ORNL/Sub-80/31048/1
(SAI-099-78-PA)
(Appendix to ORNL/TM-6986)

ACTINIDE PARTITIONING-TRANSMUTATION PROGRAM FINAL REPORT.

VI. SHORT-TERM RISK ANALYSIS OF REPROCESSING,
REFABRICATION, AND TRANSPORTATION (APPENDIX)

R. R. Fullwood
R. Jackson
with contributions by
other SAI staff

January 1980

Report prepared by

SCIENCE APPLICATIONS, INC.
Palo Alto, California 94304

Under Purchase Order No. 85B31048X-07

for

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830

operated by

UNION CARBIDE CORPORATION
NUCLEAR DIVISION

for the

DEPARTMENT OF ENERGY

Under Contract No. W-7405-eng-26

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

<u>Section</u>	<u>Page</u>
ACKNOWLEDGEMENTS-----	viii
ABSTRACT-----	ix
1. RESULTS-----	1-1
2. OVERVIEW AND QUALIFICATIONS-----	2-1
2.1 Objectives of This Work-----	2-1
2.2 Comparison of PT and Reference LWR Fuel Cycles-----	2-2
2.3 Cost-Risk-Benefit Concept-----	2-6
2.4 Relationship of Present Work to Overall Analysis-----	2-10
2.5 General Analytical Approach-----	2-11
2.6 Qualifying Remarks-----	2-13
2.6.1 Fuel Cycle-----	2-13
2.6.2 Facilities of Interest-----	2-13
2.6.3 Accidents Not Analyzed-----	2-14
3. RISK ANALYSIS-----	3-1
3.1 Risk of the FRP-----	3-1
3.1.1 Accident Probabilities and Consequences-----	3-1
3.1.1.1 Overview and Preliminary Hazards Analysis-----	3-1
3.1.1.2 Hydrogen Fire and Explosion in the HAF Tank-----	3-5
3.1.1.3 Solvent Fire in the HA Contactor-----	3-7
3.1.1.4 Red Oil Explosion in HLW Concentrator--	3-10
3.1.1.5 Explosion in the HLW Calciner-----	3-12
3.1.1.6 Red Oil Explosion in the Fuel Product Concentrator-----	3-15
3.1.1.7 Explosion in the Fuel Product Denitrator-----	3-16
3.1.1.8 Criticality in a Process Cell-----	3-18
3.1.1.9 Failure of a Krypton Storage Cylinder--	3-21
3.1.1.10 Hydrogen Explosion in Uranium Recutor--	3-21
3.1.1.11 Fuel Assembly Drop-----	3-23
3.1.1.12 Hydrogen Explosion in Fuel Product Denitrator Feed Tank-----	3-24
3.1.1.13 Accident Summary-----	3-24
3.1.2 Relative Public Atmospheric Risk of PT and Reference FRP-----	3-31
3.1.3 Relative Occupational Radiological Risks of PT and Reference FRP-----	3-31

SectionPage

3.1.4	Relative Nonradiological Risks of PT and Reference FRP-----	3-32
3.2	Risk of the FRP-WTF-----	3-33
3.2.1	FRP-WTF Dominant Accidents-----	3-33
3.2.1.1	Overview and Preliminary Hazards Analysis-----	3-33
3.2.1.2	Earthquake in Excess of Design Basis----	3-39
3.2.1.3	Waste Treatment Facility Criticality----	3-41
3.2.1.4	Radiolytic Hydrogen Explosion-----	3-42
3.2.1.5	Resin Explosion in Actinide Product Concentrator-----	3-49
3.2.1.6	"Red Oil" Explosions in WTF Concentrators-----	3-52
3.2.1.7	Resin Fire and/or Explosion-----	3-53
3.2.1.8	Facility Fire-----	3-58
3.2.1.9	Radiological Public Accident Risk-----	3-62
3.2.2	Public Atmospheric Risk of FRP-WTF-----	3-66
3.2.3	Occupational Radiological Risk of FRP-WTF-----	3-66
3.2.4	Nonradiological Risks of the FRP-WTF-----	3-66
3.3	Risk of the FFP-----	3-68
3.3.1	Reference Plant Probabilities and Consequences---	3-68
3.3.1.1	Overview and Preliminary Hazards Analysis-----	3-68
3.3.1.2	Earthquake in Excess of Design Basis----	3-73
3.3.1.3	Air Crash Into Powder Storage-----	3-75
3.3.1.4	Hydrogen Explosion in the Reduction- Oxidation-Reduction Furnace-----	3-75
3.3.1.5	Hydrogen Explosion in the Sintering Furnace-----	3-76
3.3.1.6	Hydrogen Explosion in Wet Scrap Recovery-----	3-76
3.3.1.7	Criticality Accident in Wet Scrap Recovery-----	3-76
3.3.1.8	Powder Shipping Container Spill-----	3-77
3.3.1.9	Exothermic Reaction in Powder Storage---	3-80
3.3.1.10	Radiological Public Accident Risk-----	3-82
3.3.2	Relative Public Atmospheric Risk of PT and Reference FRP-----	3-82
3.3.3	Relative Occupational Radiological Risk of PT and Reference FFP-----	3-89
3.3.4	Relative Nonradiological Risks of PT and Reference FFP-----	3-90
3.4	Risk of the FFP-WTF-----	3-91
3.4.1	FFP-WTF Dominant Accidents-----	3-91
3.4.1.1	Overview and Preliminary Hazards Analysis-----	3-91

SectionPage

3.4.1.2	Earthquake in Excess of Design Basis-----	3-92
3.4.1.3	Waste Treatment Facility Criticality-----	3-92
3.4.1.4	Radiolytic Hydrogen Explosion-----	3-92
3.4.1.5	Resin Explosion in Actinide Product Concentrator-----	3-95
3.4.1.6	"Red Oil" Explosions in WTF Concentrators-----	3-95
3.4.1.7	Resin Explosion/Fire-----	3-100
3.4.1.8	Facility Fire-----	3-100
3.4.1.9	Radiological Public Accident Risk-----	3-103
3.4.2	Public Atmospheric Risk of FFP-WTF-----	3-103
3.4.3	Occupational Radiological Risk of FFP-WTF-----	3-107
3.4.4	Nonradiological Risks of the FFP-WTF-----	3-107
3.5	Risks of Transportation-----	3-109
3.5.1	Risk of the Reference Cycle Spent Fuel Transportation-----	3-109
3.5.1.1	Overview and Preliminary Hazards Analysis-----	3-109
3.5.1.2	Reference Spent Fuel Transportation Category 1: Loss of Neutron Shielding-----	3-113
3.5.1.3	Reference Spent Fuel Transportation Category 2: Exposure of the Inner Spent Fuel Containing Cavity-----	3-115
3.5.1.4	Reference Spent Fuel Transportation Category 3: Exposure of the Inner Fuel Containing Cavity and 50% Fuel Damage-----	3-116
3.5.1.5	Reference Spent Fuel Transportation Category 4: Exposure of Severe Fuel With Severe Damage and Fire-----	3-118
3.5.1.6	Reference Spent Fuel Transportation Risk Summary-----	3-119
3.5.2	Risks of PT Spent Fuel Transportation-----	3-120
3.5.2.1	Overview and Preliminary Hazards Analysis-----	3-120
3.5.2.2	PT Spent Fuel Transportation Category 1: Loss of Neutron Shielding-----	3-120
3.5.2.3	PT Spent Fuel Transportation Category 2: Exposure of the Inner Spent Fuel Containing Cavity-----	3-124
3.5.2.4	PT Spent Fuel Transportation Category 3: Exposure of the Inner Fuel Containing Cavity and 50% Fuel Damage-----	3-124
3.5.2.5	PT Spent Fuel Transportation Category 4: Exposure of Fuel With Severe Damage and Fire-----	3-124
3.5.2.6	PT Spent Fuel Transportation Risk Summary-----	3-125

SectionPage

3.5.3	Risk of Powder Transportation-----	3-127
3.5.3.1	Risk of Reference Powder Transportation-----	3-127
3.5.3.2	Risk of PT Powder Transportation-----	3-128
3.5.3.3	Summary of Powder Transport Risk-----	3-128
3.5.4	Risk of Fresh Fuel Transportation-----	3-128
3.5.4.1	Risk of Reference Fresh Fuel Transportation-----	3-128
3.5.4.2	Risk of PT Fresh Fuel Transportation---	3-131
3.5.4.3	Fresh Fuel Transportation Risk Summary-	3-131
3.5.5	Transportation of Fuel Element Structural Materials and Cladding Hulls-----	3-131
3.5.5.1	Reference Cycle Clad Transport Category 1: Loss of Lead Shielding--	3-134
3.5.5.2	Reference Cycle Clad Transport Category 2: Release From the Cask Interior But No Canister Damage-----	3-134
3.5.5.3	Reference Cycle Clad Transport Category 3: Release From the Cask With Extensive Canister and Hull Damage-----	3-134
3.5.5.4	Reference Cycle Clad Transport Category 4: Release From the Cask, Extensive Canister and Hull Damage and Fire-----	3-134
3.5.5.5	PT Cycle Clad Transportation Risk-----	3-136
3.5.5.6	Summary of Fuel Element Structural Material and Cladding Hulls Transport-----	3-136
3.5.6	Risk of Solidified High Level Waste Transport---	3-136
3.5.6.1	Risk of Solidified High Level Reference Waste Transport-----	3-136
3.5.6.2	Risk of Solidified High Level PT Waste Transport-----	3-139
3.5.6.3	Summary of Risks of High Level Waste Transport-----	3-139
3.5.7	Risks of Transportation Non-High Level Waste by Truck-----	3-139
3.5.7.1	Non-High Level Waste Transport: Category 1: One Drum Spillage-----	3-141
3.5.7.2	Non-High Level Waste Transport: Category 2: 21 Drum Spillage-----	3-141
3.5.7.3	Non-High Level Waste Transport: Category 3: 21 Drum Spillage and Fire-----	3-141
3.5.7.4	PT Non-High Level Waste Transport-----	3-144
3.5.7.5	Summary of Reference and PT Non-High Level Waste Transport-----	3-144

SectionPage

3.5.8	Public Routine Radiological Risks from PT and Reference Transportation-----	3-144
3.5.8.1	Cask Shipments-----	3-144
3.5.8.2	Non-High Level Waste Shipment-----	3-144
3.5.9	Occupational Radiological Risk of PT and Reference Cycle Transportation-----	3-147
3.5.9.1	Cask Shipments-----	3-147
3.5.9.2	Non-High Level Waste Shipments-----	3-147
3.5.10	Nonradiological Risk of PT and Reference Cycle Transportation-----	3-147
3.5.10.1	Cask Shipments-----	3-147
3.5.10.2	Non-High Level Waste Shipments-----	3-147
3.5.11	Summary Risk of Transportation-----	3-147
4.	SUMMARIZED RISKS OF THE REFERENCE AND PT FUEL CYCLE-----	4-1
4.1	Public Radiological Accident Risk-----	4-1
4.2	Public Radiological Routine Risk-----	4-2
4.3	Occupational Radiological Risk-----	4-3
4.4	Nonradiological Risk-----	4-3
5.	CONCLUSIONS AND RECOMMENDATIONS-----	5-1
5.1	Conclusions-----	5-1
5.2	Recommendations-----	5-2
6.	REFERENCES-----	6-1
APPENDIX A	PLANT AND CASK DESCRIPTIONS-----	A-1
A.1	Fuel Reprocessing Plant-----	A-1
A.1.1	Overview-----	A-1
A.1.2	Conceptual Modifications-----	A-3
A.1.2.1	Cask Receiving Area-----	A-3
A.1.2.2	Remote Process Cell-----	A-3
A.1.2.3	High Level Cell (HLC)-----	A-7
A.1.2.4	High Intermediate Level Cell (HILC)---	A-7
A.1.2.5	Fuel Product Cell (FPC)-----	A-7
A.1.3	Product Thermal Denitrator-----	A-9
A.1.4	Fluorocarbon Krypton Recovery System-----	A-9
A.2	Fuel Reprocessing Plant-Waste Treatment Facility (FRP-WTF)-----	A-12
A.2.1	General WTF Description-----	A-12
A.2.2	Waste Treatment Facility Operations-----	A-12
A.2.2.1	Solid Alpha Waste Treatment-----	A-17
A.2.2.2	Salt Waste Treatment-----	A-18
A.2.2.3	Solvent Extraction Treatment of Parent Facility Aqueous Raffinate---	A-21

<u>Section</u>	<u>Page</u>
A.2.2.4 Cation Exchange Chromatography (CEC)---	A-22
A.2.2.5 Acid/Water Recycle-----	A-22
A.2.2.6 Processes Specific to the FRP-WTF-----	A-24
A.3 Fuel Fabrication Plant-----	A-27
A.4 Fuels Fabrication Plant-Waste Treatment Facility (FFP-WTF)-----	A-40
A.4.1 Functional Description-----	A-40
A.4.2 FFP-WTF Unit Operations-----	A-40
A.4.3 Design Philosophy-----	A-40
A.5 Spent Fuel Shipping Casks-----	A-43
A.5.1 Reference Fuel Shipping Cask-----	A-43
A.5.2 PT Fuel Shipping Cask-----	A-43
A.6 High Efficiency Particulate Filters-----	A-47
A.6.1 Filter Efficiency-----	A-47
A.6.2 HEPA Filter Reliability-----	A-48
APPENDIX B FAULT TREES-----	B-1
APPENDIX C CALCULATION OF RADIOLOGICAL DOSE AND HEALTH EFFECTS---	C-1
C.1 Input Data Preparation for Accident Analysis-----	C-1
C.2 Input Data Preparation for Routine Analysis-----	C-12
APPENDIX D OCCUPATIONAL RISK CALCULATIONS-----	D-1
D.1 Introduction-----	D-1
D.2 Calculation of Routine Exposure-----	D-1
D.2.1 Background Information-----	D-1
D.2.2 Estimated Routine Occupational Exposure-FRP----	D-2
D.2.3 Estimated Routine Occupational Exposure-FFP----	D-3
D.2.4 Estimated Routine Occupational Exposure-WTFs---	D-6
D.3 Occupational Exposure Associated With Plant Maintenance-----	D-6
D.3.1 Background Information-----	D-6
D.3.2 Occupational Exposure for Maintenance-FRP-----	D-6
D.3.3 Occupational Exposure for Maintenance-FFP-----	D-8
D.3.4 Occupational Exposure for Maintenance-WTFs-----	D-8
D.4 Abnormal Occurrences-----	D-8
D.5 Occupational Exposure Associated With Plant Accidents-----	D-9
D.5.1 Summary Results-----	D-9
D.5.2 Inadvertent Exposure of Operating Personnel During Accidents-----	D-9
D.5.2.1 Exposure Due to Shielding Failure-----	D-14
D.5.2.2 Exposure Due to Airborne Activity-----	D-14
D.5.2.3 Exposure Due to Spread of Contamination-----	D-15

<u>Section</u>	<u>Page</u>
D.5.3.4 Planned Occupational Exposure During Repair and Recovery Activities After an Accident-----	D-15
D.6 Occupational Dose Due to Facility Decommissioning-----	D-16
APPENDIX E NONRADIOLOGICAL RISKS-----	E-1
E.1 Introduction-----	E-1
E.2 Chemical Effluents-----	E-1
E.2.1 Basis for Assessment-----	E-1
E.2.2 Reference and PT FRP Chemical Releases-----	E-1
E.2.2.1 FRP Gaseous Releases-----	E-4
E.2.2.2 FRP Liquid Effluents-----	E-5
E.2.2.3 FRP Solid Nonradioactive Waste-----	E-5
E.2.3 WTF Chemical Releases-----	E-5
E.2.3.1 WTF Gaseous Effluents-----	E-5
E.2.3.2 WTF Liquid Effluents-----	E-7
E.2.4 FFP Chemical Effluents-----	E-8
E.2.5 Summary of Chemical-Caused Health Effects-----	E-8
E.3 Industrial Hazards and Accidents-----	E-10
E.3.1 Basis for Assessment-----	E-10
E.3.2 Construction of Fuel Cycle Facilities-----	E-11
E.3.3 Plant Operations-----	E-13
E.3.4 Decommissioning of Fuel Cycle Facilities-----	E-14
E.4 Conventional Transportation Accidents-----	E-15
APPENDIX F PROBABILITY OF CASK PENETRATION-----	F-1
APPENDIX H SENSITIVITY AND ERROR ANALYSIS-----	H-1
APPENDIX I GLOSSARY OF KEY TERMS AND ACRONYMS-----	I-1

ABSTRACT

The Chemical Technology Division of the Oak Ridge National Laboratory has prepared a set of documents that evaluate a Partitioning-Transmutation (PT) fuel cycle relative to a Reference cycle employing conventional fuel-material recovery methods. The PT cycle uses enhanced recovery methods so that most of the long-lived actinides are recycled to nuclear power plants and transmuted to shorter-lived materials, thereby reducing the waste toxicity.

This report compares the two fuel cycles on the basis of the short-term radiological and nonradiological risks they present to the public and to workers. The accidental radiological risk to the public is analyzed by estimating the probabilities of sets of accidents; the consequences are calculated using the CRAC code appropriately modified for the material composition. Routine radiological risks to the public are estimated from the calculated release amounts; the effects are calculated using the CRAC code. Radiological occupational risks are determined from prior experience, projected standards, and estimates of accident risk. Nonradiological risks are calculated from the number of personnel involved, historical experience, and epidemiological studies.

The result of this analysis is that the short-term risk of PT is 2.9 times greater than that of the Reference cycle, primarily due to the larger amount of industry. This conclusion is strongly dominated by the nonradiological risk, which is about 150 times greater than the radiological risk. If the radiological risk is considered alone, the ratio of PT to Reference risk is 3 composed as follows: radiological operations affecting the public - 5, radiological operations affecting the workers - 1.7, and radiological accidents affecting the public - 1.4, all in the order of decreasing risk.

