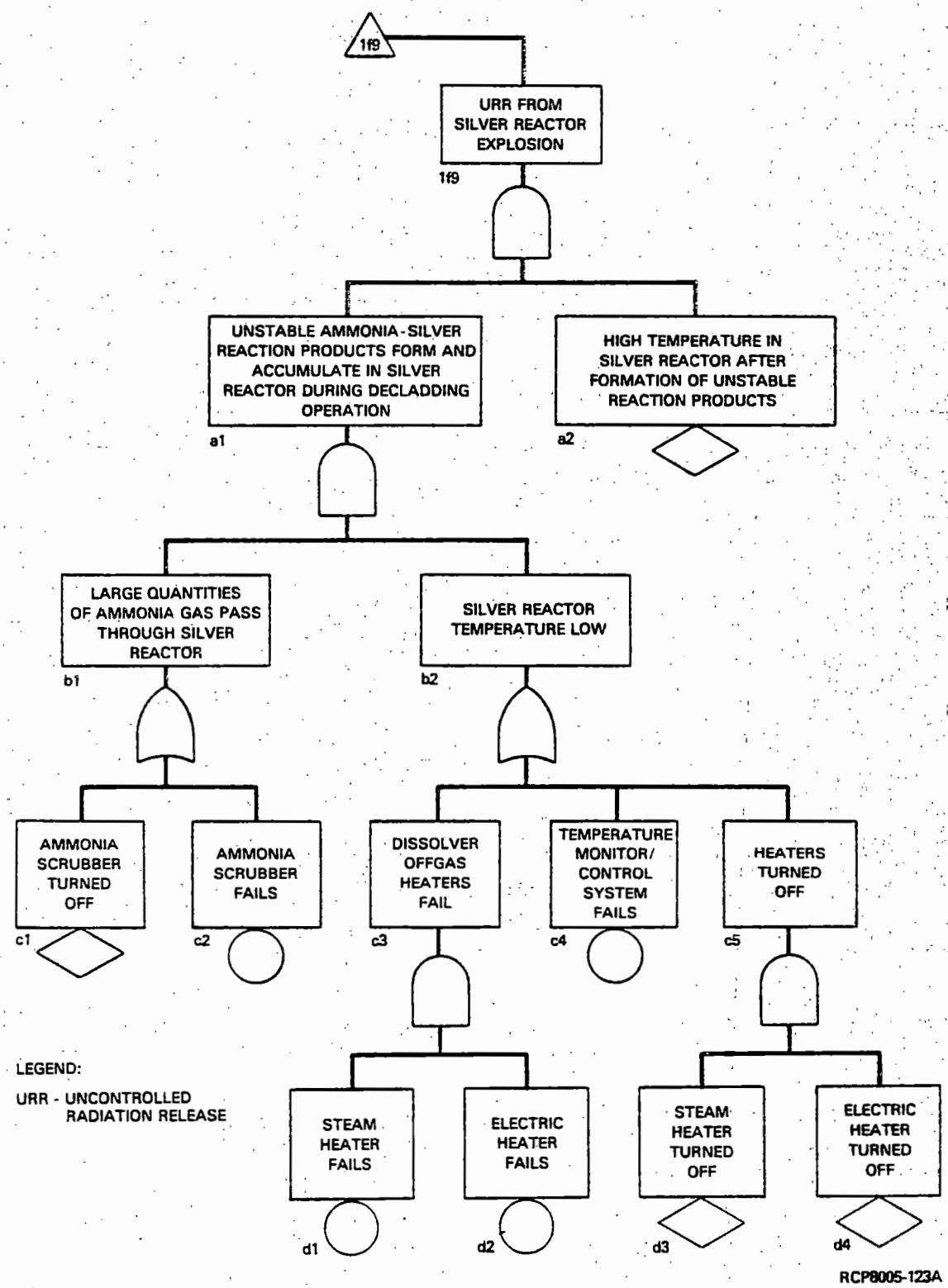


FIGURE D-15. FAULT TREE 14: URR from Silver Reactor Explosion.

Quantification of 1f9

URR From Silver Reactor Explosion

Event	Remarks	Failure frequency (declad cycle^{-1})
c1 - Ammonia scrubber turned off	Passive redundancy human error failure rate of 5×10^{-3}/demand; (5×10^{-3}/demand)(1 demand/declad cycle)	5×10^{-3}
c2 - Ammonia scrubber fails	Similar to heat exchanger failure rate of 2.5×10^{-5}/h; (2.5×10^{-5}/h)(6 h/declad cycle)	1.5×10^{-4}
d1 - Steam heater fails	See c2	1.5×10^{-4}
d2 - Electric heater fails	General electrical heater failure rate of 2.3×10^{-6}/h; (2.3×10^{-6}/h)(6 h/declad cycle)	1.4×10^{-5}
c4 - Temperature monitor control system fails	General failure rate for controllers failure rate of 3.3×10^{-5}/h; (3.3×10^{-5}/h)(6 h/declad cycle)	2×10^{-4}
d3 - Steam heater turned off	Passive redundancy human error failure rate of 5×10^{-3}/demand; (5×10^{-3}/demand)(1 demand/declad cycle)	5×10^{-3}
d4 - Electric heater turned off	See d3	5×10^{-3}
a2 - High temperature in Silver Reactor after formation of unstable reaction products	Human error rate of 0.2/demand for stressful situations	0.2

$$P(1f9) = (a1)(a2)$$

$$= [(b1)(b2)][a2]$$

$$= [(c1 + c2)(c3 + c4 + c5)][a2]$$

$$[(c1 + c2)][(d1)(d2) + c4 + (d3)(d4)][a2]$$

$$= [(5 \times 10^{-3} + 1.5 \times 10^{-4})[(1.5 \times 10^{-4})(1.4 \times 10^{-5}) + 2.0 \times 10^{-4} +$$

$$(5 \times 10^{-3})(5 \times 10^{-3})][.2]$$

$= [5.1 \times 10^{-3}][2.0 \times 10^{-4}][.2]$

$= 2.0 \times 10^{-7}$/declad cycle

$$\left\{2.0 \times 10^{-7} \frac{\text{failures}}{\text{declad cycle}}\right\} \left\{2 \times 10^{2} \frac{\text{declad}}{\text{year}}\right\}$$