The absolute risk as estimated for the fuel cycle portions considered in this report is 0.91 fatalities/GWe-year for the PT cycle and 0.34 fatalities/GWe-year for the Reference cycle. This should be compared with Inhaber's estimate of 1.5 for nuclear and 150 for coal.¹ All of the risks assumed here are associated with the production of one billion watts of electricity (GWe) per year. The present results, which encompass only a portion of a fuel cycle, are slightly higher than Inhaber, possibly as a result of using different data in estimating the health effects of non-radiological pollutants.

ACKNOWLEDGEMENTS

The authors and contributors to this report wish to acknowledge the support and complete cooperation of Drs. J. O. Blomeke, A. G. Croff, B. Finney, and W. Tedder of the Chemical Technology Division of the Oak Ridge National Laboratory. Valuable assistance was also provided by D. Davis, A. Smith, and others at the Ralph M. Parsons Company.

Contributing SAI Staff members include: J. E. Hammelman, S. Donnelson, J. Maly, W. Tan, A. Unione, R. La Riviere, N. Bhatnagar, A. Sugarman, M. Cohn, A. Hansen, H. Cothran, J. Walls, W. Rhyne, and S. Adams.

Peer Reviewers of this document and the appendices are: R. C. Erdmann (SAI), M. Szulinski (SAI), R. F. Williams (EPRI), A. Carson (EPRI), and C. Claiborne (ORNL). We are grateful for their constructive review.

1. RESULTS

The results are that the PT cycle imposes additional risk compared with the Reference cycle. Whether this is compensated by benefits from a risk reduction due to storing PT waste compared with Reference wastes, is outside the scope of this work. This result is primarily due to the non-radiological risk which dominates by a factor of 150 over the radiological risks. If radiological risks alone are considered, the risk of PT is about 3 times that of the Reference cycle. This result is primarily due to the larger work force rather than isotopic differences in the cycles. Table 1.1 summarizes these results.

Table 1.1
SUMMARIZED COMPARISON OF THE FUEL CYCLES
(Fatalities/GWe capacity-year)

Risk Types	Reference Cycle	PT Cycle	Ratio $\left(\frac{\text{PT}}{\text{Reference}}\right)$
Radiological Accident - Public	5×10^{-7}	7×10^{-7}	1.4
Radiological Routine - Public	8×10^{-4}	4×10^{-3}	5
Radiological Occupational	1.2×10^{-3}	2×10^{-3}	1.7
Radiological Subtotal	2×10^{-3}	6×10^{-3}	3
Non Radiological	0.34	0.91	2.9
Risk Total	0.34	0.91	2.9

Table 1.2 presents a comparison by operations. More detailed comparisons are presented in Sections 4.0 and 5.0.

Table 1.2
SUMMARIZED COMPARISON OF RISKS*
(fatalities/GWe capacity-year)

Operation	RADIOLOGICAL RISK		NON-RADIOLOGICAL RISK	
	Reference Cycle	PT Cycle	Reference Cycle	PT Cycle
FRP	8×10^{-4}	1.9×10^{-3}	0.24	0.24
FRP-WTF	--	1.6×10^{-3}	--	0.4
FFP	6×10^{-4}	6×10^{-4}	1.1×10^{-2}	1.1×10^{-2}
FFP-WTF	--	8×10^{-4}	--	0.16
Transportation	8×10^{-5}	8.5×10^{-5}	9×10^{-2}	0.12

* The glossary in Appendix I explains unusual abbreviations.

2. OVERVIEW AND QUALIFICATIONS

2.1 OBJECTIVES OF THIS WORK

Partitioning and transmutation (PT), when taken together, form a waste management option for reducing the amounts of certain long-lived actinide isotopes normally present in radioactive wastes and converting them to shorter-lived isotopes. Partitioning refers to special chemical procedures used to remove a large fraction of the actinides from fuel cycle facility wastes. Transmutation refers to the means by which materials are processed -- in a high neutron flux, such as occurs in a nuclear power plant -- to cause the elements to be transformed to lighter and more rapidly decaying species.

Implementation of a PT nuclear fuel cycle would result in a reduction in the amount of long-lived radioactive material that goes to the waste repository. This long-term risk reduction is countered by an increase in near-term risk resulting from increased process complexities, higher actinide specific activities, and increased actinide inventories.

The objective of this work is to compare the risk of the PT cycle with a Reference fuel cycle. In order to do this, it is necessary to determine:

1. The most significant nuclides contributing to the routine and accident risks
2. The expected routine dose rate and the probabilistic accidental dose rate from each of the fuel cycle operations (spent fuel transportation, spent fuel reprocessing, fresh fuel fabrication, fresh fuel transportation, and waste transportation) per year of operation
3. The expression of the routine and accident risk in terms of health effects
4. The expected annual health effects from routine chemical effluent releases
5. The parameters to which the results are sensitive, uncertainties in the analysis, and the effects of the uncertainties on the final results.

2.2 COMPARISON OF PT AND REFERENCE LWR FUEL CYCLES

The Reference nuclear power fuel cycle begins with the mining of the uranium. It continues through the steps of milling, conversion, enrichment, uranium fuel fabrication, consumption (in a nuclear power plant), reprocessing, mixed-oxide fuel fabrication, and waste disposal with transportation interconnecting the geographically dispersed plants. A PT fuel cycle is similar in outline except that special actinide waste treatment facilities must be added to the reprocessing and mixed-oxide fuel fabrication plants to greatly reduce the quantity of actinides reporting to the wastes. These actinides build up in the PT fuel cycle to a higher equilibrium inventory than found in the Reference fuel cycle. Certain actinide isotopes are highly radioactive, some emit substantial amounts of decay heat, and others, by virtue of spontaneous fissioning and alpha emission, require substantial neutron shielding. The PT fuel poses additional requirements in the handling of the fresh fuel at the power plant, and compared with the Reference fuel, it could pose an increased risk in case of a nuclear power plant accident.

This study simplifies the cycle somewhat by not investigating the differences between the Reference and PT cycles that occur before and including the power plant. The work reported here does not address the differences in the waste repository risks nor facility cost differences for the two cases inasmuch as these are addressed in companion reports.* It is believed, however, that the major public and occupational risk differences are encompassed in the reprocessing, fuel fabrication, and transportation steps considered here. Figure 2.1 diagrams the steps in the Reference cycle and Figure 2.2 shows the steps in the PT cycle that are being considered. Table 2.1 presents the annual material flows and quantities for the two cycles.

Both the Reference and PT facilities are assumed to be colocated at the site of the proposed EXXON reprocessing plant (Docket 50-564), which is also near the proposed Clinch River Breeder Reactor site. Both of these locations have been extensively studied and data are readily available in the public domain.

Risk analysis predictions increase in accuracy with the specificity of the model being analyzed. To this end, portions of the Allied General Nuclear Services plant, located at Barnwell, South Carolina, are used in the model of the reprocessing plant and portions of the Westinghouse MOX fabrication plant

*Reference 1, the summary of this report, provides references to the companion reports.

Figure 2.1. Reference Fuel Cycle (MOX Recycle - LWR)

Figure 2.2. Conceptual Actinide Partitioning and Transmutation Cycle

Table 2.1

MASS FLOWS AND QUANTITY COMPARISON BETWEEN REFERENCE AND PT CYCLES

Material	Units ^{a,b,c}	Amount	
		Reference Case	PT Case
<u>Fuel Materials</u>			
Fresh and spent MOX fuel	tonnes HM/year	660	660
Fresh and spent UO ₂ fuel	tonnes HM/year	1,340	1,340
Recycled actinide oxide powder	tonnes HM/year	~660	~660
Excess uranium	tonnes HM/year	~1,000	~1,000
<hr/>			
<u>Consolidated Wastes</u>			
High-level solidified waste	m ³ /tonne HM	0.063	0.063
	m ³ /year	126	126
	m ³ accumulated	11,781	11,781
Nonimmobilized TRU wastes	m ³ /tonne HM	0.977	0.977
	m ³ /year	1,954	1,954
	m ³ accumulated	182,700	182,700
Fuel residue	m ³ /tonne HM	0.212	0.212
	m ³ /year	424	424
	m ³ accumulated	39,644	39,644
Iodine waste	m ³ /tonne HM	0.0033	0.0033
	m ³ /year	6.6	6.6
	m ³ accumulated	617	617
¹⁴ C waste	m ³ /tonne HM	0.0017	0.0017
	m ³ /year	3.4	3.4
	m ³ accumulated	318	318
Concreted wastes	m ³ /tonne HM	0.827	1.20
	m ³ /year	1,674	2,435
	m ³ accumulated	156,500	227,600

^aHM = heavy metal (U + Np + Pu + Am + Cm + Bk + Cf).

^bUnits of "tonnes HM/year" and "m³/year" are based on 2,000 tonnes HM/year reprocessing plant, 660 tonnes HM/year MOX fuel fabrication plant, and 225 GW(th) [=75 GW(e)] of reactor capacity.

^cUnits of "m³ accumulated" is based on waste accumulation in a repository for the wastes from the reprocessing of 187,000 tonnes HM.

proposed to be located at Anderson, South Carolina (Docket 70-1432), are used in the model of the fuels fabrication plant. It should be noted that both plants have undergone extensive conceptual changes for the purpose of this study. In the reprocessing plant the major changes are:

1. Two additional cask receiving and unloading areas
2. A codecontamination and coprocessing flowsheet
3. A krypton recovery system
4. Thermal conversion of nitrate, to oxide powder product (co-denitration)
5. A waste solidification facility
6. A carbon-14 recovery system.

With these changes, the model plant is assumed to be suitable for processing either Reference or PT spent fuel. The fuel fabrication plant is modified from the Anderson Plant by the conceptual addition of:

1. Remote processing, fabrication, and handling facilities
2. A wet scrap recovery system
3. Remote maintenance capabilities.

These changes are applicable to fabricating Reference or PT fuel. Of course, the Reference cycle omits the two actinide waste treatment facilities characteristic of the PT cycle.

2.3 COST-RISK-BENEFIT CONCEPT

The partitioning and transmutation of actinides is aimed at lessening the inventory of long-lived radionuclides in the repository at which the waste is eventually disposed. To accomplish this end, significant changes must be made in the nuclear fuel cycle operations including additional facilities and increased and differing transportation requirements. In order to rationally choose to implement or not implement these changes it is essential that the present and future costs, risks, and benefits be quantified to permit comparison between the two options.

One of the most elementary economic concepts is for the income (benefit) of any endeavor to exceed the cost. When this occurs, a positive profit results and the cost-benefit ratio is less than one. These benefits, or costs, may accrue continuously or randomly.

Costs that occur continuously are calculated by simply adding up all expenses: labor, supplies, taxes, and insurance. Similarly, continuous health effects are determined by adding up historically-based exposure data or by calculating exposures in performing certain tasks and adding up the frequency and personnel-weighted tasks. It should be noted that costs or benefits need not necessarily be measured in dollar units but may be in terms of property, health effects,* radiation dose, or other units. It is essential, however, that comparisons be made in terms of common units.

Not all effects are continuous; some occur as a sudden release or lump-sum expenditure with possibly a very small average occurrence frequency. It is shown below that the effects are additive and hence may be superimposed. The aggregate effect may be calculated as the product of the frequency times the magnitude of the effect. This product is called risk and is applied to both monetary and health situations.

To demonstrate this interpretation of risk, imagine a world with a very large nuclear power plant population, N . All plants are identical and each plant has identical demography and meteorology. Each plant is so separated from the other that one does not affect the other's demography and vice-versa. Each year, n_i plants fail in the i -th failure mode, causing a total population dose, d_i .

Linear superposition says that the population dose (other risk measures could be used) is linearly proportional to the number failing.

$$d_i = c_i n_i \quad (1)$$

Hence, c_i is the population dose due to a single plant.

*Some risk-benefit authors attempt to place a monetary value on health effects. Here comparisons are made between like objects with no attempt being made to cross-relate units.

The total population dose, D , is the sum of the population doses resulting from all failure modes, M .

$$D = \sum_{i=1}^M d_i = \sum_{i=1}^M c_i n_i \quad (2)$$

The probability of occurrence of the i -th failure mode is

$$p_i = \lim_{N \rightarrow \infty} \frac{n_i}{N} . \quad (3)$$

The population dose per year per plant and hence the risk in its actuarial meaning is

$$R = \frac{D}{N} = \sum_{i=1}^M c_i p_i . \quad (4)$$

This may be viewed as the weighted mean of the consequences, called the expectation value of the consequences.

Much has been said about accident probabilities and consequences but nothing of the mechanics of constructing a probabilistic risk analysis. Figure 2.3.1 shows, schematically, the steps in the process. The work begins with an understanding of the plant (or transportation mode). Because of the "defense in depth" utilized in all aspects of nuclear power, it is to be expected that any accident significantly affecting the public will be complex and requiring multiple active or passive barrier failures.

Using a Preliminary Hazards Analysis or similar procedure, a table is prepared that correlates initiating events (1) with the failure of plant protection and packaging barriers. The source of information regarding the initiating events may be from a data base (1A), from the literature, or from an understanding of the processes and possible dangers. The set of circumstances that make an accident possible are called the event sequences and may be diagrammed as a tree called an event tree. Each step in an event sequence is analyzed (4) by fault tree analysis, using component data obtained from a data base (4A) that indicates the observed failure rate of these components.

Figure 2.3.1. Steps in Constructing a Probabilistic Risk Analysis

The reason that system failure rates must be evaluated from fault trees and not from historical data is the absence of failure experience with these systems. This absence of system failures is not surprising. Although the components comprising the system are conventional with known failure rates from use in other applications and environments, the systems themselves are designed for very low failure rates by the use of redundancy.

In parallel with the evaluation of accident sequence probabilities, the amount, isotopic composition, and physical and chemical form of the released material is calculated using physical-chemical methods (3). This makes use of a data base of measurements (3A). The consequence-probabilities are synthesized as a release model (5), and the ex-plant consequences are calculated from a knowledge of the site (6A). From this the integrated risk (7), and the relationship of this to the fuel cycle, is obtained. These calculations may be iterated to analyze alternative fuel cycles, to examine data or assumption sensitivities, and to calculate the estimated error bounds on the results.

These are the steps employed in preparing this report. The next section discusses the fuel cycle composition so that the risk can be related to a gigawatt of electrical energy, the ultimate benefit of the nuclear fuel cycle. The section following this describes the parameters and assumptions used in calculating the atmospheric dispersion of the radioactivity. This is followed by major sections on each of the fuel cycle steps.

2.4 RELATIONSHIP OF PRESENT WORK TO OVERALL ANALYSIS

The Chemical Technology Division of Oak Ridge National Laboratory is evaluating the incentives for implementing a partitioning-transmutation nuclear fuel cycle. This program consists of two major phases. The first phase investigated the experimental, calculational, and conceptual studies concerning various specialized aspects of PT. This involved experimental studies for reducing the actinide content of the wastes, calculational studies of the transmutation of the actinides, the impact of PT on nuclear fuel cycle operations, and studies concerning the integration of partitioning into reprocessing and refabrication plant flowsheets.

The second phase involves evaluating the incentives for commercial implementation of PT. The principal tasks are:

1. Determine the total costs of implementing PT for the commercial nuclear fuel cycle.
2. Determine the short-term risks imposed by the additional handling of increased amounts of transuranic elements.
3. Determine the change in the long-term risks from a geologic radioactive waste repository with reduced actinide content.

The work described in this report is in support of the second task, the short-term risk of PT as compared with a Reference cycle.

2.5 GENERAL ANALYTICAL APPROACH

The general analytical approach begins with assumptions regarding the Reference cycle. This is assumed to be a 75 GWe (@ 100% capacity factor) nuclear electric power industry.* The reprocessing and fabrication plant capacities and the transportation are sized to support this amount of power production.

The next step is the description of the Reference facilities. Because of the lack of detailed plant designs for coprocessing and remote production MOX fuel fabrication, it is assumed that the plants for the Reference cycle are identical to the PT plants. That is, the Reference and PT plant only differ by the addition of the Waste Treatment Facility (WTF) and the respective PT reprocessing and fuel fabrication plants. Regarding transportation, it is assumed that the Reference cycle moves fuel in NLI 10/24 casks by rail and the PT cycle moves MOX fuel in the ORNL conceptual casks, also by rail. This includes 1) MOX powder movement between the colocated reprocessing and fuel fabrication plants and 2) fresh as well as spent fuel. While this may be overly conservative regarding fresh fuel and powder for both cycles, it is believed that it is not possible to move fresh fifth recycle fuel for either the Reference or PT cycles in unshielded casks. Lacking designs for shielded fresh fuel and powder casks,

*It should be pointed out that this work expresses some results in units of per gigawatt electric if the plants were operating at full capacity. This procedure reduces the sensitivity to the results to assumed size of the industry. Nevertheless, it is important to choose a reasonable approximate size because all effects may not scale linearly with the amount of electric power generated.

the above conservative approach is adopted. The other aspects of transportation, such as the transportation of TRU, cladding hulls, and high-level wastes to the repository, are all assumed to be the same for both cycles.

With this background, accident risks are calculated by preparing Preliminary Hazard Analyses (PHA) for all of the plants. These analyses are a tabular listing of accidents that could possibly happen along with barriers* and mitigating barriers to the occurrence. Working with the PHAs, certain accidents are selected on the basis of their preliminary estimated contribution to the overall risk. These selected accidents are subjected to detailed fault tree analysis to refine the estimated accident probability. The magnitude of the release is estimated using experimental aerosol data and dispersion modeling. The amounts of material leaving the plant are estimated using experimental data for filtration efficiencies, and the effects of the radioactivity on the public is estimated using Gaussian plume modeling in the form of the computer code CRAC (Consequences of Reactor Accident Code), which calculates health effects.