$= 4 \times 10^{-5}$ failures/yr

0432019.311

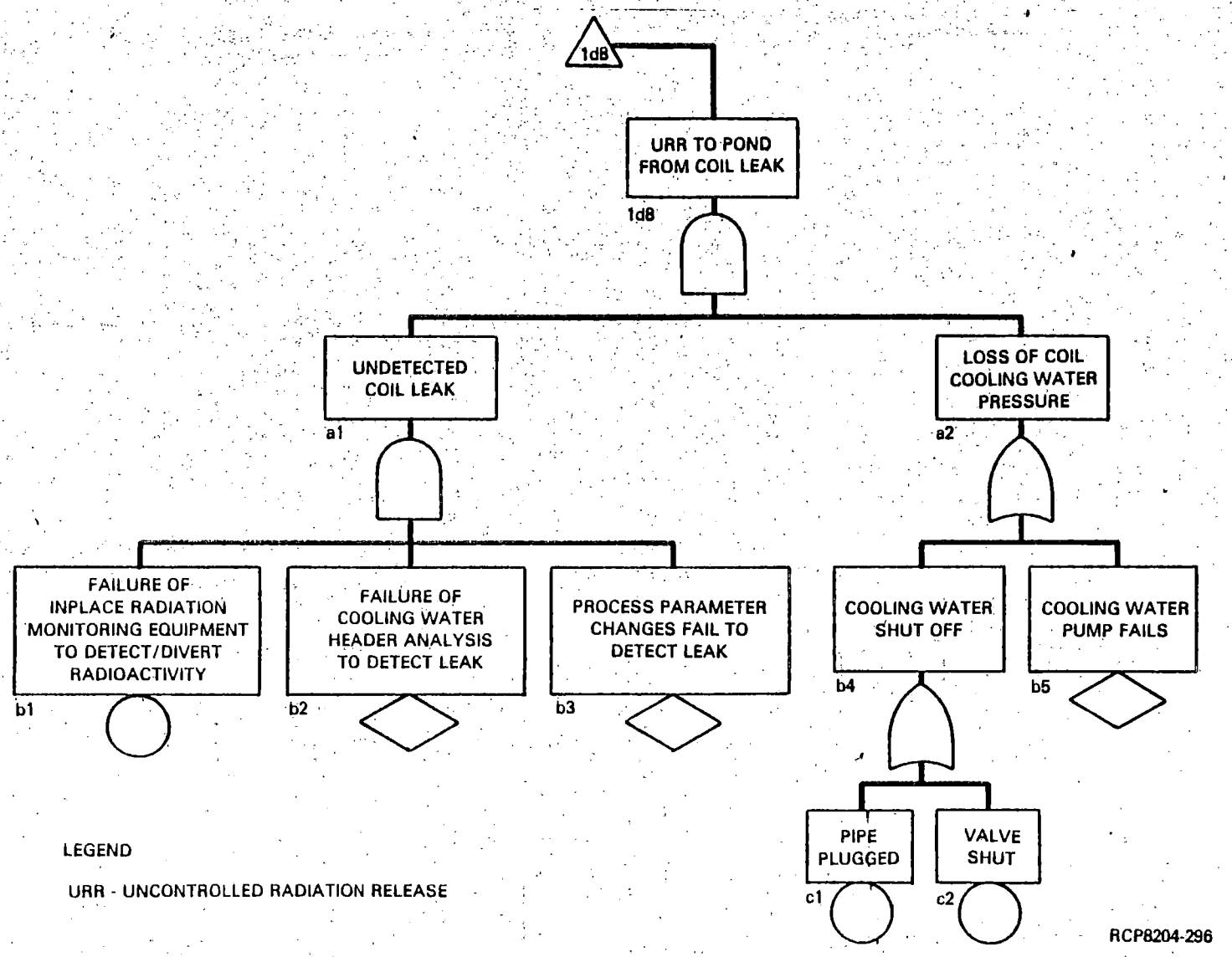


FIGURE D-16: FAULT TREE 15: URR to Pond from Coil Leak.

Quantification of 1d8

URR to Pond From Coil Leak

Event	Remarks	Failure frequency (d^{-1})
b1 - In place radiation detector fails to detect/divert	Failure rate for solid state devices; $(3 \times 10^{-6}/h)(24\ h)$	7.2×10^{-5}
b2 - Failure of cooling water header analysis to prevent release of radioactivity	Cooling water is analyzed once each 7 d, hence, failure of the analysis is not required to allow a release of less than 7 d duration	1.0
b3 - Process parameter changes fail to prevent release of radioactivity	For any but the largest leaks, probability of detection by changes in process parameters is very low	1.0
c1 - Pipe plugged	Pipe plug failure rate of; $(1 \times 10^{-9}/h)(24\ h)$	2.4×10^{-8}
c2 - Valve shut	Valve has 11.4 L/min bypass which precludes total shutoff	0.0
b5 - Cooling water pump fails	Electrical pump is backed up by a steam driven pump. Failure rate is assessed as that for general active components, $(3 \times 10^{-7}/h)(24\ h)$	7.2×10^{-6}

$$P(1d8) = (a1)(a2)$$

$$= [(b1)(b2)(b3)][b4 + b5]$$

$$= [(b1)(b2)(b3)][(c1 + c2) + b5]$$

$$= [(7.2 \times 10^{-5})(1)(1)][3 \times 10^{-8} + 0 + 7.2 \times 10^{-6}]$$

$$P(1d8) = 5.2 \times 10^{-10}/d \times 250 = 1.3 \times 10^{-7}/yr$$

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK

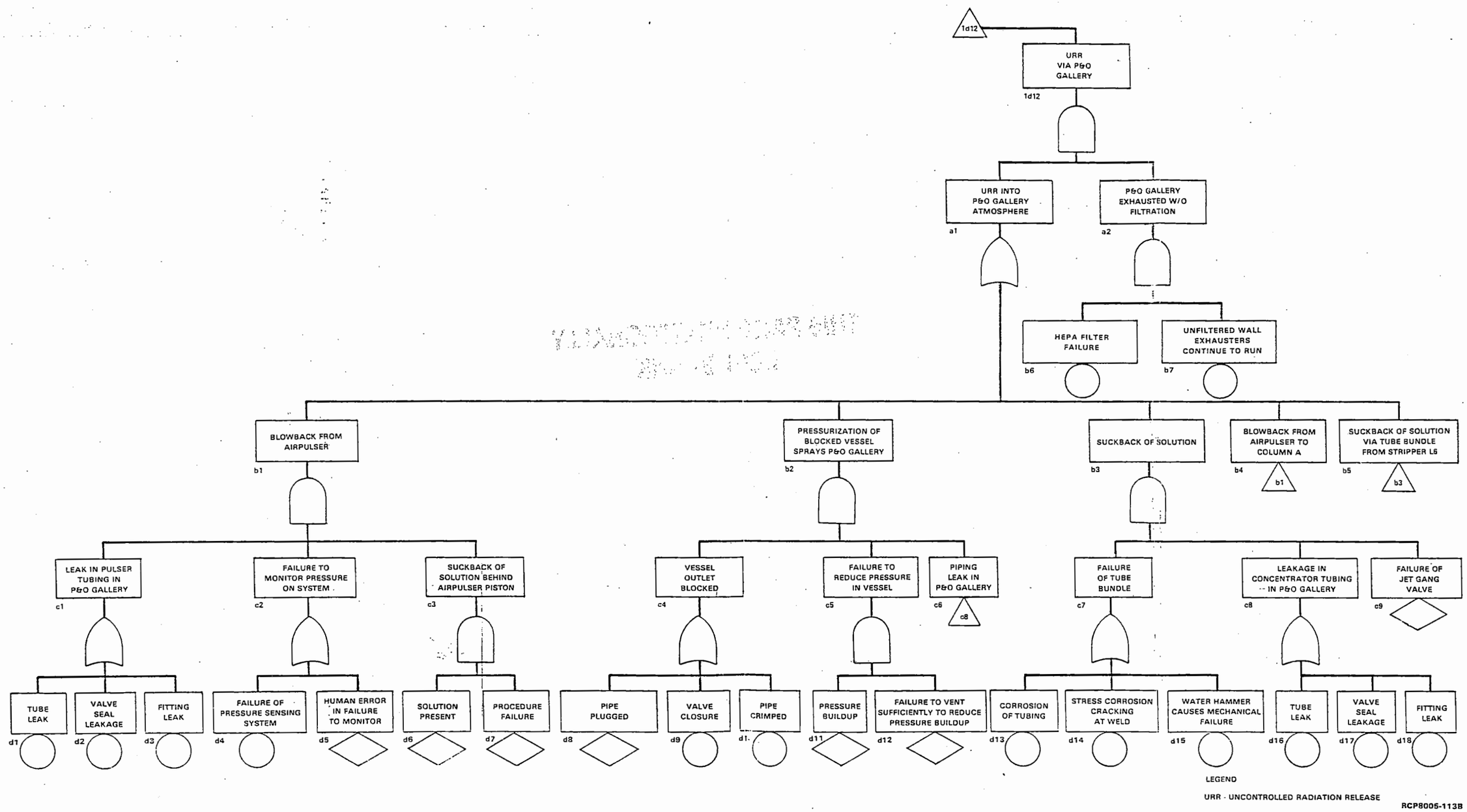


FIGURE D-17. FAULT TREE 16: URR from P&O Gallery.

THIS PAGE INTENTIONALLY
LEFT BLANK

This page intentionally left blank.