Routine public radiological exposures are estimated from calculated process information and off-gas and filtration depletion factors. Occupational exposures are estimated from past plant experience and ALARA goals. The results for the Reference and PT fuel cycles differ because of radioactive inventory differences, in numbers of personnel and in the WTFs.

The chemical risks are calculated similarly to the radiological risks, with the primary difference being in the calculation of the effect of the chemical species on the health of exposed population.

These results are assembled to provide a measure of the public and occupational risks posed by each fuel cycle and especially the relative risk of one cycle as compared to the other. Comparisons are made in terms of risk as the product of frequency and consequences.

*The term barrier is used in a generalized sense of a constraint which may be a physical wall or may be such things as administrative controls or electronic or electrical interlocks.

2.6 QUALIFYING REMARKS

2.6.1 Fuel Cycle

This study, in assessing the differential risks between Reference and PT fuel cycles, does not address uranium mining and milling, reactor plant construction and operation, or waste repository construction and operation. It is assumed that any incremental change in risk between the two fuel cycles would be small compared to the incremental increase presented by the addition of the two WTFs.

2.6.2 Facilities of Interest

In analyzing the facilities of interest (FRP, FFP, and WTFs), it is to be recognized that reliance had to be placed on analogies and similarities between the postulated production plants and existing ones. The proposed flowsheets and required unit operations differ to varying degrees from demonstrated flowsheets and currently applied technology. Further, the facilities analyzed were available only as preconceptual designs. Precise facility layouts, unit operations, and process control systems frequently had to be assumed during the course of the analyses. Such assumptions are detailed in subsequent text. A final factor affecting the risk analysis is the point in time when the analysis applies. The proposed facilities are scoped for the fifth recycle. Implementation of either the Reference MOX case or the PT MOX case would result in a minimum of two generations of facilities to arrive at the fifth recycle. This elapsed time would alter the political and regulatory climate under which the facilities would be designed, constructed, and operated. Furthermore, initial implementation of the Reference and PT technologies for the relatively mild first recycle (mild in terms of radioisotopic inventories) would enable future plant designers to upgrade facilities and processes in light of actual plant experience. In this risk assessment, the general assumption is made that existing technology is applied in the present political and regulatory climate. One important exception is the use of 500 mrem/yr as shielding design criterion (in contrast to the present 1 rem/yr requirement). This value was chosen based on anticipated reductions in radiation exposure limits..

2.6.3 Accidents Not Analyzed

During the course of selecting and ranking accidents for more rigorous analysis, a number of accidents analyzed for other facilities in different contexts (PSARs, SAR, etc.) were examined and rejected as significant risk contributors. Table 2.2 lists some of the more common accidents, indicates applicable facilities, and gives reasons for eliminating them from detailed analysis.

One final qualification is necessary regarding the basis for risk comparison. During the risk analysis, two categories of risk were developed: one being expressed as person-rem, and the other as injury or fatality. The first, of course, came from the various radiation exposure mechanisms. The second is derived from conventional accident statistics and exposure of humans to toxic/noxious materials. In order for these two categories to be put on an equivalent and, hence, comparable basis, it was necessary to make a quantitative conversion. The basis for the conversion of radiological dose to fatalities was ERDA-77-75⁽²⁾ that assumes 50% of the latent cancers eventually lead to fatalities. This factor was applied to CRAC calculations to obtain fatality predictions.

Some of the analysis presented here does not use CRAC calculations but predicts effects in terms of person-rem which is converted to fatality predictions using a value of 1×10^{-4} fatalities/person-rem obtained as an average of the BEIR data⁽³⁾. It should be noted that this implies the assumption of a linear relationship between dose and health effects. There are physiological reasons for believing this assumption to be conservative.

Table 2.2
POSSIBLE MAJOR ACCIDENTS NOT ANALYZED

<u>Accident</u>	<u>Applicable Facility</u>	<u>Remarks</u>
Incinerator fire/explosion	Site incinerator, WTFs	The effects are generally confined within cell or hood by virtue of design provisions for such occurrences
Radioactive releases from Transfer Tunnel	FRP -- FRP-WTF Tunnel FFP -- FFP-WTF Tunnel	The tunnel is underground, protected by 7' thick concrete roof and is vented to the WTF. No credible release pathway was apparent.
Volatile FP or FP Gas release from offgas treatment equipment (Kr-85 recovery, Iodex scrubber, ruthenium absorber)	FRP, WTFs	Unit operation vessel inventories are small. Release mechanisms not apparent. Bounding case of Kr bottle rupture analyzed.
Zr Fines Fire	FRP, FRP-WTF	This accident has been analyzed elsewhere and found to be a very minor risk contributor (2).
Natural Forces in Excess of Design Basis Accident	All	Non design basis accidents include accidents which exceed site criteria (seismic, tornadic, floods and nearby man made hazards), or require independent failure of process and multiple containment barriers. These are not analyzed based on the rationale given in DOE/ET-0028 (Technology for Commercial Radioactive Waste Management, Vol. I p. 3.7.3, May 1979). The accidents are judged to be extremely unlikely during the design life of the plant, by virtue of present plant design and operational techniques.

3. RISK ANALYSIS*

3.1 RISK OF THE FUELS REPROCESSING PLANT (FRP)

3.1.1 Accident Probabilities and Consequences

3.1.1.1 Overview and Preliminary Hazards Analysis

Appendix A.1 contains an overview discussion of the plant and the material quantities and flows. This plant is designed for reprocessing 5th recycle Reference or PT fuel having a burnup of 33,000 MWD/T after a cooling period in excess of 1.5 years with an annual capacity of 2,000 MTHM of MOX and UO₂ fuel. This plant differs significantly from present or proposed plants in the following regards: (1) coprocessing (fissile material is not separated from fertile material), (2) krypton and carbon-14 recovery from off-gas, and (3) product thermal denitration. This study assumes that shielding and maintenance methods are consistent with ALARA. It also assumes the shielding, criticality margins, and other process items are sized and designed such that either Reference or PT fuel may be processed; however, processing in the PT mode requires the addition of the FRP-WTF discussed in Appendix A.2.

The FRP is somewhat larger than the Allied General Nuclear Services plant and there are significant process differences. Nevertheless, a convenient point of departure for this risk analysis is the work of Cooperstein, et al.⁽⁴⁾ who used the AGNS plant and that of Erdmann, et al.⁽⁵⁾ who used the EXXON plant proposed to be located near Oak Ridge, Tennessee.

Table 3.1.1 presents a preliminary hazards analysis (PHA), the thrust of which is accident completeness. Through preliminary scoping accident calculations, the accidents presenting the greatest public risk are selected for detailed fault tree analysis and accident consequence evaluations.

From scoping calculations, the following accidents (Table 3.1.2) are selected as the plant accidents that dominate the risk envelope.

*Unusual abbreviations are explained in the Glossary (Appendix I).

Table 3.1.1

FRP PRELIMINARY HAZARD ANALYSIS

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Fuel Receiving	Cask Drop Into Unloading Pool	Crane Failure	Water Slows Rate of Fall Aluminum Crush to Absorb Shock	Noble Gases Iodine Semi-Volatiles	Low probability but large consequences. Emergency action would be effective.
	Earthquake	Natural	Design	"	"
	Tornado	Natural	Design	"	"
	Fuel Element Drop	Mechanical Failure	Cell Confinement Double HEPAs	Noble Gases	Small
	Overheated Cask and Fuel	Human Error	Inspection and Testing of Cask Before Opening		Small public but could result in occupational exposures
Fuel Dissolution	Fuel Element Drop	Mechanical Failure	Cell Confinement and Double HEPA Filters	Noble Gases and Volatile	Slight
	Hydrogen Explosion	Air Purge Failure	Design and Above	"	Could be serious.
	Air Reversal	Control Failure or Tornado	Reverse Flow Dampers	Noble Gases, Iodine, CO ₂ , Semi-Volatiles Without Filtration	Could be serious for plant personnel.
	Spill of Chopped Fuel on Cell Floor	Human Error and Control Failure	Interlocks	Nobles, Volatiles and possibly Semi-Volatiles Into Dry HEPA Filters	Slight
	Fuel Dissolution Tank Failure	Stress Corrosion Cracking	Design and Corrosion Samples	Noble Gases, Iodine, CO ₂ , Semi-Volatiles Into Dry ² HEPA Filters	Serious but very improbable to occur suddenly.
	Pipe Failure	Stress Corrosion Cracking Earthquake Poor QA	QA, Design	"	Slight
	Criticality	Lack of Neutron Poison	Procedures, Criticality Monitors	Noble Gases, Iodine	Small for the public, serious for workers.

Table 3.1.1 (cont'd)

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Co-Decontamination	Solvent Leak to Cell and Fire	Pipe Failure, Pulsar Blowback	Design	Semi-Volatiles	Low
	Explosion	Solvent Nitration	Control	"	"
		Lack of Hydrogen Purge	Design	"	Serious
	Criticality	Plutonium Complexing or Precipitation	Temperature and Acid Control	Noble Gases, Iodine	Slight for the public, serious for workers.
		Lack of Neutron Poison	Personnel Supervision		
Partitioning	Solvent Leak and Fire	Pipe Failure, Pulsar Blowback	Design	Semi-Volatiles	Serious
	Explosion	Solvent Nitration	Control	"	"
		Lack of Hydrogen Purge	Design	"	"
	Criticality	Separation of Plutonium From Process Upset	Criticality Monitor and Process Control	Noble Gases, Volatiles	Slight for the public, serious for workers.
		Plutonium Complexing or Precipitation	Temperature and Acid Control		
		Lack of Neutron Poison	Personnel Supervision		
Uranium Purification	Solvent Fire	Pipe Failure, Pulsar Blowback	Design	Uranium	Very slight
Product Denitration	Explosion	Process Upset	Design	Actinides	Serious but filtration should limit release.
	Criticality	Process Upset	Analysis, Supervision, Criticality Monitors, Neutron Poison	Noble Gases, Volatiles	Slight for public but serious for workers.

Table 3.1.1 (cont'd)

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Product Storage	Criticality	Powder Spill Into Unsafe Geometry and Moderation	Design	Noble Gases and Volatiles	Slight for public, serious for workers
	Powder Spill	Vessel Failure	Design	Actinides	Slight
Off-Gas	Krypton Cylinder Breakage	Metal Failure, Improper Handling	Low Cylinder Pressure, Pressure Regulation, Quality Control	Krypton	Serious
	Explosion in Silver Zeolite	Ammonium in system, Silver Azide	Process Control	Iodine	Serious
High-Level Waste	Explosion in Waste Calciner	Process Upset (Red Oil)	Design	Fission Products	Serious
	Loss of Cooling	Loss of Pool Water	Stainless Steel Liner, Leak Monitors, Anti- Siphon Devices, Emergency Action	Semi-Volatiles	Serious but improbable
Organic Recovery	Solvent Fire	Pipe Break Vessel Rupture	Quality Control	Actinides, Semi-Volatiles	Low

Table 3.1.2
DOMINANT ACCIDENTS IN THE FRP

<u>Accident Section</u>	<u>Accident Description</u>
3.1.1.2	Hydrogen Fire and Explosion in the HAF Tank
3.1.1.3	Solvent Fire in the HA Contactor
3.1.1.4	Red Oil Explosion in HLW Concentrator
3.1.1.5	Explosion in the HLW Calciner
3.1.1.6	Red Oil Explosion in the Fuel Product Concentrator
3.1.1.7	Explosion in the Fuel Product Denitrator
3.1.1.8	Criticality in a Process Cell
3.1.1.9	Failure of a Krypton Storage Cylinder
3.1.1.10	Hydrogen Explosion in Reductor after Co-denitration
3.1.1.11	Fuel Assembly Drop
3.1.1.12	Hydrogen Explosion in Fuel Product Denitrator Feed Tank
3.1.1.13	Accident Summary

3.1.1.2 Hydrogen Fire and Explosion in the HAF Tank

A radiolytic hydrogen explosion could occur in various parts of the FRP where considerable alpha activity occurs in aqueous solutions. In this regard, the HA feed tank is a primary location for this condition. The hydrogen, once formed, must exceed 4 vol. % concentration in the vessel void space. Upon ignition, the resultant explosion may rupture the vessel and adjacent equipment may be damaged by missiles.

Figure B-1.1 (Appendix B) presents a generalized fault tree which must be particularized for this case. The fire and explosion could result from excessive hydrogen production, insufficient air dilution, heating, the catalytic presence of heavy metal and spark source.

In applying the fault tree to this case, it is assumed that the Radionuclide Concentration ($C1 = \epsilon^*$) and Air Inbleed Less Than Limit are within design ($C1 - C2 = \epsilon$).

Water Vapor Less Than Limit is a function of tank temperature and is estimated to occur during startup or shutdown. Assuming a plant availability of 0.75, this is estimated as $C3 = 0.25$.

Air Supply 1 Failure is taken to be dominated by air compressor failure. LMEC⁽⁶⁾ gives an air compressor failure rate of 2×10^{-4} /hour or $C4 = 2$ /year.

The probability that a second compressor would be out when the first one failed assuming 12-hour repair time and a 6-hour air supply is $C5 = (12-6) \times 2 \times 10^{-4} = 1.2 \times 10^{-3}$ /demand. Hence the probability of both failing is $G8 = 2 \times 1.2 \times 10^{-3}$ /year = 2×10^{-3} /year. Lack of compressed air could also result from an electrical outage. The outage would need to last about 12 hours, however, to be dangerous. Nicholson and Fullwood⁽⁷⁾ found that the probability of an outage of this duration is about 10^{-3} per electrical outage incident. So if a single line feeds the FRP (0.5/year) with a single diesel backup (10^{-2} /demand⁽⁷⁾), the probability of losing onsite power for this time is $C6 = 5 \times 10^{-6}$ /year.

Loss of Ventilation due to plugging seems to be extremely unlikely, hence $C7 = \epsilon$.

Excessive heater temperature is very unlikely because of the use of steam heating. Pipes would rupture before this temperature was reached: $C8 = \epsilon$. Failure to dissipate radionuclide decay heat presupposes a large quantity of dried material in the cell. This is unlikely due to routine cell wash downs so $C9 = \epsilon$.

Mechanical friction heating to ignition temperature is not possible because of the absence of machinery in the cell, so $C10 = \epsilon$.

*Epsilon (ϵ) represents a small number, smaller than other probabilities in the fault tree which may be ignored without serious error.

A catalyst present that could result in a chemical reaction at sub-ignition temperature $C14 = \epsilon$.

The probability of a spark capable of igniting the hydrogen-air mixture is taken as $G7 = 0.5/D$ over this time period.

By combining these probabilities as indicated in the fault tree,

$$G1 = 0.25 \times (2 \times 10^{-3}/\text{year} + 5 \times 10^{-6}/\text{year}) \times 0.5/D = 3 \times 10^{-4}/\text{year}.*$$

Should this occur, any fire would be very short; however, the explosion could result in an aqueous aerosol of 100 mg/m^3 (This is a nominal value chosen from the data of reference 8). The High-Level Cell is $1 \times 10^3 \text{ m}^3$ and thus 0.1 kg (70% liquid) is suspended in the aerosol and the rest falls to the floor of the cell and is collected in the sump and transferred to a storage tank.

This suspended aerosol is assumed to consist of the solution and all of the iodine present in the HAF tank; however, the iodine is only 80% of the fraction present in the fuel due to volatilization and only 80% of the ruthenium is present due to insolubility during dissolution. Table 3.1.3 presents the quantities released to challenge the air filtration system. Before this aerosol can escape to the atmosphere, it must pass through the deep-bed filter and dual HEPA filters.

3.1.1.3 Solvent Fire in the HA Contactor

Within the FRP are large inventories of solvent contaminated to varying degrees with extractable radionuclide species. The greatest risk of radioactive releases would be associated with the codecontamination contactor where the largest inventory of complexed radionuclides is in the solvent. In order that a fire occur, solvent leaked from the contactor must accumulate and ignite. This accident was analyzed by Cooperstein, et al.⁽⁴⁾ as having a probability of $5 \times 10^{-6}/\text{year}$ and by Cohen and Dance⁽⁹⁾ as being in the range of $10^{-5} - 10^{-7}/\text{year}$. It is reanalyzed here in order to trace the data sources (refer to the fault tree shown in Figure B-1.2).

*Here as in subsequent evaluations the product term for an "or" gate is neglected wherever it is a small fraction of the summed terms.

Table 3.1.3

ESTIMATED CURIE RELEASES TO THE AIR FILTRATION
SYSTEM FROM AN HAF TANK EXPLOSION*

Species	Reference Cycle	PT Cycle
I	9×10^{-2}	9×10^{-2}
Ru	7.1	7.5
Zr	0.13	0.14
Nb	0.30	0.34
Y	1.9	1.9
Cs	6.1	6.6
Ce	10.1	10.4
Pm	3.2	3.1
Pu	13.2	13.5
Am	0.067	0.76
Cm	1.4	5.8
Total	43.	50.

Cells containing combustibles are equipped with fire extinguishing systems (CO_2) and with a floor sump, which has the effect of fire mitigation by drawing off most of the combustible liquid. Primary fire protection is provided by isolation of the organic from the atmosphere in the pipe and process equipment boundaries.

Solvent Available Outside of Process Equipment (G6) is estimated to occur once per year. The most likely cause is from vessel overflow. Spills due to transfer leaks in a welded system are unlikely as is vessel rupture. G6 = 1/year.

*A. G. Croff (ORNL) provided the ORIGEN runs for the material compositions in the Reference and PT cycles.