Quantification of 1c9

URR via P&O Gallery

Event	Remarks	Failure frequency (d^{-1})
d1 - Tube leak	Failure rate of 5×10^{-4}/h; (5×10^{-4}/h)(24 h/d)	1.2×10^{-2}
d2 - Valve leak	See d1	1.2×10^{-2}
d3 - Fitting leak	See d1	1.2×10^{-2}
d4 - Failure of pressure sensing system	Failure rate for pressure detector systems in contact with dirty fluids 2×10^{-4}/h; (2×10^{-4}/h) (24 h/d)	4.8×10^{-3}
d5 - Human error in failure to monitor	Passive redundancy; human error failure rate of 5×10^{-3}/demand; (5×10^{-3}/demand)(3 demand/d)	1.5×10^{-2}
d6 - Pu solution present in 3B column	Assumed	1.0
d7 - Procedure failure	Failure to follow procedure failure rate of 1×10^{-3}/demand; (1×10^{-3}/demand)(3 demands/d)	3×10^{-3}
d8 - Pipe plugged	Failure rate of 1×10^{-9}/h; (1×10^{-9}/h)(24 h/d)	2.4×10^{-8}
d9 - Valve closed	Failure rate of 1×10^{-4}/demand; (1×10^{-4}/demand)(3 demands/d)	3×10^{-4}
d10 - Pipe crimped	See d8	2.4×10^{-8}
d11 - Pressure buildup	Minor transient pressure increases may occur with some frequency. A significant pressure excursion would most likely only occur as a result of a major uncontrolled chemical reaction, such as a solvent fire, H_2 or red oil explosion. An additional source of pressure may come from maintenance activities involving efforts to clear blocked lines with steam or air pressure. A pressurization of this magnitude is estimated to occur at a rate of 1/yr or 2.7×10^{-3}/d	2.7×10^{-3}

Event	Remarks	Failure frequency (d^{-1})
d12 - Failure to vent sufficiently	It is estimated that of the significant pressure excursions, the vessel vent systems will be unable to provide sufficient vent capacity at a rate of 1×10^{-1}/demand; (1×10^{-1}/demand) (2.7×10^{-3}/demand/yr	2.7×10^{-4}
c6 - Piping leak in P&O gallery	Quantified for c8	3.6×10^{-2}
d13 - Tubing corrosion	See d1	1.2×10^{-2}
d14 - Weld stress corrosion	See d1	1.2×10^{-2}
d15 - Water hammer causes mechanical failure	See d1	1.2×10^{-2}
d16 - Tube leak	See d1	1.2×10^{-2}
d17 - Valve seal leakage	See d1	1.2×10^{-2}
d18 - Fitting leakage	See d1	1.2×10^{-2}
c9 - Failure of jet gang valve	General active component failure rate of 1×10^{-4}/demand (1×10^{-4}/demand)(3 demands/d)	3×10^{-4}
b4 - Blowback from airpulser to column 3A	Quantified for b1	
b5 - Suckback of solution via tube bundle from stripper L6	Quantified for b3	

Event	Remarks	Failure frequency (yr^{-1})
b6 - Failure of white room HEPA filter	HEPA filter failure rate of 1.1×10^{-4}/yr. For a 12 HEPA filter bank, the failure rate is (12)(1×10^{-4}/yr) = 1×10^{-3}/yr	1×10^{-3}

Event	Remarks	Failure frequency (yr^{-1})
b7 - Unfiltered wall exhausters continue to run	Based upon the redundancy of CAM's automatic alarm upon CAM failure or power loss, and probable visual observation of URR into P&O gallery, it is highly unlikely that the wall exhausters will continue to run indefinitely	ε

$P(1d12) = (a1)(a2)$

$P(a1) = (b1 + b2 + b3 + b4 + b5)$

$= [(c1)(c2)(c3) + (c4)(c5)(c6) + (c7)(c8)(c9) + b4 + b5]$

$= \{[d1 + d2 + d3][d4 + d5][(d6)(d7)]\} + \{[d8 + d9 + d10][(d11)(d12)][c6]\} + \{[d13 + d14 + d15][d16 + d17 + d18][c9]\} + b4 + b5$