Solvent Mist Formation Due to Mechanical Causes (C1) is very small due to the absence of machinery. Spilling Onto a Heated Surface (C2) might occur 0.1 of the time, so $G7 = 0.1/D$ and $G4 = 0.1/\text{year}$.

Oxygen is always assumed present since the cells are ventilated ($C13 = 1$).

Ignition Source (G8) data are not available because "near miss" fires are not generally reported. This value is estimated as $0.1/D$ from considerations of the fairly high solvent inventory.

Solvent Temperatures above Auto-Ignition (G9) should not occur in practice because pressurized steam is supposed to be the only heat source in the cell. $G9 = \epsilon$.

Blocked or Restricted Floor Drain (C8). No data are available and it is assumed to be $C8 = 0.1/D$.

Failure of Personnel to Intervene (C14) may occur not because of a failure of personnel to respond, but because of the few actions that can be taken to prevent a fire given the solvent spill. Personnel could activate the fire prevention system ($C14 = 1$).

Failure of Solvent Spill Detection (C9). The dominate failure mode is assumed to be a liquid level sensor failure whose failure rate is given by Anyakora, et al.⁽¹⁰⁾ as 1.7/year. Assuming monthly test and maintenance, the probability of this being failed when required is $1.7/24 = 0.07/D$.

Failure of the CO_2 System (C11) is reported by Antocicco, et al.⁽¹¹⁾ as $0.17/D$ which is the value used here even though Miller^(12,13) reports a larger value.

Loss/Failure of Utility Service (C10) is taken to have no effect because of failsafe design practices; so $C10 = \epsilon$.

Combining these probabilities as shown in the fault trees gives

$$G1 = 0.1 \times 0.01 \times 1 \times (0.07 + 0.17) = 2 \times 10^{-4} / \text{year}.$$

Should this accident occur, the fire will be oxygen-limited and very smokey. Some tests show very short-time HEPA filter blockage from the soot - here this is taken as 10 minutes. The air flow rate into the cell is 4,200 cfm unless mitigation is taken which is assumed very unlikely in this short time.

Complete combustion of the solvent requires 15.2:1 ratio of air weight to solvent weight; however, it is assumed that the combustion is only 50% complete. With these assumptions, it is estimated that 280 kg of solvent will be burned. This is a smaller quantity than the amount of solvent that could be spilled from the major tanks so the actual source is not significant with regards to the quantity of solvent that could be consumed. This quantity of solvent represents 3.3% of the solvent in the cell. Using the fact that the solvent feed tank represents 2 hours of throughput for the plant, Table 3.1.4 is prepared. This table assumes that 80% of the original iodine, 99.5% of Pu and U, and 0.5% for the Am. Cm and other fission products are in the organic.

3.1.1.4 Red Oil Explosion in HLW Concentrator

By process upset, it may be possible to get organic into the waste stream. Because of the temperature and acidity, this may become nitrated and explode. A "red oil" explosion may result from a complex mixture of nitrated organics rapidly heated above the temperature range of 125-150°C and undergoing autocatalytic decomposition⁽¹⁴⁾. It is likely that such an explosion would rupture the vessel and possibly damage adjacent equipment. Such a sequence of events is depicted in the fault tree of Figure B-1.3.

Alarm ineffective (C1). This could be due to a human error of omission ($10^{-2}/D$ ⁽¹⁵⁾) or failure of the alarm system which is probably dominated by the level sensor. This latter probability is taken the same as C9, Section 3.1.1.3: $C1 = 0.07/D$.

Level Sensor Ineffective (C2). This is the same as C1 or $C2 = 0.07/D$. It is possible that there may be common mode failures between the alarm system and the pump cutoff system through the maintenance on the two level sensors. The geometric mean of the completely dependent case and completely independent cases is: $G4 = 2 \times 10^{-2}/D$.

TBP Concentration (C3). No data are available on this occurrence. In a commercial, fixed purpose plant, this is assumed to occur such that $C3 = 1/\text{year}$.

Excessive Nitric Acid Concentration. (C4). This could be due to pH meter failure. This is reported⁽¹⁰⁾ as 6 faults/year. These are located for accessibility and it is assumed that a shift change is the mean time between failures. $C4 = 6 \times 4/8760 = 3 \times 10^{-3}/D$.

Table 3.1.4

ESTIMATED CURIE RELEASES TO THE AIR FILTRATION
SYSTEM FROM A SOLVENT FIRE IN THE HA CONTACTOR

Species	Reference Cycle	PT Cycle
I	0.001	0.001
Ru	0.5	0.6
Sr	0.1	0.1
Y	0.1	0.1
Zr	0.008	0.008
Nb	0.02	0.02
Cs	0.4	0.4
Ce	0.6	0.6
Pm	0.2	0.2
Pu	150.	160.
Am	0.004	0.005
Cm	0.09	0.03
Total	152	162

Failsafe Control System Fails. (C5). This is defined as a failsafe system; to fail unsafe implies a design and checkout error; hence, $C5 = \epsilon$.

Limit Set Point Error. (C6). This is taken as an error of commission⁽¹⁵⁾; hence, $C6 = 10^{-2}/D$.

Solution Transfer Fails. (C7). This is treated not as the failure of the transfer capability but as an error of commission⁽¹⁵⁾; $C7 = 10^{-2}/D$.

Secondary Loop Overheats. (C8). Since this is a steam loop, overheating can only occur if a pressure relief valve fails closed. Balfanz⁽¹⁶⁾ lists 33 failure rates for safety valves; whose geometric mean is $C8 = 9 \times 10^{-3}/\text{year}$.

Primary Loop Overheats Secondary (C9) is always possible, so $C9 = 1$.

Solution Self-Heating (C10) is possible, so $C10 = 1$.

Solution Transfer. (C11). Same as before; $C7 = C11 = 10^{-2}/D$.

Emergency Cooling System Fails. (G14). The emergency cooling is estimated to be needed less than $10^{-1}/\text{year}$. The emergency system is very simple, using once-through cooling and the failure is estimated to a human error of omission $10^{-2}/D^{(15)}$, so $G14 = 10^{-3}/\text{year}$.

The Normal Temperature Control System (C15) is estimated to fail every 10 years, so $G15 = 10^{-1}/\text{year}$.

Denitrator Pressurized. (G7). In this design, the denitrator operates at atmospheric pressure so $G7 = \epsilon$.

Heavy Metals/Nitric Present. (C19). They should not be present but their absence only modifies the reaction temperature 10 - 20 °C; hence, $C19 = 1$.

Combining these probabilities gives an overall probability for a Red Oil Explosion as $G1 = 4 \times 10^{-6}/\text{year}$.

The consequences of this accident are estimated based on a Remote Process Cell volume of $1,900 \text{ m}^3$ forming an aerosol of 100 mg/m^3 density (reference 8. nominal value for aqueous aerosols). The discharge of the self-heating liquid is assumed to provide the mechanical forces for the generation of this stable aerosol although a considerably larger quantity of liquid remains on the floor than goes into the aerosol.

Using this model, the release that challenges the cell ventilation system is presented in Table 3.1.5.

3.1.1.5 Explosion in the HLW Calciner

An explosion may occur in the HLW calciner as a result of rapid addition of fuel or ignition of accumulated combustibles. It is assumed that the calciner burns kerosene in a manner similar to the proposed calciner for the EXXON FRP. The explosion would probably rupture the vessel and damage adjacent equipment. The calcined powder would be dispersed in the cell. The fault tree shown in Figure B-1.4 depicts the sequence of events that could result in the explosion.

Table 3.1.5

ESTIMATED CURIE RELEASES TO THE AIR FILTRATION
SYSTEM FROM A RED OIL EXPLOSION IN THE HLW CONCENTRATOR

Species	Reference Cycle	PT Cycle
I	5×10^{-5}	5×10^{-5}
Ru	75	79
Sr	20	20
Y	20	21
Zr	1.4	1.4
Nb	3.2	3.3
Cs	64	69
Ce	105	109
Pm	33	32
Pu	0.7	0.7
Am	0.7	0.8
Cm	15	58
Total	340	390

Process Upset Results in Organic in HLW, (C1). Because of the process operations preceding the calciner/vitrifier, this event is extremely unlikely C1 = ϵ .

Excessive Zirconium Fines in HLW, (C2). Should fines pass through the preceding processes, getting sufficient quantities to result in a metal powder explosion in the calcining step is very unlikely C2 = ϵ .

Steam Explosion of Glass Melt, (C3). Should the calciner fail during operation, moisture could fall onto the molten glass. This is the wrong order for an explosion (i.e., glass onto water); therefore, C3 = ϵ .

Heat Vaporizes Fuel, (C4). This normally occurs when in operation. Using the plant availability factor of 75%, $C4 = 0.75$.

Ignition Source, (C5). As in 3.1.1.2, this is taken as $C5 = 0.5$ although ignition must occur in a fairly short time interval or else a fuel will cool possible ignition surfaces.

Ignition Failure, (C6). It is assumed that a gas pilot light is the igniter and that this might fail at a rate of $C6 = 0.5/\text{year}$.

Solenoid Valve Failure, (C7). Balfanz⁽¹⁶⁾ gives a failure rate of $1 \times 10^{-2}/\text{year}$ for solenoid valves in a safety service. Since the failure mode is not specified, it is assumed that half of the failures are in an unsafe mode for this application. This valve is only challenged when the ignition fails, so $C7 = 1 \times 10^{-2} \times 0.5 \times 0.5 = 3 \times 10^{-3}/\text{yr}$.

Manual Cutoff Failure, (C8). Should the valve fail and allow unburned organic into the calciner, the operator could cut off the flow. This would require fast action and a high stress failure rate is used: $C8 = 0.1/D$.

Instrumentation Failure, (G7). It is extremely difficult to see how a fire detector could fail unsafe. Anyakora, et al.⁽¹⁰⁾ gives a failure rate of flame detectors in chemical plants (Works A) as 0.3/year. It is assumed that the probability of an unsafe failure mode is 0.1. Assuming monthly testing, $G7 = 0.03 \times 0.03/24 = 4 \times 10^{-5}/\text{year}$. It is possible there could be a common failure mode; however, the detectors should be diverse to prevent this possibility.

HLW Powder Present in Solidifier, (C11). The sequence of events leading to an explosion could only happen during startup of the solidifier because when it achieves operating temperature, the fuel will self-ignite. This plant is capable of supplying the reprocessing need of 75 GWe power generation; however, with this high throughput, the furnace would probably not be shut down between campaigns. It is, therefore, assumed that quarterly maintenance and shutdown is practiced (for accountability inventory), so $C11 = 0.25$.

Fuel Explosion (G3) is the combination of probabilities. $G3 = (0.5/\text{year}) \times (0.75) \times [(3 \times 10^{-3} \times 0.1) + 4 \times 10^{-5}] = 6.5 \times 10^{-4}/\text{year}$.

In summary, the probability of release from the HLW calciner is $G1 = 6.5 \times 10^{-4}/\text{yr} \times 0.25 = 2 \times 10^{-5}/\text{year}$.

The consequences of such an accident in a cell volume of $1,000 \text{ m}^3$ would be to produce an aerosol of 1 gm/m^3 (higher mid-range value from the data of reference 8). The material ejected from the calciner is assumed to contain 50% fission product oxides and 50% inert material; hence, 500 gm of radioactive material constitutes the aerosol. The reference case is assumed to contain 0.5% of the plutonium and the PT case is assumed to contain 0.01% of the initial actinides. With these assumptions, Table 3.1.6 is prepared.

Table 3.1.6
ESTIMATED CURIE RELEASED FROM
AN HLW CALCINER ACCIDENT

Species	Reference Cycle	PT Cycle
Ru	3,300	3,300
Sr	860	850
Zr	60	60
Nb	140	140
Cs	2,800	2,900
Ce	4,600	4,600
Pm	1,400	1,400
Pu	30	0.6
Am	31	--
Cm	660	0.3
Total	13,900	13,300

3.1.1.6 Red Oil Explosion in the Fuel Product Concentrator

This accident sequence is similar to that of 3.1.1.4 and as depicted in the fault tree of Figure B-1.3. This probability is, therefore, taken to be $G1 - 4 \times 10^{-6}/\text{year}$.

This explosion would release an aqueous aerosol of 100 mg/m^3 density⁽⁸⁾ into a cell of 800 m^3 volume having 9.5 air changes/hour. The concentration of the product takes place in 2 stages but the red oil explosion could occur in

either. By assuming the probability of a red oil explosion is the same in each, the weighted effect is found by adding the release quantities for the probability of one failing. These values are presented in Table 3.1.7. In preparing this table, for the in-cell releases from each of these concentrators, it is assumed that 8% of the daily input of Np, 1% of the Zr, Nb, Ru, Rh and 0.1% other fission products, and Am and Cm are in the concentrators.

Table 3.1.7
ESTIMATED CURIE RELEASES FROM A RED OIL
EXPLOSION IN THE FUEL PRODUCT CONCENTRATOR

Species	Reference Cycle	PT Cycle
Ru	1.0	0.9
Rh	1.0	0.9
Sr	2×10^{-2}	1.8×10^{-2}
Y	2×10^{-2}	19
Zr	1.5×10^{-2}	1.3×10^{-2}
Nb	3×10^{-2}	30
Cs	7×10^{-2}	6×10^{-2}
Ce	0.12	0.1
Pm	4×10^{-2}	3×10^{-2}
Pu	150	130
Am	8×10^{-7}	7×10^{-4}
Cm	1.6×10^{-2}	5×10^{-2}
Total	150	180

3.1.1.7 Explosion in the Fuel Product Denitrator

The difference between the conditions for a red oil explosion in a product concentrator and in a co-denitrator is that the primary safety mode in the former is through maintaining the temperatures below the reaction temperature

while in the latter the operating temperature is above the reaction temperature and explosions are avoided by burning solvent in the feed. The explosive danger in the product denitrator occurs at startup if solvent gets into the denitrator when it is below the reaction temperature and reacts when it is heated to operating temperature. This sequence of events is depicted in Figure B-1.5.

Solvent in Feed (C1) was evaluated in 3.1.1.4 and is taken to be the same probability $C1 = 2 \times 10^{-2}/D$.

Denitrator Cooldown, (C2). The material feeding the denitrator has to be carefully temperature controlled or else it will set up; but even if it is kept fluid, it is very viscous, typically requiring a screw feed. Cooling the denitrator when there is any concentrated product feed would be highly unusual and would likely be due to a failure in the denitrator resulting in inadvertent cooldown. Here it is assumed that heat failure to the denitrator could occur every 5 years, so $C2 = 0.2/\text{year}$.

Heavy Metal Present is expected: $C3 = 1$.

Equipment Failure Caused Transfer (C5) could result from temperature sensor failure and continued feed into an unheated denitrator. Balfanz⁽¹⁶⁾ provides 12 different failure rates, the geometric mean of which is $3 \times 10^{-2}/\text{year}$. This switch is checked with each monthly maintenance so the probability of its being failed when needed $C5 = 3 \times 10^{-2}/24 = 1 \times 10^{-3}/D$.

Human Error Caused Transfer (C6) is taken as a general error of commission⁽¹⁵⁾ $C6 = 10^{-2}/D$.

Denitrator Heated to Operating Temperature (C4) could be due to a human error of commission: $C4 = 10^{-2}/D$. It could also be due to intermittent controls that spontaneously bring the heater on but this seems more unlikely than human error.

In summary, Red Oil Explosion in Fuel Denitrator is the combination of these or $G1 = 2 \times 10^{-2}/D \times 0.2/\text{year} \times 1 \times 10^{-2}/D \times 10^{-2}/D = 4 \times 10^{-7}/\text{year}$.

This accident, should it occur, would take place in the same cell as the previous accident (3.1.1.6) and the same assumptions apply except that a powder aerosol would be formed with a density of 1 g/m^3 ⁽⁸⁾ hence 790 gm of fuel powder would be released as an aerosol in the cell. The elemental curie quantities are presented in Table 3.1.8.

Table 3.1.8

ESTIMATED CURIE RELEASES FROM A
FUEL PRODUCT CO-DENITRATOR EXPLOSION

Species	Reference Cycle	PT Cycle
Ru	18	17
Rh	18	17
Sr	0.4	0.3
Y	0.4	0.4
Zr	0.3	0.2
Nb	0.6	0.6
Cs	1.3	1.2
Ce	2.1	1.9
Pm	0.7	0.6
Pu	2,800	2,500
Am	0.014	15
Cm	0.3	900
Total	2,840	3,455

3.1.1.8 Criticality In a Process Cell

Fault Tree B.1-6 is the generic fault tree for a nuclear excursion in a radiochemical plant cell. It was developed from provisions of NRC Regulatory Guide 3.33. The sequences of events most likely to produce a nuclear excursion in a coprocessing plant are:

- Inadvertent separation of uranium and plutonium in partitioning
- Omission of soluble neutron poison and precipitation of plutonium in a non-geometrically safe configuration

- Omission of soluble neutron poison and excessive overbatching
- Transfer of fissile material to non-safe geometry (this is assured to require two independent errors by virtue of facility design)
- Accumulation of plutonium solids in non-safe geometry

In a coprocessing plant, an additional level of safety is afforded by not separating plutonium from uranium. However, some increase in solution reactivity occurs upon separation of a portion of the uranium; and upon addition of the mixed actinide nitrates from the WTF (in the PT case only). The latter effect, however, is small and is assumed to occur in geometrically favorable vessels.

Fault Tree B-1.6 is quantified as follows:

The plant is assumed to be designed to a minimum of two absolute barriers consisting of design and administrative controls to preclude a criticality. Inadvertent solution transfers either to Unsafe Geometry (G4) to Normally Safe Geometry (G5) or to Normally Safe Solutions such as overbatching (G6) are equally likely, the likelihood being determined by two independent task performance errors. The two errors would consist of Route Installed (C13) and Route Used (C14) each estimated at $10^{-2}/\text{yr}$ ⁽¹⁵⁾. Unsafe Geometry (C3) and Excess Fissile Solution (C4 and C5) all exist; these are assessed as 1. Therefore, Inadvertent Transfers (G4, G5 and G6) are each assessed at $10^{-4}/\text{cell-yr}$.

Solution Overconcentration (G7) requires quantification of two separate event sequences, Overconcentration (G16) and Dilution Failure (G17).

Dilution Supply Failure (C29) is assumed to fail due to human error as $10^{-2}/\text{yr}$. Dilution Control Failure (C30) is taken as $0.57/\text{yr}$ ⁽¹⁰⁾. Failure of Manual Override (C17) is taken to result from two independent human errors or $10^{-4}/\text{demand}$. Combining these gives $(10^{-4}) \times (0.01 + 0.57) = 5.8 \times 10^{-5}/\text{yr}$.

Instrumentation Failure (C27) is taken from reference 15 as $8.8 \times 10^{-3}/\text{yr}$. Heater Control Supply Failure (C28) is estimated to be $9 \times 10^{-3}/\text{yr}$ as before. Failure of Manual Override (C16) is as C17 above or $10^{-4}/\text{demand}$. Combining these gives $(10^{-4}) \times (1.8 \times 10^{-2}/\text{yr})$ or 1.8×10^{-6} for Solution Heater Control Failure (G16). Since overconcentration is always possible C6 is 1. Combining these gives a probability of $6 \times 10^{-5}/\text{yr}$ for Solution Overconcentration (G7).

Failure of Neutron Absorption (G8) may occur because of Insufficient Fixed Poison (C7) or Insufficient Soluble Poison (C8). Fixed Poisons must be monitored periodically to verify presence so C7 is taken to be ϵ . Insufficient Soluble Poison (C8) requires three independent task performance errors (failure to add, failure to verify, and failure to detect) for an estimated 100 batches per year or $(10^{-2}) \times (10^{-2}) \times (10^{-2}) \times (100)$ or $1 \times 10^{-4}/\text{yr}$. This means in turn that Failure of Neutron Absorption (G8) is equal to $1 \times 10^{-4}/\text{yr}$.

Precipitation of Fissile Material from Solution (G9) requires quantifying two independent event sequences.

Precipitating Additives (G14) is assumed to be governed by inadvertent in-cell transfer, which was assessed at $10^{-4}/\text{yr}$ in G4, G5 and G6.

Process Control Failure (G15) requires Instrument Failure (G19), Mechanical Failure (G20), and Operator Oversight (C15). Instrument Failures (G19) are taken to be $0.5/\text{yr}^{(10)}$ since for any given process unit, only one of the instrument failures need apply.

For Mechanical Failure (G20), both supply systems will be present for any unit operation. Steam System Malfunction (C25) was previously assessed at $9 \times 10^{-3}/\text{yr}$. The acid supply is not known in detail so is assumed to be the same as the steam system or $9 \times 10^{-3}/\text{yr}$. Monitoring Less Than Adequate (C15) is a single operator error or $10^{-2}/\text{demand}$.

Process Control Failure (G15) may then be assessed by combining G19, G20, and C15, or $(0.5) \times (10^{-2}) \times (1.8 \times 10^{-2}) = 9 \times 10^{-5}/\text{yr}$. Precipitation of Fissile Solids (G9) may then be assessed by combining G14 and G15 or $(10^{-4} + 8 \times 10^{-6}) = 2 \times 10^{-4}/\text{yr}$ for G9.

Moderator Introduction (G10) requires either a fire or a fire protection system malfunctioning or an unspecified method of filling the cell with water. Moderator introduction under these conditions is assessed as ϵ by virtue of the cell connection to the air tunnel which would provide an outlet for any liquid.

Deformation to Nonsafe Geometry (G11) is also assumed to be ϵ in that a low probability accident is required for this to occur, i.e., dropped crane load, vessel void space explosion, nonrupturing missile impact, etc.

Potential Criticality in Cell (G2) may then be estimated by combining the subordinate events G4 through G11. The sum is $6.4 \times 10^{-4}/\text{yr}$.

Failure of Personnel Action (C1) is taken as 1 since the rapidity of the excursion would prevent effective action. Failure of Criticality Instrumentation (C2) to detect unsafe accumulations is assessed at $1 \times 10^{-2}/\text{yr}^{(10)}$.

Combining Potential Criticality (G2) and Failure to Prevent (G1) gives $6.4 \times 10^{-6}/\text{cell-yr}$ for Criticality in Cell (G1).

In the FRP there are three cells of concern; the RPC, HLC, and HILC. The FPC is not of concern by virtue of geometrical safe or favorable equipment cell sumps and welded piping. This cell multiplicity gives an expected criticality frequency of $3 \times 6.4 \times 10^{-5}$ or 2×10^{-5} per year in the FRP.

Should this accident occur, it is assumed to result in 1×10^{19} fissions in 8 hours (Regulatory Guide 3.33). Physical damage does not usually accompany criticality accidents but it is assumed that the tank in which it occurs vents into the cell releasing all the noble gases from the reaction and 25% of the iodine. The releases are presented in Table 3.1.9.

3.1.1.9 Failure of a Krypton Storage Cylinder

This accident is not modeled using a fault tree, but uses the analysis of Hodge and Jarrett⁽¹⁷⁾ who estimated the probability of cylinder failing in a truck accident as $2 \times 10^{-6}/\text{accident}$. They estimated 0.033 cylinders/MTU of fuel processed. For a 2,000 MTU plant, this is 66 cylinders/year. Assuming the handling of cylinders in the plant involves no more risk than the transportation environment $2 \times 10^{-6} \times 66 = 1.3 \times 10^{-4}/\text{year}$ is conservatively estimated as the failure rates of these cylinders.

Should this accident occur, it is estimated that 1×10^6 curies of 85 Kr are released in 30 minutes at a height of 100 m.

3.1.1.10 Hydrogen Explosion in Uranium Reductor

After denitration of the uranium product, it must be reduced in a hydrogen atmosphere prior to fluorination to UF_6 . An explosion in the reduction furnace may result from improper mixing of the reducing gas, improper air purge on startup or oxygen evolution in the reduction process. This is diagrammed in the fault tree of Figure B-1.7.

Table 3.1.9

ESTIMATED RADIONUCLIDE RELEASES FOR A
CRITICALITY ACCIDENT IN A PROCESSING CELL

Species	Curies Released in 8 Hours
Kr-83m	37
Kr-85m	170
Kr-87	1,000
Kr-88	660
Kr-89	41,000
Xe-133m	.55
Xe-133	13
Xe-135m	110
Xe-135	170
Xe-137	39,000
Xe-138	11,000
I-131*	2
I-132	7
I-133	35
I-134	480
I-135	120
Total	93,805

*It is assumed that the releases are vented to the cell bypassing the vessel offgas treatment. If the gases are confined in the vessel vent, the iodine fraction will be reduced by a factor of 10^3 - 10^4 by iodine removal equipment.

Mixing Valve Failure, (C1). The Reliability Analysis Center^(17A,p.74) gives a mixing valve failure rate $C1 = 2 \times 10^{-2}/\text{year}$.

Valve Shutoff Failure (C2) is given^(17A,p.203) as $C2 = 1 \times 10^{-4}/D$.

Combustion Detector 1 and 2 failure rates are given in Anyakora, et al.^(10,p.399) as $C4 = C5 = 4 \times 10^{-2}/D$ where in the above, a 1 month test and maintenance interval is assumed.

Failure to Properly Purge Furnace (C5) is estimated as a failure to comply^(18,p.11), $C5 = 1.4 \times 10^{-4}/D$.

Failure of Supervision (C6) is estimated^(18,p.11) as $C6 = 3.2 \times 10^{-5}/D$.

Assuming 100 purges/year but that there is a common-mode interaction between supervision and supervised. This common mode is estimated as the geometric mean of complete independence and complete dependence. This step provides an estimated improper purge frequency $G3 = 8 \times 10^{-4}/\text{year}$.

Oxygen Evolution (C7) is taken as ϵ since the furnace treats only MOX.

Hydrogen Explosion in the Reduction Furnace is the combination of these probabilities $G1 = 9 \times 10^{-4}/\text{year}$.

This accident would occur in the Fuel Product Cell and the releases would be as indicated in Table 3.1.8.

3.1.1.11 Fuel Assembly Drop

A fuel assembly drop could occur during transfer from the cask to the storage basket, from the basket to the conveyor or from the conveyor to the shear. This latter occurrence is of minor effect compared with what the shear does to the fuel assembly. All of these transfer operations take place under water which will impede the rate of fall should an element be dropped.

No fault tree is used in the analysis of this event, but it is considered to be the product of the probability of dropping an element and the probability of breaking from the fall.

Frequency of Element Dropping. Reactor operating experience^(19,20) indicates 2 instances of dropped elements. Assuming 300 fuel transfers per year-reactor and 250 plant years of experience, the probability is $3 \times 10^{-5}/\text{transfer}$. Each assembly contains about 0.45 mt. so 4,400 elements must be handled for a plant throughput of 2,000 mt.; hence, the probability of a drop $p_D = 2.7 \times 10^{-5} \times 4,400 = 0.12/\text{year}$.

Probability of Breaking When Dropped. If this were fresh fuel, greater than a 30-foot drop would be necessary to break it (10CFR70); however, spent fuel is believed to be more fragile. If an element falls through water, there is a tendency to fall in an orientation that minimizes drag, hence hitting on the end which is the strongest orientation. Based on this, it is estimated that the probability of breaking in a water drop is 1%.

Probability of a Fuel Assembly Fracture is $0.12 \times 0.01 = 1.2 \times 10^{-3}$ /year.

Should this accident occur, the estimated releases are 1,300 Ci of ^{85}Kr and 0.2 Ci of ^{133}Xe . (It is assumed that iodine is retained in the water.)

3.1.1.12 Hydrogen Explosion in Fuel Product Denitrator Feed Tank

Alpha emitters are highly concentrated in this feed tank with a consequent high rate of radiolytic hydrogen evolution. However, the system is designed for this and therefore accident probabilities are modeled as in 3.1.1.2 with a probability of 3×10^{-4} /year.

Should this occur, it would be in the Fuel Product Cell and the releases would be as indicated in Table 3.1.8.

3.1.1.13 Accident Summary

The following tables summarize these accidents and atmospheric releases using HEPA filter penetration fraction data from Appendix A.6. In the FRP, all accident releases must pass through deep bed and double HEPA filtration in order to reach the atmosphere. Accidents from dry processes are assumed to possess aerosol sizes characteristic of dry processes as given in Appendix A.6, and similarly for wet processes.

Table 3.1.9 assumes no filter failures and uses the probabilities directly as determined. Table 3.1.10 assumes one filter failure and adjusts the probability accordingly, and Table 3.1.11 assumes two filter failures with appropriate probability adjustments. Reference cycle FRP radiological accidents are designated "a" and the PT cycle accidents are designated "b". The overall radiological accident risk of the Reference FRP is the sum of the risks over the failure states of the HEPA filters. These results are presented in Table 3.1.12. It will be noted that the risk of failing a krypton cylinder is the dominant risk for either cycle to the extent that this overshadows the isotopic differences in the plants.

Table 3.1.9a

PROBABILITIES, CONSEQUENCES AND RISK FOR THE REFERENCE CYCLE FRP RADIOLOGICAL ACCIDENTS
(No HEPA Filters Failed)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.1.1.2	H ₂ fire and exp. in HAF tank	3×10^{-4}	3×10^{-8}	9×10^{-8}	1.1×10^{-10}	3×10^{-11}	3×10^{-14}
3.1.1.3	Solvent fire in HA contractor	2×10^{-4}	9×10^{-8}	7×10^{-8}	9×10^{-10}	1.4×10^{-11}	1.8×10^{-13}
3.1.1.4	Red oil exp. in H-W conc.	4×10^{-6}	2×10^{-7}	8×10^{-7}	2×10^{-10}	3×10^{-12}	9×10^{-16}
3.1.1.5	Explosion in H-W calciner	2×10^{-5}	7×10^{-6}	2×10^{-5}	7×10^{-9}	8×10^{-11}	1.4×10^{-13}
3.1.1.6	Red oil exp. in fuel prod. conc.	4×10^{-6}	9×10^{-8}	6×10^{-8}	9×10^{-10}	3×10^{-13}	4×10^{-15}
3.1.1.7	Exp. in fuel product denitr.	4×10^{-7}	1.4×10^{-7}	1.0×10^{-7}	1.4×10^{-9}	4×10^{-14}	6×10^{-16}
3.1.1.8	Criticality in process cell	2×10^{-5}	9×10^4	2	2×10^{-4}	4×10^{-5}	4×10^{-9}
3.1.1.9	Failure of krypton cylinder	1.3×10^{-4}	1×10^6	4×10^1	6×10^{-3}	5×10^{-3}	7×10^{-7}
3.1.1.10	Hydrogen exp. in reductor	9×10^{-4}	1.7×10^{-6}	1.2×10^{-6}	1.7×10^{-8}	1.1×10^{-9}	1.5×10^{-11}
3.1.1.11	Fuel assembly drop	1.2×10^{-3}	1×10^3	5×10^{-2}	7×10^{-6}	6×10^{-5}	9×10^{-9}
3.1.1.12	Hydrogen exp. in fuel product denitr. feed	3×10^{-4}	1.7×10^{-6}	1.2×10^{-6}	1.7×10^{-8}	4×10^{-10}	5×10^{-12}
Total Risk						5×10^{-3}	7×10^{-7}

Table 3.1.9b

PROBABILITIES, CONSEQUENCES AND RISK FOR THE PT CYCLE FRP RADIOLOGICAL ACCIDENTS
(No HEPA Filters Failed)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.1.1.2	H ₂ fire and exp. in HAF tank	3×10^{-4}	3×10^{-7}	8.4×10^{-8}	1.2×10^{-10}	3×10^{-11}	4×10^{-14}
3.1.1.3	Solvent fire in HA contractor	2×10^{-4}	1×10^{-6}	7.3×10^{-8}	9×10^{-10}	1.5×10^{-11}	1.9×10^{-13}
3.1.1.4	Red oil exp. in HLW conc.	4×10^{-6}	2×10^{-6}	7×10^{-7}	3×10^{-10}	3×10^{-12}	1.2×10^{-15}
3.1.1.5	Explosion in HLW calciner	2×10^{-5}	7×10^{-6}	2.2×10^{-6}	5×10^{-10}	4×10^{-11}	1×10^{-14}
3.1.1.6	Red oil exp. in fuel prod. conc.	4×10^{-6}	1×10^{-6}	5.1×10^{-8}	7×10^{-10}	2×10^{-13}	3×10^{-15}
3.1.1.7	Exp. in fuel product denitr.	4×10^{-7}	2×10^{-6}	3×10^{-7}	4×10^{-9}	1.2×10^{-13}	1.6×10^{-15}
3.1.1.8	Criticality in process cell	2×10^{-5}	9×10^4	2	2×10^{-4}	4×10^{-5}	4×10^{-9}
3.1.1.9	Failure of krypton cylinder	1.3×10^{-4}	1×10^{-6}	3.8×10^1	6×10^{-3}	5×10^{-3}	7×10^{-7}
3.1.1.10	Hydrogen exp. in reductor	9×10^{-4}	1.5×10^{-5}	1×10^{-6}	1.4×10^{-8}	9×10^{-10}	1.3×10^{-11}
3.1.1.11	Fuel assembly drop	1.2×10^{-3}	1.3×10^3	4.9×10^{-2}	7×10^{-6}	6×10^{-5}	9×10^{-9}
3.1.1.12	Hydrogen exp. in fuel product denitr. feed	3×10^{-4}	1.5×10^{-5}	1×10^{-6}	1.4×10^{-8}	3×10^{-10}	4×10^{-12}
Total Risk						5×10^{-3}	7×10^{-7}

Table 3.1.10a

PROBABILITIES, CONSEQUENCES AND RISK FOR THE REFERENCE CYCLE FRP RADIOLOGICAL ACCIDENTS
(1 HEPA Filter Failed)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.1.1.2	H ₂ fire and exp. in HAF tank	3×10^{-6}	3×10^{-4}	9×10^{-4}	1.1×10^{-6}	3×10^{-9}	3×10^{-12}
3.1.1.3	Solvent fire in HA contractor	2×10^{-6}	9×10^{-4}	7×10^{-4}	9×10^{-6}	1.4×10^{-9}	1.8×10^{-11}
3.1.1.4	Red oil exp. in HLW conc.	4×10^{-8}	2×10^{-3}	8×10^{-3}	2×10^{-6}	3×10^{-10}	9×10^{-14}
3.1.1.5	Explosion in HLW calciner	2×10^{-7}	8×10^{-2}	2×10^{-1}	8×10^{-5}	5×10^{-8}	1.6×10^{-11}
3.1.1.6	Red oil exp. in fuel prod. conc.	4×10^{-8}	9×10^{-4}	6×10^{-4}	9×10^{-6}	3×10^{-11}	4×10^{-13}
3.1.1.7	Exp. in fuel product denitr.	4×10^{-9}	1.7×10^{-2}	1.2×10^{-2}	1.7×10^{-4}	5×10^{-11}	7×10^{-13}
3.1.1.8	Criticality in process cell	2×10^{-7}	9×10^4	5×10^{-1}	7×10^{-5}	5×10^{-9}	7×10^{-13}
3.1.1.9	Failure of krypton cylinder	1.3×10^{-6}	1×10^6	4×10^1	6×10^{-3}	5×10^{-5}	7×10^{-9}
3.1.1.10	Hydrogen exp. in reductor	9×10^{-6}	1.7×10^{-2}	1.2×10^{-2}	1.7×10^{-4}	1.1×10^{-7}	1.5×10^{-9}
3.1.1.11	Fuel assembly drop	1.2×10^{-5}	1.3×10^3	5×10^{-2}	7×10^{-6}	6×10^{-7}	9×10^{-11}
3.1.1.12	Hydrogen exp. in fuel product denitr. feed	3×10^{-6}	1.7×10^{-2}	1.2×10^{-2}	1.7×10^{-4}	4×10^{-8}	5×10^{-10}
Total Risk						5×10^{-5}	9×10^{-9}