$= \{[1.2 \times 10^{-2} + 1.2 \times 10^{-2} + 1.2 \times 10^{-2}][4.8 \times 10^{-3} + 1.5 \times 10^{-2}][(1.0)(3 \times 10^{-3})]\} + \{[(2.4 \times 10^{-8} + 3 \times 10^{-4} + 2.4 \times 10^{-8}][(2.7 \times 10^{-3})(2.7 \times 10^{-4})(3.6 \times 10^{-2})]\} + \{[1.2 \times 10^{-2} + 1.2 \times 10^{-2} + 1.2 \times 10^{-2}][1.2 \times 10^{-2} + 1.2 \times 10^{-2}\ 1.2 \times 10^{-2}][3 \times 10^{-4}]\} + b4 + b5$

$= \{[3.6 \times 10^{-2}][2 \times 10^{-2}][3 \times 10^{-3}]\} + \{[3 \times 10^{-4}][2.6 \times 10^{-8}]\} + \{[3.6 \times 10^{-2}][3.6 \times 10^{-2}][3 \times 10^{-4}]\} + b4 + b5$

$= \{[2.1 \times 10^{-6}] + [7.8 \times 10^{-12}] + [3.9 \times 10^{-7}]\} + b4 + b5$

$= (2.1 \times 10^{-6}) + (7.8 \times 10^{-12}) + (3.9 \times 10^{-7}) + (2.1 \times 10^{-6}) + (3.9 \times 10^{-7})$