Table 3.1.10b

PROBABILITIES, CONSEQUENCES AND RISK FOR THE PT CYCLE FRP RADIOLOGICAL ACCIDENTS
(1 HEPA Filter Failed)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.1.1.2	H ₂ fire and exp. in HAF tank	3×10^{-6}	3×10^{-4}	8×10^{-4}	1.2×10^{-6}	3×10^{-9}	4×10^{-12}
3.1.1.3	Solvent fire in HA contractor	2×10^{-6}	1×10^{-3}	7×10^{-4}	9×10^{-6}	1.5×10^{-9}	1.9×10^{-11}
3.1.1.4	Red oil exp. in HLW conc.	4×10^{-8}	2×10^{-3}	7×10^{-3}	3×10^{-6}	3×10^{-10}	1.2×10^{-13}
3.1.1.5	Explosion in HLW calciner	2×10^{-7}	8×10^{-2}	3×10^{-1}	6×10^{-5}	5×10^{-8}	1.2×10^{-11}
3.1.1.6	Red oil exp. in fuel prod. conc.	4×10^{-8}	1×10^{-3}	5×10^{-4}	7×10^{-6}	2×10^{-11}	3×10^{-13}
3.1.1.7	Exp. in fuel product denitr.	4×10^{-9}	2×10^{-2}	3×10^{-1}	4.2×10^{-4}	1.2×10^{-10}	1.7×10^{-13}
3.1.1.8	Criticality in process cell	2×10^{-7}	5,200	0.5	7×10^{-5}	5×10^{-9}	7×10^{-13}
3.1.1.9	Failure of krypton cylinder	1.3×10^{-6}	1×10^6	40	6×10^{-3}	5×10^{-5}	7×10^{-9}
3.1.1.10	Hydrogen exp. in reductor	9×10^{-6}	1.5×10^{-2}	1×10^{-2}	1.4×10^{-4}	9×10^{-8}	1.3×10^{-9}
3.1.1.11	Fuel assembly drop	1.2×10^{-5}	1.3×10^3	5×10^{-2}	7×10^{-6}	6×10^{-7}	9×10^{-11}
3.1.1.12	Hydrogen exp. in fuel product denitr. feed	3×10^{-6}	1.5×10^{-2}	1×10^{-2}	1.4×10^{-4}	3×10^{-8}	4×10^{-10}
Total Risk						5×10^{-5}	9×10^{-9}

Table 3.1.11a

PROBABILITIES, CONSEQUENCES AND RISK FOR THE REFERENCE CYCLE FRP RADIOLOGICAL ACCIDENTS
(2 HEPA Filters Failed)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total latent Cancer	Total Person-Rem Plant-Yr	Total latent Cancer Plant-Yr
3.1.1.2	H ₂ fire and exp. in HAF tank	8×10^{-11}	4	1.2×10^1	1.5×10^{-2}	1×10^{-9}	1.2×10^{-12}
3.1.1.3	Solvent fire in HIA contractor	5×10^{-11}	15	1.1×10^1	1.5×10^{-1}	6×10^{-10}	7×10^{-12}
3.1.1.4	Red oil exp. in HLW conc.	1×10^{-12}	30	1.1×10^2	3×10^{-2}	1.1×10^{-10}	3×10^{-14}
3.1.1.5	Explosion in HLW calciner	5×10^{-12}	1.4×10^3	4×10^3	1.4×10^0	2×10^{-8}	7×10^{-12}
3.1.1.6	Red oil exp. in fuel prod. conc.	1×10^{-12}	15	1.1×10^1	1.5×10^{-1}	1.1×10^{-11}	1.5×10^{-13}
3.1.1.7	Exp. in fuel product denitr.	1×10^{-13}	3×10^2	2×10^2	3×10^0	2×10^{-11}	3×10^{-13}
3.1.1.8	Criticality in process cell	6×10^{-12}	9×10^4	2.1	2.4×10^{-4}	6.3×10^{-13}	7.2×10^{-17}
3.1.1.9	Failure of krypton cylinder	4×10^{-11}	1×10^6	4×10^1	6×10^{-3}	1.5×10^{-9}	2×10^{-13}
3.1.1.10	Hydrogen exp. in reductor	2×10^{-10}	3×10^2	2×10^2	3×10^0	4×10^{-8}	6×10^{-10}
3.1.1.11	Fuel assembly drop	3×10^{-10}	1.3×10^3	5×10^{-2}	7×10^{-6}	1.5×10^{-11}	2×10^{-15}
3.1.1.12	Hydrogen exp. in fuel product denitr. feed	8×10^{-11}	3×10^2	2×10^2	3×10^0	1.7×10^{-8}	2×10^{-10}
Total Risk						8×10^{-8}	8×10^{-10}

Table 3.1.11b

PROBABILITIES, CONSEQUENCES AND RISK FOR THE PT CYCLE FRP RADIOLOGICAL ACCIDENTS
(2 HEPA Filters Failed)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.1.1.2	H ₂ fire and exp. in HAF tank	8×10^{-11}	5	1.4×10^1	1.9×10^{-2}	1.1×10^{-9}	1.5×10^{-12}
3.1.1.3	Solvent fire in HIA contractor	5×10^{-11}	16	1.2×10^1	1.5×10^{-1}	6×10^{-10}	8×10^{-12}
3.1.1.4	Red oil exp. in HLW conc.	1×10^{-12}	40	1.4×10^2	6×10^{-2}	1.4×10^{-10}	6×10^{-14}
3.1.1.5	Explosion in HLW calciner	5×10^{-12}	1.4×10^3	4×10^3	11	2×10^{-8}	5×10^{-12}
3.1.1.6	Red oil exp. in fuel prod. conc.	1×10^{-12}	18	9	1.2×10^{-1}	9×10^{-12}	1.2×10^{-13}
3.1.1.7	Exp. in fuel product denitr.	1×10^{-13}	3×10^2	5×10^2	7	1.5×10^{-10}	2×10^{-12}
3.1.1.8	Criticality in process cell	6×10^{-12}	9×10^4	2.1	2.4×10^{-4}	1.2×10^{-13}	2×10^{-15}
3.1.1.9	Failure of krypton cylinder	4×10^{-11}	1×10^6	40	6×10^{-3}	1.5×10^{-9}	2×10^{-13}
3.1.1.10	Hydrogen exp. in reductor	2×10^{-10}	3×10^2	2×10^2	3	4×10^{-8}	6×10^{-10}
3.1.1.11	Fuel assembly drop	3×10^{-10}	1.3×10^3	5×10^{-2}	7×10^{-6}	1.5×10^{-11}	2×10^{-15}
3.1.1.12	Hydrogen exp. in fuel product denitr. feed	8×10^{-11}	3×10^2	2×10^2	3	1.7×10^{-8}	2×10^{-10}
Total Risk						8×10^{-8}	8×10^{-10}

Table 3.1.12

SUMMARY OF RADIOLOGICAL ACCIDENT RISKS FOR THE FRP

Cycle	Risk Measure			
	<u>Total Person-Rem Plant-Year</u>	<u>Total Latent Cancer Plant-Year</u>	<u>Total Person-Rem GWe-Year</u>	<u>Total Latent Cancer GWe-Year</u>
Reference	5×10^{-3}	7×10^{-7}	7×10^{-5}	9×10^{-9}
PT	5×10^{-3}	7×10^{-7}	7×10^{-5}	9×10^{-9}

3.1.2 Relative Public Atmospheric Risk of PT and Reference FRP

Consequences to the public from routine radioactive effluents from the PT and Reference FRPs are developed in Appendix G. The results for the two plants are presented in Table 3.1.13.

Table 3.1.13

SUMMARY ROUTINE RADIOLOGICAL RISKS TO PUBLIC FOR THE FRP

Cycle	<u>Total Person-Rem Plant-Year</u>	<u>Total Latent Cancer Plant-Year</u>	<u>Total Person-Rem GWe-Year</u>	<u>Total Latent Cancer GWe-Year</u>
Reference	680	0.12	9.1	1.6×10^{-3}
PT	730	0.29	9.7	3.9×10^{-3}

3.1.3 Relative Occupational Radiological Risks of PT and Reference FRP

No difference between the reference FRP and the PT FRP occupational risks is discerned. Both facilities are to be so designed as to shield workers from combined neutron and gamma radiation. The design criterion is set at 500 millirem-man per year. All activities are conducted so as to meet ALARA guidelines. The occupational exposures are given in Table 3.1.14. Three

exposures are estimated routine, maintenance, and accidental exposure. The accidental exposure has three component parts; abnormal occurrences, exposure as a result of accidents, and exposure during repair and recovery efforts from the accident. The calculations and assumptions made to arrive at the tabulated values are given in Appendix D.

3.1.4 Relative Nonradiological Risks of PT and Reference FRP

No difference between the Reference FRP and the PT FRP is discerned. The chemical risks and industrial risks developed in Appendix E are summarized in Table 3.1.15. The primary chemical hazards are associated with the site power plant effluents. The health effects associated with these effluents completely dominate the nonradiological risk.

Table 3.1.14
ESTIMATED OCCUPATIONAL EXPOSURE
FRP

Activity	Person-Rem Plant-Year	Person-Rem GWe-Year	Fatalities** GWe-Year
Routine Operation	220	3	3×10^{-4}
Maintenance	246	3	3×10^{-4}
Abnormal Occurrence*	10	0.1	1×10^{-5}
Accidents	1×10^{-2}	1×10^{-4}	1×10^{-8}
Decommissioning	39	0.5	5×10^{-5}
Total			6.6×10^{-4}

*Defined as personnel exposure exceeding limits set in 10CFR20.

**From the BEIR report equivalence estimate of 1×10^{-4} fatalities per person-rem.

Table 3.1.15

SUMMARY OF EFFECTS ON NONRADIOLOGICAL CAUSES

Cause	Premature Deaths (fatality/GWe-Yr)	Economic Death (fatality/GWe-Yr)	Total (fatality/GWe-Yr)
Effluents	0.05	0.19	0.24
Industrial Accidents	3×10^{-4}	0.02	0.02
Total			0.26

3.2 RISK OF THE FUELS REPROCESSING PLANT - WASTE TREATMENT FACILITIES (FRP-WTF)

3.2.1 FRP-WTF Dominant Accidents

3.2.1.1 Overview and Preliminary Hazards Analysis

The Fuels Reprocessing Plant Waste Treatment Facility (FRP-WTF) is summarized in Appendix A.2. Facility description, unit operations and process flows, and material quantity flows are presented. The FRP-WTF receives all actinide bearing waste streams generated within the reprocessing plant. Actinide bearing solids are either leached with ceric acid or with hydrofluoric acid, the latter being used for solids containing gross quantities of ruthenium. Liquid waste streams are subjected to various solvent extraction processes depending on waste origin and nature.

A preliminary scoping of accidents is given in Table 3.2.1 for the WTFs. The unit operations and tasks likely to significantly increase the risk to operating staff and to the general offsite population were examined in light of known process hazards. Those accidents judged to be serious, and the associated causes are listed. Characteristic source terms and qualitative estimates of significance are made to scope and rank potential accidents. From these, the accidents were selected for analysis. Table 3.2.2 lists these accidents and the appropriate section of this report.

Table 3.2.1 WTF PRELIMINARY SCOPING OF ACCIDENTS

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Waste Treatment Facility	Fractional dispersion of facility inventory	Seismic event exceeds DBE	Primary and secondary containment are assumed breached	Liquids, solids, aerosols (dusts, smoke and soot)	Extremely low proba- bility of occurrence, potentially signifi- cant release offsite
		Tornado exceeds DBT	same as above	same as above	same as above
		Fire exceeds DBA	same as above	same as above	same as above
Waste Treatment Facility Ventilation System	Radiolytic H ₂ explo- sion	[H ₂] > 4 vol% plus source of ignition	Dilution of H ₂ by normal air flow to less than explosive limits	Dust and fines deposited on duct work and filters	Low (reactor experience)
	Ruthenium absorber overheats and releases inventory	Excessive load on bed exceeds cooling capa- bility	Temperature monitor- ing, subsequent off- gas treatment pro- vides Ru df	RuO ₄	Low
	Loss of ventilation or air flow reversal	Normal ventilation exhaust fails and emer- gency backups fail to actuate	Multiple non-common mode failures required	Dusts, aerosols, liquids	low, primarily occupational risk
		Zone 1 supply exceeds exhaust capability	same as above	same as above	same as above
Waste Treatment Facility Closed Loop Cooling/Heating Systems Fail	Loss of utility	Total power outage; or failure to isolate failed unit; loss of onsite services	Component isolation capability, backups for power and onsite services	Liquids, aerosols and dusts	Low
Waste Treatment Facility Loss of Com- pressed Air	Loss of utility	same as above	same as above	same as above	Low
Solid Waste Shipping/ Receiving and Handling	Spill of contaminated ash	Containers rupture • overheat • H ₂ explosion • container drop	Cell filter and facility exhaust filtration preclude release	Fine powder	Likely to occur, but limited to contents of one container
	Fire in combustible waste storage/handling	Spontaneous combustion, decay heat or ignition source	same as above	Ash and soot	Likely to occur, limited to storage cubicle inventory
	Spread of contamina- tion during manipula- tor changeout	Breached doughnut or greenhouse	Operating gallery ventilation filtra- tion precludes release	Powder	Likely, primarily contributor to occupational risk

Table 3.2.1 (Cont.)

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Liquid Feed/Waste/ Product Transfer	Loss of transfer capability with self boiling in transfer line	Pump failure, jet gases out or line plugs	Line slopes to fully drain, line enclosed in second pipe, encased and buried in soil	Liquid, aerosol and possibly some volatilized fission products; batch size about 3000 product, 20000 waste (max.)	Interruption of transfer likely, uncontrolled release very unlikely
	Loss of cooling on product transfer line	same as above	same as above	same as above	same as above
	Line fails	Corrosion or weld/pipe failure	same as above	Liquid (same range of batch sizes as above)	Low frequency, minimal consequence
FRP-WTF Hulls Leach	Zircalloy fire	Ignition of zircalloy fines	Material either packaged or covered by liquid; main exhaust filtration system intact	Aerosol (smoke and soot), volatilized radionuclides	Low probability of occurrence
	Criticality	Gadolinium omitted from leachate, plutonium selectively precipitated	Two barriers required to prevent sequence of events resulting in excursion	Noble fission product gases	Low probability of occurrence, potentially significant release
	Fluoride corrosion of subsequent process equipment • coils • tube bundles	Failure to complex with ANN	Coils/tube bundles on closed loop utility system	Dilute leachate	Primarily occupational risk
Dissolver Solids Leach (FRP-WTF)	Formation of volatile ruthenium	Dissolutions and oxidations of ruthenium metal	Oxidizing conditions not strong enough, offgas ruthenium absorber	RuO	Low
	Criticality	same as Hulls Leach, above	same as Hulls Leach, above	same as Hulls Leach, above	same as Hulls Leach, above
	Fluoride corrosion of subsequent process equipment • coils • tube bundles	same as Hulls Leach, above	same as Hulls Leach, above	same as Hulls Leach above	same as Hulls Leach above
CHP/TBP Solvent Extraction	Solvent fire	Solvent leak plus ignition	Low temperatures, vessel integrity exhaust filtration	Smoke, soot, volatile radionuclides	Solvent leaks likely, ignition conditions unlikely. Fire duration brief due to suppression systems
	Nitrated Solvent Explosion	"Red oil" formation and detonation in concentrators	Temperature limitations on concentrates, decanting of feed, exhaust gas filtration	Liquid aerosol volatile fission products	"Red oil" formation unlikely due to temperature limits

Table 3.2.1 (cont.)

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
CMP/TMP Solvent Extraction (cont.)	Radiolytic hydrogen explosion	Loss of purge or dilution capability and ignition conditions in tanks	Lack of ignition source, exhaust gas filtration	Liquid, aerosol maximum H ₂ generation rate in feed/product tanks	Low due to lack of ignition source and passive purge by vessel in-leakage
	"Blowback"/"Suckback"	Line connecting process vessel and access gallery pressurized on process side (pulse leg, wf/spg, dip tube, jets, etc.) with respect to gallery	Hydraulic head between process and personnel zones, process vented to preclude pressurization personnel gallery vent filters	Liquid, aerosol	Mostly contributes to occupational risk
	Criticality	Plutonium separated by precipitation in non-safe geometry	Two barriers required preventing a sequence of events resulting in a criticality	Noble fission product gases released	Low probability of occurrence potentially significant release
Salt Waste Treatment	Solvent Fire	Solvent leak plus ignition	Low temperature vessel integrity exhaust filtration	Smoke, soot volatile radionuclides maximum tank volume is 4000± 2-EH0H	2-EH0H greater fire hazard than CMP, other wise see CMP, above
	"Blowback"/"Suckback"	see CMP above	see CMP above	see CMP above	see CMP above
	Solvent mist/vapor explosion	Leak from pressurized line produces 2-EH0H mist which ignites	Cell air purge dilutes mist, exhaust filtration	Loose contamination in cell, material on cell exhaust filter if filter disrupted	Unlikely as long as cell ventilation flow maintained; mist generating operations would be suspended if vent flow lost
Cation Exchange Chromatography	Degraded resin explodes	<ul style="list-style-type: none"> Loss of cooling and loaded resin Loaded resin dries out Degraded resin exposed to high nitric 	Emergency resin dump to quench tank, small column sizes, individual cubicles to prevent explosion propagation	Smoke, soot resin beads and fragments possible dusting. Maximum resin-41± per rack, maximum tank ≤ 410± resin	Moderate probability of occurrence but small inventories and isolation mitigate consequences
	Hydrogen explosion	Hydrogen released from scrubs, elutants, etc., upon return to ambient pressure explodes	Dilution by air and water vapor	Liquid, aerosol	Lack of ignition source
		Radiolytic hydrogen generated in feed/product tanks explodes	same as above	same as above	same as above
	"Blowback"	Check valve and pump failure	Venting in secondary confinement zone	Liquid, aerosol limited to rack inventory or tank inventory	Moderate, primary occupational risk

Table 3.2.1 (cont.)