$P(a1) = 5.0 \times 10^{-6}/d \times 365\ d/yr = 1.8 \times 10^{-3}/yr$

$P(a2) = b6 + b7$

$= (1 \times 10^{-3}/yr) + (\varepsilon) = 1 \times 10^{-3}/yr$

$P(1d12) = (a1)(a2)$

$= (1.8 \times 10^{-3}/yr)(1 \times 10^{-3}/yr)$

$= 1.8 \times 10^{-6}/yr$

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK

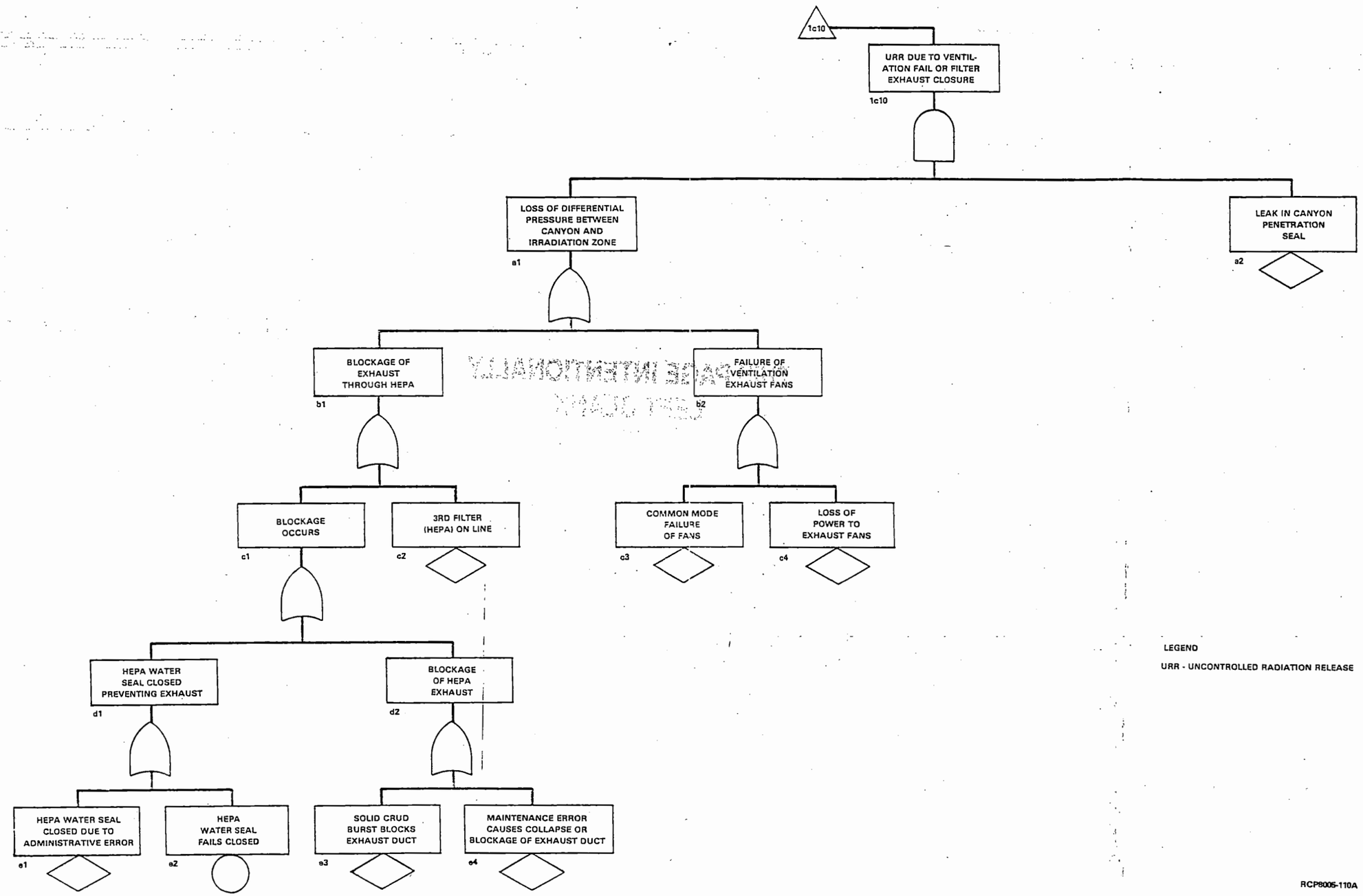


FIGURE D-18. FAULT TREE 17: URR from Sudden HEPA Filter Closure or Failure.

THIS PAGE INTENTIONALLY
LEFT BLANK

9413219.1322

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK

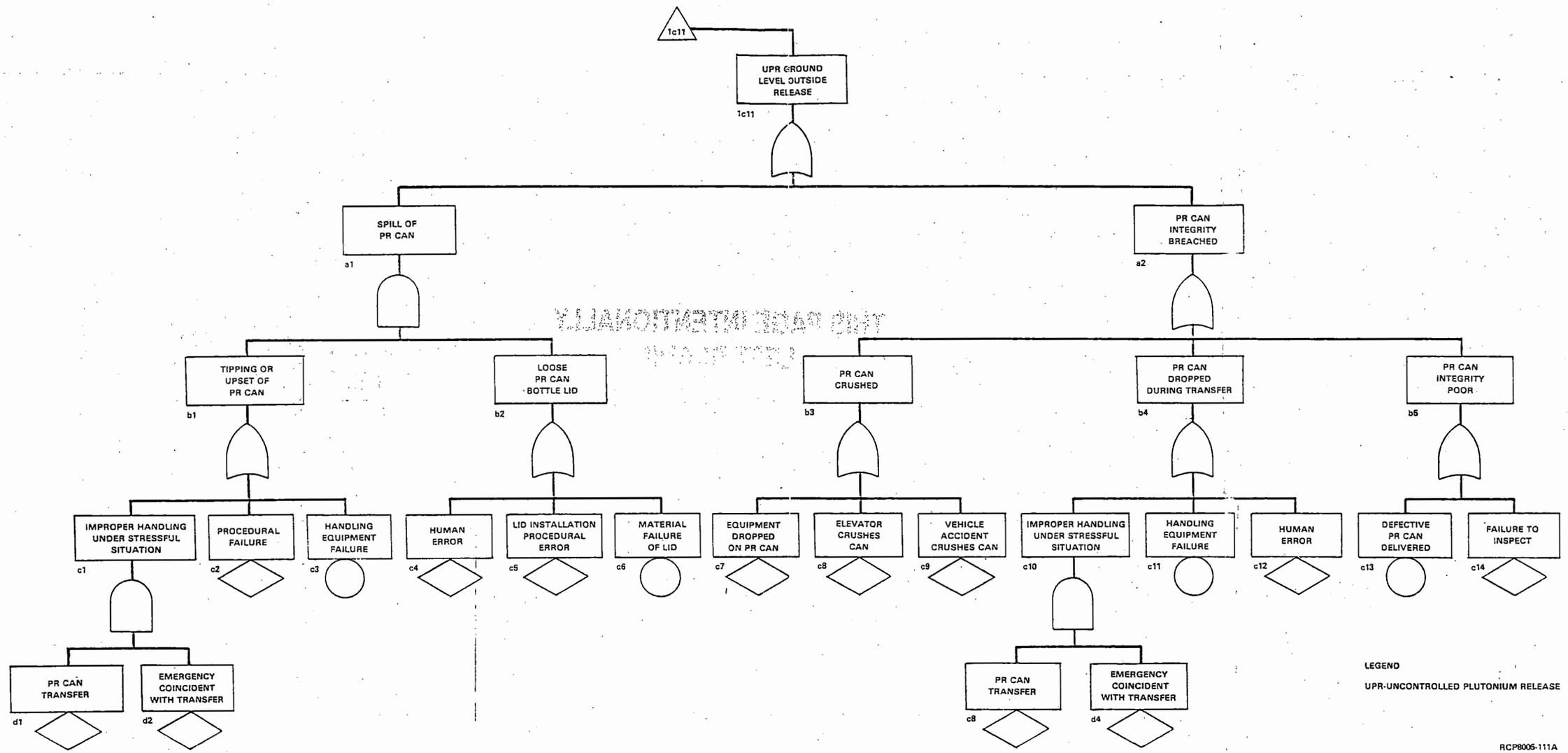


FIGURE D-19. FAULT TREE 18: UPR Ground Level Outside Release.

THIS PAGE INTENTIONALLY
LEFT BLANK

9413262.1434

This page intentionally left blank.

REFERENCES

1. Reactor Safety Study, WASH-1400 (NUREG-75/014) (October 1975).

2. S. N. Anyakara, C. F. M. Engle, and F. P. Lees, The Chemical Engineer (November 1971).

3. Component Failure-Rate Data with Potential Applicability to the Hot Experimental Facility, DPST-CFRP-80-113, E. I. duPonte de Nemours and Company, Savannah River Laboratory, Aiken, South Carolina (December 1980).

4. A. E. Green and A. J. Bourne, Reliability Technology, John Wiley New York (1972).

5. J. P. Barris and D. L. Conte, Fire Journal, Russian (May 1980).

6. "Status Report on the EPRI Fuel Cycle Accident Risk Assessment," EPRI NP-1128, Science Applications, Inc., Palo Alto, California.

7. Fault Tree Handbook, NUREG-0492, Draft (March 1980).

8. R. Browning, "Relative Probabilities of Loss Incidents," pp. 134, Chemical Engineering (December 15, 1969).

9. DOE, Final Environmental Statement, Waste Management Operations, Hanford Reservation, Richland, Washington, ERDA-1538, United States Energy Research and Development Administration (December 1975).

10. Rockwell, Tornado-Induced Damages to the PUREX Plant and the Resulting Offsite Radiological Consequences, RHO-CD-1551, Rockwell Hanford Operations, Richland, Washington (October 1981).

11. Rockwell, PUREX Plant Seismic Hazards Onsite Risk Analysis, RHO-CD-1550, Rockwell Hanford Operations, Richland, Washington (October 1981).

12. Rockwell, PUREX Plant Seismic Hazards Risk Analysis and Cost Benefit Studies Based on Offsite Dose Commitments, RHO-CD-1483, Rockwell Hanford Operations, Richland, Washington (October 1981).

9206.0124.10

This page intentionally left blank.

THIS PAGE INTENTIONALLY
LEFT BLANK