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Low Temperature Ashing	Organic vapor explosion in furnace	Introduction of combustible organics into furnace with ignition of vapors	Feed sorted, feed non-combustible furnace offgas system direct and dissipate energy	Ash	Low
	Spread of contamination from manipulator removal	see Solid Waste	see Solid Waste	see Solid Waste	see Solid Waste
	Criticality	Plutonium accumulation and precipitation in scrubber tanks	Sintered metal filters (Pu to tank)	Noble fission product gases	Low
Ceric Acid Leach System	Structural leakage	Ceric acid leak breaches cell liner seepage via expansion joint to soil	Protective curbs on expansion joints	Leachate	Primarily occupational risk, remote possibility of seepage to ground water.
	RuO ₄ release	Ruthenium oxidized to RuO ₄ in leacher and released via vessel vent or via cell vent (spill)	Ruthenium absorber in facility offgas treatment	300 curies Ru per MTHM equivalent	Low potential for breakthrough in offgas absorber. Direct release to cell released via stack
	Hydrogen explosion	Radiolytic H ₂ ignited	Purge gas dilutes H ₂ , no source of ignition	Leachate, aerosol	Low, water vapor in gas would blanket reaction
	Criticality	Omission of Gadolinium and overcharge leacher or precipitate plutonium	Two barriers required to preclude sequence of events resulting in excursion	Noble fission product gas, potential boiling and aerosol generation	Low probability of occurrence, potentially significant release
	Ceric acid corrosion of subsequent process equipment • coils • tube bundles	Failure to reduce ceric to cerous with oxalic acid	Redundant system, process control simple cooling/heating water and steam utilities are closed loop	Leachate	Moderate consequences primarily occupational risk
Actinide Product Concentration	Radiolytic H ₂ explosion	Accumulation of H ₂ due to failure purge, source of ignition	Lack of ignition, exhaust gas filtration	Liquid, aerosol	Low due to lack of ignition source. Effects instigated by small H ₂ volume available
	"Red oil" explosion	Nitrated organic/resin in concentrator explodes	Solution decanted/filtered	Liquid, aerosol, product solution	Low
		Nitrated organic and product solution explodes in sump	same as above, spillage minimized	Dust, Am/Cm oxides	Low

Table 3.2.1 (cont.)

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Actinide Product Concentration (cont.)	Fire	Calcining product ignites combustibles in cell	Lack of combustibles in cell solution spill unlikely, cell, vent exhaust filtration	Soot, ashes	Low
	Criticality	Spilled product calcines in non-geometrically safe configuration	Spillage minimized, two barriers in place to prevent sequence of events resulting in an excursion	Noble fission product gases	Low, sump geomet- rically safe
		Plutonium selectively precipitated in non- geometrically safe concen- tration	Two barriers in place to prevent sequence of events resulting in an excursion	same as above	Low

Table 3.2.2
ACCIDENTS SELECTED FOR ANALYSIS

Accident Section	Accident Description
3.2.1.2	Waste Treatment Facility Criticality
3.2.1.3	Radiolytic Hydrogen Explosion Vessel Vessel Offgas System
3.2.1.4	Resin Explosion in Actinide Product Concentrator
3.2.1.5	"Red Oil" Explosions in WTF Concentrators
3.2.1.6	Resin Explosion/Fire
3.2.1.7	Facility Fire Solvent Fire Solvent Mist Explosion
3.2.1.8	Accident Summary

3.2.1.2 Waste Treatment Facility Criticality

Fault Tree B.1-6 is the generic fault tree for a nuclear excursion in a radiochemical plant cell developed previously from provisions of NRC Regulatory Guide 3.3.3 (see Section 3.1). The sequences of events most likely to produce a nuclear excursion are:

- Omission of soluble neutron poison and precipitation of plutonium in a nongeometrically safe configuration
- Omission of soluble neutron poison and excessive overbatching
- Transfer of fissile material to nonsafe geometry (this is assumed to require two independent errors by virtue of facility design)
- Accumulation of plutonium solids in nonsafe geometry

While an americium/curium mixture may, under optimum conditions, produce an excursion, this possibility is substantially less likely than a plutonium excursion because of the lower reactivity of the mixture.

The criticality fault tree was previously quantified in 3.1.1.8 at 6.4×10^{-6} per cell-year. In the FRP-WTF, there are five primary areas of applicability: the salt waste treatment feed, alpha waste treatment, high level solids, low level extraction feed, and the actinide product. Since the precise cell configurations and utility provisions for each cell have not been designed, it is assumed that each area includes all of the elements of Fault Tree B-1.6. Therefore, the estimated frequency is $5 \times (6.4 \times 10^{-6})$ or 3.2×10^{-5} /plant-year.

As noted previously, this is somewhat lower than previous assessments (Selby, et al.⁽²¹⁾) but consistent with increased regulatory requirements for in-depth system protection by multiple barriers.

The FRP-WTF vessels considered possible candidates are listed in Table 3.2.3. In all cases, fissile material accumulation is assumed to be in excess of the nominal flowsheet values.

The source term for this accident was previously given in Table 3.1.9. The values are based on an event with 1×10^{19} fissions total yield. It is assumed that the energy release is sufficient to vent the vessel to the cell and occurs over an eight-hour period with the excursion initially spiking and then pulsing over this period. It is assumed that the facility is evacuated and that no attempt is made to shutdown the process, but excursion self-terminates either by dispersion or solution evaporation.

3.2.1.3 Radiolytic Hydrogen Explosion

Hydrogen gas is evolved from radiolysis of water in process solutions. The rate of H_2 release is determined by the type and energy of the radiation, and by the solution chemistry. Effects of NO_3^- , Br^- , and Cl^- , all of which lower the rate of H_2 evolution as their concentrations increase, are documented.

Hydrogen generation rates for radiolysis of organic ion exchangers have been extensively documented⁽²²⁾. In the waste treatment facilities, three possibilities are considered:

- Radiolytic hydrogen explosion in main canyon exhaust system
- Radiolytic hydrogen explosion in vessel void space
- Radiolytic hydrogen explosion in a vessel vent filter

A radiolytic hydrogen explosion in the main canyon exhaust system, either within a filter or exhaust blower housing is considered to be incredible. Total H_2 generation rates are ≤ 5500 l/hour for the FRP-WTF and ≤ 1500 l/hour.

Table 3.2.3

FRP-WTF CANDIDATE VESSELS FOR NUCLEAR EXCURSION

Subsystem/Vessel	Plutonium (kgs)	H ⁺ (M)	Event Sequence
<u>Salt Waste Extraction</u>			
TK-24 Alkaline Salt Waste	1.35	---	Fissile material accumulation and precipitation
Concentrator #6 Alkaline Salt Waste	3.70	---	Fissile material accumulation and precipitation
Concentrator #11	2.00	3.5	Fissile material accumulation and precipitation
<u>High Level Solids</u>			
TK-1 Clarified Solution	2.5	4.0	Omission of soluble poison, fissile material accumulation and precipitation
Concentrator #3	5.7	7.0	Omission of soluble poison, fissile material accumulation and precipitation
TK-45 #3 Bottoms Solution	2.8	7.0	Omission of soluble poison, fissile material accumulation and precipitation

Table 3.2.3 (cont'd)

Subsystem/Vessel	Plutonium (kgs)	H ⁺ (M)	Event Sequence
TK-50	2.5	3.7	Fissile material accumulation and precipitation
<u>Low Level Extraction</u>			
TK-1206 Feed Tank	6.0	3.7	Fissile material accumulation and precipitation
<u>An Product</u>			
TK-10 CEC Product Tank	5.5	0.3	Fissile material accumulation and precipitation
TK-28 An Tank	5.0	0.2	Fissile material accumulation and precipitation
Concentrator #7	7.6	3.0	Deformation from safe geometry
TK0958/959 An Product	2.3	3.0	Deformation from safe geometry
<u>Alpha Waste Treatment</u>			
Ash Leacher	0.9	3.3	Omission of soluble poison, accumulation or precipitation of plutonium
HEPA Leacher	0.9	2.2	Same as above
TK-15	1.2	1.56	Same as above

for the FFP-WTF. The natural stack draft would dilute this concentration to below explosive limits within the main canyon exhaust system in the unlikely event of complete loss of exhauster capability (including emergency exhaust).

Both vessel void space and vessel vent offgas explosions are considered to be credible. The vessel void space explosion is represented by Fault Tree B-1.1, which was previously quantified (Section 3.1.1.2) as 3×10^{-4} /plant-year. The accident likelihood is found to be dominated by air compressor failure resulting in insufficient dilution. Candidate vessels in the FRP-WTF are listed in Table 3.2.4. It is to be noted that steam and actinide product concentrators are significant contributors to the total amount of radiolytic hydrogen, but that associated water vapor during normal operations serves to dilute the hydrogen to below the explosion limits in the concentrator offgas system (COG).

VESSEL EXPLOSIONS

Two examples are considered; the HLLW tank (TK-2) and one of the partitioned actinide product tanks (tanks 958/959). In both cases, a stoichiometric mixture of hydrogen and air are assumed to react completely to produce water vapor. Under initial conditions of ambient temperature and pressure, it has been shown for adiabatic explosions in closed vessels that peak temperatures of 2500°C and pressure approaching 8.2 atmospheres occur⁽²³⁾. Vessel distortion is likely to result from a transient pressure in excess of 65 atmospheres⁽²⁴⁾. In either vessel a rupture will most likely occur. Missile fragments may be generated with possible damage to adjacent equipment. The pressure surge may cause the following:

- Blowback of process solution into personnel occupied zones with possible spread of contamination
- Spillage of vessel contents to cell floor with attendant potential self-boiling
- Release of aerosol burst in air of cell
- Momentary pressurization of cell

However, insufficient energy is available to significantly raise a cover block⁽⁴⁰⁾ or to rupture a filter in the VOG system⁽²³⁾. Tables 3.2.5 and 3.2.6 summarize the associated aerosol releases for the two explosions.

Table 3.2.4

FRP-WTF HYDROGEN EXPLOSION CANDIDATE VESSELS

Vessel	Function	w/l	Estimated H ₂ (l/hour)	Time* (Minutes) (≥4%)
Centrifuge #1	Dissolver Solids	428	≥1	≥240
TK-2	HLLW	»2.9		≥300
TK-9	Evaporator #2 Bottoms	10	≈300	≥80
TK-58	CEC Feeds	10	≈400	≥80
TK-10**	Am/Cm Product	34	≈300	≥120
TK's 958/959	Evaporator #7 Bottoms (Partitioned Actinides)	71	≈23	≥30

*based on 50% void space

**excludes H₂ from CEC depressurization

HYDROGEN EXPLOSION IN VESSEL VENT FILTER

A hydrogen explosion in a VOG filter is represented by fault tree B-2.1 in appendix B.

The probability of a Source of Ignition (C1) is conservatively taken to be 1 due to lack of specific details about vessels and filter housings. The probability of Radiolytic Hydrogen being Present (C7) is taken as 1 from a knowledge of the process.

The event, Reduction in Exhauster Capability (G3), could result from Fan Performance Deterioration (C2) from blower corrosion, motor failure, and fan failure. Based on the use of 2 fans (150% capacity), this is assessed as similar to dual compressor failure or $C2 = 2 \times 10^{-4}/\text{yr}$ (reference 5). Loaded Filters Restrict Flow (C3) result from task performance errors (detect and

Table 3.2.5

RELEASES FROM HYDROGEN EXPLOSION IN TK-2*
(curies)

Species	Inventory	Release to Air Tunnel**
I	9.5×10^{-2}	3×10^{-8}
Ru	7.5×10^5	2×10^{-1}
Sr	1.6×10^6	5×10^0
Y	1.6×10^6	5×10^0
Zr	1.1×10^4	3×10^{-2}
Nb	2.5×10^4	8×10^{-2}
Cs	5.5×10^5	2×10^0
Ce	8.5×10^5	3×10^0
Pm	8.5×10^5	3×10^0
Pu	8.0×10^3	2×10^{-3}
Am	1.8×10^4	5×10^{-2}
Cm	1.5×10^6	5×10^0
Totals	7.8×10^6	2.3×10^1

*2.5 MTHM (equivalent) inventory

**RF total = 3×10^{-6} , assumed for tank in small cubicle.

Table 3.2.6

RELEASES FROM HYDROGEN EXPLOSION IN TK-958/959

Species	Inventory (curies)	Release to Air Tunnel (curies)
Pu	3.6×10^4	3.0
Am	7.9×10^3	6.0×10^{-1}
Cm	6.6×10^5	5.4×10^1
Totals	7.0×10^5	57

remedy) or $(10^{-2}) \times (10^{-2}) = 1 \times 10^{-4}/D$ on a minimum basis of 1 per year. Combining C2 and C3 gives $3 \times 10^{-4}/\text{yr}$ for Reduction in Exhauster Capability (G3).

VOG-COG Failure (C4) would occur for the same reasons as G3. There are two independent systems, hence $(3 \times 10^{-4}) \times (3 \times 10^{-4}) = 9 \times 10^{-8}/\text{yr}$ for C4.

The event Sealed System Failure (G4) could result from concurrent controller failure (C5) and task performance error (C6). Controllers in this service may fail open 0.5/yr (reference 10), but failing closed is assumed to occur one-tenth of that, giving $C5 = 5 \times 10^{-2}/\text{yr}$. The single task performance error probability is $10^{-2}/\text{demand}$, as before. This results in Sealed System (G4) failure rate of $5 \times 10^{-4}/\text{yr}$. Combining G3, C4 and G4 gives $8 \times 10^{-4}/\text{yr}$ for Loss of Dilution Air. Since C1 and C7 were previously assessed to be 1, a Hydrogen Explosion (G1) probability is estimated to be $8 \times 10^{-4}/\text{yr}$.

Moeller⁽²⁵⁾ found two cases of filter rupture for 140 plant years. Assuming 30 filters, this corresponds to $4.8 \times 10^{-4}/\text{yr}$. Moeller lists

several examples of H₂ explosions in reactor offgas systems accompanied by release of radioactive material. However, the releases were within operating technical specifications for the facilities involved. For this accident it is assumed that a VOG filter on the actinide product system is ruptured. Specifically, it is assumed that a hydrogen explosion occurs in the second HEPA unit which contains the second and the final HEPA filters, and both are ruptured. The filters are assumed to contain the maximum loading of 25 grams of actinide nitrates. It is further assumed that 10% of the inventory is released from the rupture of the filters. Table 3.2.7 gives the resulting release.

Table 3.2.7
FILTER RELEASE UPON RUPTURE BY HYDROGEN EXPLOSION

Species	Inventory (curies)	Release (curies)
Am	1.8	
Cm	1.3×10^2	1.3×10^2
<hr/> Total	<hr/> 1.3×10^2	<hr/> 1.3×10^2

3.2.1.4 Resin Explosion in Actinide Product Concentrator

A nitrated organic explosion in the actinide product concentrator could result from the addition of resin, resin fines, or resin degradation products into the concentrator from the CEC units. The other essential conditions, nitric acid/heavy metal nitrates and initial temperature are indicated in Fault Tree B-2.2. The possibility that the Temperature Exceeds the Reaction Threshold (C1) is taken as 1 because the normal bottoms temperature is 100°C and 75°C is a threshold temperature limit for resin reactions ⁽²²⁾. The probability of Nitric Acid/Heavy Metals being present (C2) is also taken to be 1 by the nature of the process as well as Organic Materials being present (G2)

because of introduction of resin, resin fines or solidified resin. Under high radiation, resin will deteriorate and dissolve⁽²²⁾.

The probability of the presence of Other Organics (C3) is estimated as ϵ because of the process. The presence of solubilized resin (C5) in sufficient quantity to result in nitrated organic is estimated to be ϵ by virtue of the decanting of the concentrator feed and concurrent steam stripping of the feed in the concentrator tower.

Sequential Metal Filter Failures (C4) system failure are estimated to have a probability of 0.2/unit-yr and inline filter failure as 0.02 per year for a combined failure probability of 4×10^{-3} /unit-yr. However, the inline filter is checked and periodically replaced so a task performance error probability of 10^{-2} /demand must occur independently giving a final probability estimate of 4×10^{-5} /unit-yr for Resin/Degraded Resin in Feed (G3). The Nitrated Resin Explosion (G1) probability is 4×10^{-5} /unit-yr.

There are 50 online units in the FRP-WTF, hence a combined value of 2×10^{-3} /yr is found.

About 4 liters (or roughly 8 kilograms) of resin could be introduced into the product concentrator. For purposes of accident analysis, it is assumed that this quantity would result in an explosion of sufficient magnitude to rupture the vessel. Upon rupture of the vessel at its base, the entire vessel inventory is released to the cell floor where the solution continues to boil until quenched by water dilution from deliberate intervention or from the fire-fog system. The initial release would be a nominal 100 milligrams of liquid aerosol per cubic meter of volume and the releases to the air tunnel are given in Table 3.2.8.

Table 3.2.8

RELEASES TO AIR TUNNEL ATTENDANT ON EXPLOSION/
RUPTURE OF PRODUCT CONCENTRATOR
(Curies)

<u>Species</u>	<u>Concentrator</u>	
	<u>Inventory</u>	<u>Release</u>
Pu	1.2×10^5	1.5
Am	2.6×10^4	0.3
Cm	2.2×10^6	30
Total	2.3×10^6	30

Table 3.2.9

FRP-WTF CONCENTRATORS AND ASSOCIATED INVENTORIES

Vessel	Concentrator Purpose	Inventory (liters)	Decay Heat (w/L)	Solution Acidity (moles H ⁺)	U (kgs)	Pu (kgs)	Np (kgs)	Am (kgs)	Cm (kgs)	Fission Products (kgs)	Comments
#1	Ln/An Concentrator (U strip[mp])	8000	3.4 ⁻²	0.3	28	---	---	---	---	88 Ln 1.8 Ru	Red oil - loss of formic and inversion
#2	Ln/An Concentrator (Ln/An strip)	8000	10	0.5	15	0.4	1.4	16	24	76 Ln 0.2 Ru	Red oil - loss of formic and inversion H ₂ - shutdown Criticality - ppt. Ru - loss of reductant and high nitric
#3	Dissolver Wastes Solution	6000	6.4	7.0	90	5.7	1	11	15	---	Uncomplexed F ⁻ ceric ion H ₂ - shutdown
#4	Column 11 U Strip	8000	---	0.3	0.6	---	---	---	---	---	Red oil - loss of formic and inversion
#5	Column 10 Ln/An Strip	8000	7.1	4.0	0.8	<0.1	0.6	16	24	---	H ₂ - shutdown/criticality
#6	Alkaline Salt Waste	8000	0.24	---	120	3.4	0.3	0.5	0.6	---	Criticality Red oil
#7	Actinide Product	100	70.8	3.0	169	7.6	2.6	20	28	---	Red oil solvent resin H ₂ shutdown LOCA Criticality
#8	Acid Recovery	8000	---	7.0	---	---	---	---	---	---	
#9	Recycle Nitric (CEC)	6000	---	6.0	---	---	---	---	---	---	
#10	CEC-FP + Gd	12000	3.9	6.0	---	---	---	---	---	170 kg Ln ~33 kg Nm (14 kg Ru)	
#11	Acid Salt Waste Feed	8000	3.2 ⁻²	3.5	60	18	5	<0.1	<0.1	6 Ru	
#12	Salt Waste CMP Extraction Column Raffinate	8000	---	7.0	---	---	---	---	---	---	
#13	Concentrator 10 Overheads	12000	---	4	---	---	---	---	---	---	
#14	Overheads Reboiler (1, 6, & 11)	---	---	1.0	---	---	---	---	---	---	
#15	Overheads Reboiler (product concen- trator overheads)	2000	---	1.0	---	---	---	---	---	---	
#16	Acid recovery	8000	---	4.0	---	---	---	---	---	---	
TK-1505	Sump Recovery	3000		≥4.0							

3.2.1.5 "Red Oil" Explosions in WTF Concentrators

Violent exothermic reactions of organics reacting in high concentrations of nitric acid have been noted both with and without the presence of heavy metals⁽¹⁴⁾. This decomposition of organo-phosphate-nitrate complexes is conventionally referred to as "red oil." It is generally agreed that for such a reaction to occur, a mixture of organic and acid must be heated above an initiation temperature and that rapidity of the temperature change enhances the violence of the reaction. Chief design barriers to preventing a "red oil" type reaction include:

- Prevention of organic introduction into concentrators by steam stripping and pump decanting the feed to the concentrator
- Top overflow of concentrate to preclude accumulation of a light organic phase in the concentrator
- Temperature controls and temperature-steam supply interlocks to preclude exceeding bottoms temperature limits (125°C for nitric acid-heavy metal solutions and 135°C for nitric acid concentrators, Reference 14)

The three FRP-WTF solvents are all capable of violent reactions with nitric acid. The FRP-WTF concentrators and associated inventories are listed in Table 3.2.9.

Fault Tree B-1.3 relates the various failures producing a "red oil" explosion. This tree has been previously quantified (Section 3.1) so that only those portions that differ between the concentrators in Table 3.2.8 will be discussed.

Explosion Initiators Present (G3) has been previously estimated at 2×10^{-4} per year. This is common to all the concentrators, the self-heating of the process solutions having been shown to be negligible via Gate G9. Solvent in Feed (G2) and Heavy Metals/Nitric Present (C19) and Frequency of Explosion are evaluated in Table 3.2.12.

The product concentrators previously covered in Section 3.2.1.4 are omitted here, as a resin reaction has a greater likelihood than a "red oil" explosion (as can be seen by comparing accident frequencies).

A release fraction of 1×10^{-5} based on a cell volume of 800-1000 m³ is used as an estimate release fraction, and the air tunnel releases for each vessel are calculated based on the above release factor. No credit is taken for the wire mesh pad used as a roughing filter for the flow channel venting the cell to the air tunnel. The air tunnel releases are tabulated in Table 3.2.11.

3.2.1.6 Resin Fire and/or Explosion

Cation exchange chromatography is used in separating actinides and lanthanides. Due to the curium load for each batch, the resin may only be used for one run and then must be discarded due to severe deterioration as a result of radiation. The CEC operations have a number of problems requiring careful detailed design, not the least of which is the requirement to operate remotely under at 1000 psig pressure.

Should process flow be lost, the resin will rapidly decompose, initially at the curium band, then autocatalytically throughout the bed. Provision of pressure relief, rupture discs and emergency dump tanks also presents difficulties in that any reduction in pressure on the resin results in immediate release of dissolved hydrogen. This phenomenon, in conjunction with possibly burning resin, could result in a vapor explosion in the dump tank, or in the cell sump.

For purposes of assessing the overall risk of operating the FRP-WTF, it is assumed that these problems are resolved or obviated by design, equipment selection, and administrative controls such that the likelihood of uncontrolled resin reactions resulting in a vessel rupture and attendant release is no greater than for comparable, conventional radiochemical operations. Safety provisions that are assumed include:

- Failsafe interlocks to preclude putting a second run through a rack prior to resin changeout
- Safe venting via pressure relief devices on all vessels
- Provision of shock attenuation in the ventilation exhaust duct servicing the CEC units
- Two independent pressure relief systems: an emergency dump and a pressure relief to the cell

Table 3.2.10

FAULT TREE QUANTIFICATION

FRP-WTF Concentration Type*	FRP-WTF Concentration	Solvent in Feed (G2)	Heavy Metal/ Nitric Present (C19)	"Red Oil" Explosion (G1)
A	1, 2, 4, 5	1×10^{-1}	1×10^{-2}	2×10^{-7}
B	6	--	--	--
C	12	2×10^{-2} /day	1	4×10^{-6}
D	11	--	1	2×10^{-4}
E		--	1×10^{-2}	4×10^{-8}
F	3, 9, 13, 14, 15, 16	--	--	--
G	TK-1505	1	1	2×10^{-4}

*The rationale for the vessel types is as follows:

A Concentrators

- Solvent in feed is determined by likelihood of CMP phase conversion due to oversaturation
- Nitric acid concentration normally low; loss of formic acid (task performance error) most likely mechanism

B Concentrators

- Bottoms solution always alkaline

C Concentrators

- Solvent introduction governed by likelihood of misvalving or column flood transferring 2-EHOH with associated pump down of feed tank
- Heavy metals/nitric acid always present

D Concentrators

- Solvent transfer is normal due to presence of solvent in wastes
- Heavy metals/nitric acid are present by virtue of estimated waste characteristics

E Concentrators

- Reaction dependent on failure to denitrate

F Concentrators: No solvent present

G Sump Collection Tanks

- Solvent transfer normal
- Heavy metals/nitric acid are normally present

Table 3.2.11

AIR TUNNEL RELEASE: RED OIL EXPLOSION IN
FRP-WTF CONCENTRATORS
(curies)

Species	Concentrator			
	#2	#5	#11	TK-1505*
Pu	6×10^{-2}	---	3	8×10^{-3}
Am	0.2	0.2	---	4×10^{-3}
Cm	20	20	---	0.3
Totals	20	20	30	0.3

*Assume 1×10^{-8} fraction of facility inventory

For purposes of analysis a resin column rupture and resultant resin fire in the cell sump are analyzed. A resin explosion in the emergency dump tank would be significantly less serious and less likely by virtue of vessel configuration, low pressure and the large inventory of liquid which serves as a heat sink.

The fault tree presented in Appendix B, Figure B-2.3, is quantified as follows:

Dry Resin Ignition (G5) is controlled by resin drying. Dry Resin Accumulates (C10) is ϵ for an operating system. Resin Bed Drained (C9) is taken to be the result of two independent task performance errors or 10^{-4} /demand. Impact Ignites Resin (C11) and Spark Ignites Resin (C12) are also ϵ for an enclosed resin bed without a spark source inside the column. Self-Heating Ignites Resin (C26) is taken as 1 by virtue of the high thermal watt density at the curium band. Combining these probabilities gives an estimate of 1×10^{-4} /demand for the event Dry Resin Ignites (G5). The number of runs per year (G5) is $\approx 10^4$ per year. Stagnant Resin Autocatalytic Decomposition (G6) requires that the bed be stagnant for 36 hours (C6) in ≥ 7 molar nitric acid (C7). Each operation consists of a single task performance error for a combined 10^{-4} /demand or 1/year value for G6.

Introduction of a Strong Oxidant (C8) is taken as ϵ by virtue of controlled aqueous makeup and cold chemical additions to columns.

Hot Spot in Bed (G10) is taken as 1 because of the heat in the curium band.

Loss of Bed Cooling (G11) may result from Loss of Solution Flow (G12) or Cooling System Failure (G13). Loss of flow due to gas bubble (C17) is arbitrarily taken at 0.1/yr.

Aged Resin (C24) is taken as ϵ due to resin changeout after each run. Loss of Column Pressure (C25) requires loss of operating pressure. This would come about through pressure relief/rupture disc failure at 9×10^{-3} /yr⁽¹⁶⁾.

Mechanical Obstruction (C18) is assessed to be ϵ because the columns are emptied and refilled with resin daily.

High Bed Differential Pressure (G14) then becomes $G14 = C17 + C24 + C25 + C28 = 0.11/\text{yr}$.

Pump Failure (C19) is assessed 0.22/yr⁽¹¹⁾. Plugged Inlet (C20) is estimated to occur 0.1/yr. A Line Leak (C21) is assumed to result from a task performance error during yearly installation or 10^{-2} /yr. Loss of Flow (G15) is the sum of these or 0.33/yr.

In order for Bed Cooling to Fail (G13), a redundant Temperature Control Failure (C16) must occur at $2 \times 10^{-4}/D$ ⁽¹⁰⁾, or Loss of Utility Service (C23) occurs (including loss of emergency supply, backup pumping capability and high tank reservoir). C23 is assessed as ϵ . G13 is the combined value or $2 \times 10^{-4}/D$ or 2/yr for 33 runs per day.

Resin Autocatalytically Decomposes During Operation (G7) may then be calculated from the sum of the forgoing, or $G7 = G10 + C8 + G11 = 3.44/\text{yr}$.

Violent Resin Degradation (G2) may now be estimated as $G2 = G5 + G6 + G7 = 5.44/\text{yr}$.

Pressure Relief Valve Failure (C1) is assessed at $9 \times 10^{-3}/\text{yr}$ (reference 16). None Provided (C2) is assessed at ϵ based on daily inspection. Improper Installation (C3) is assumed to result from 2 independent task performance errors (installation error and failure to detect during inspection). At an assumed annual frequency for replacement, this is $10^{-4}/\text{yr}$. Pressure Relief LTA (G3) is therefore the sum or $9 \times 10^{-3}/\text{yr}$.

Resin Dump Inseparable (G4) is determined by Mechanical Failure (C4) to be $9 \times 10^{-3}/\text{yr}$ (reference 16). Personnel Intervention (C5) in this case would not be essential so is assessed at ϵ .

Resin Explosion (G1) may therefore be estimated from the product of Violent Resin Degradation (G2), Pressure Relief LTA (G3) and Resin Emergency Dump Inseparable (G4). $G1 = (5.44) (9 \times 10^{-3}) (9 \times 10^{-3}) = 4.4 \times 10^{-4}/\text{yr}$.

This suggests that on about 5.4 occasions per year violent resin degradation could occur, but a consequential serious fire or explosion is significantly less likely by virtue of two independent pressure relief systems.

For purposes of analysis, it is assumed that the explosion/vessel rupture occurs at the curium band resulting in dispersal of burning resin within the CEC cubicle but the fire extinguishers actuate thereby quenching the fire.

The estimated releases to the air tunnel are calculated assuming 10% resin combustion. Based on a combustion soot loading of 3 g/m^3 (reference 26), a release factor of 0.1 is reasonable for the cell size. The releases given in Table 3.2.12 are based on these considerations.

Table 3.2.12
AIR TUNNEL RELEASES FROM RESIN EXPLOSION

Species	FRP-WTF (curies)
Am	78
Cm	5.5×10^3
Total	5.5×10^3

3.2.1.7 Facility Fire

The Design Basis Fire as defined in Reference 27 is one that results from the burning of all flammable or combustible materials within an area enclosed by a fire resistant barrier of at least 2 hour rating. In the FRP-WTF, major inventories of flammable and combustible materials include process solvents, ion exchange resin, and contaminated solid wastes from in-cell activities. The resin fire is incorporated as part of the CEC resin explosion (3.2.1.6). Combustion of contaminated solid wastes, while a serious incident within the plant, would have minimal out-of-plant impact because the inventory of waste stored would be administratively controlled to minimize the likelihood and effects of such a fire. Furthermore, the activity released would be low and would be associated with soot and smoke particles, and hence, easily removed by the exhaust gas treatment equipment.

In the FRP-WTF, three solvents are used: CMP diluted with di-isopropyl benzene, TBP diluted with kerosene, and 2EHOH. All are moderate fire hazards with 2EHOH possibly representing the greater hazard by virtue of its lower flash point (128°F). Three accidents may occur: Ignition of a leak with attendant fire on the floor of the cell, loss of the contents of a vessel followed by ignition and combustion on the cell floor, and a vapor explosion associated with solvent mist. Of these, the ignition and combustion of solvent leakage from an extraction column presents the greatest potential for a significant release of radionuclides, while the ignition of a solvent mist offers the greatest potential for rapid energy release. Table 3.2.13 lists the columns in the FRP-WTF which contain significant quantities of solvent and radionuclides. For purposes of analysis, it is assumed that CMP and TBP are roughly equivalent in flammability.

SOLVENT FIRE

The fault tree for a solvent fire accident is presented as Figure B-1.2 in Appendix B. This was previously quantified for an assessed frequency of 2×10^{-4} /year (3.1.1.3).

Given that a leak and fire occur, possible outcomes range from immediate extinguishment to almost complete combustion of available solvent⁽²⁸⁾. The rate of combustion will be oxygen-limited given the cell size and ventilation flow. The degree of combustion will also be diminished by limited oxygen (3.1.1.3). Given the above, approximately 275 liters of solvent could be burned. From the information in Table 3.2.12 and an assumed release factor of 10% (forced-draft fire, Reference 25), the release quantities in Table 3.2.14 are obtained.

SOLVENT MIST EXPLOSION

The fault tree for mechanical formation of solvent mist is given in Figure B-2.4 in Appendix B which is transferred to Figure B-1.2 (Appendix B) for the solvent fire analysis, and quantified as follows:

Loss of Compressor (C7) is taken to be 2×10^{-3} /year (Section 3.1.1.2, Reference 6). Poppet Assembly Failure (C9) is assumed to be the same as an automatic valve failure of 1×10^{-2} /year⁽¹⁶⁾. Surge Tank Capacity (C8) lost would result from pressure relief valve failure which would be 9×10^{-3} /year⁽¹⁶⁾. Therefore Loss of Air Pressure (G7) is 2×10^{-2} /year.

Table 3.2.13
FRP-WTF PROCESS COLUMNS: SOURCE TERMS
FOR SOLVENT FIRE

Vessel	Function	Organic	Organic Vol. (liter)	U (kg)	Pu (kg)	Np (kg)	Am (kg)	Cm (kg)
Col #1	HLLW Extraction	CMP	1,000	0.65	0.02	0.6	0.5	0.7
Col #2	Actinide Strip	CMP	1,000	0.65	0.02	0.6	0.5	0.7
Col #7	Actinide Strip	TBP	1,000	36	1.6	0.7	0.6	0.8
Col #8	Actinide Retraction	TBP	1,000	36	1.6	0.7	0.6	0.8
RV 4	Salt Waste Extraction	2EHOH	12,000	trace	trace	trace	trace	trace
Col #15	Salt Waste Extraction	2EHOH	1,000	trace	trace	trace	trace	trace
Col #19	Salt Waste Extraction	2EHOH	1,000	trace	trace	trace	trace	trace

Table 3.2.14

RELEASE TERMS FOR SOLVENT FIRES
(Curies)

Species	Col. #1	Col. # 7
Pu	9	704
Am	18	21
Cm	1.5×10^3	1.7×10^3
Total	1.5×10^3	2.4×10^3

Sono Probe Failure (C6) would correspond to general instrument failures, though how it would fail unsafe is not readily apparent. Based on Reference a value of 4×10^{-5} /yr is assumed. Therefore, Solution Backup Into Pulser Leg (G6) = $(4 \times 10^{-5}) (2 \times 10^{-2})$ or 8×10^{-7} /year.

Startup (C5) means that normal back flow prevention systems are bypassed. This is assumed to be 0.25 based on a 75% plant availability.

Contaminated Solution in Gallery (G5) is therefore controlled by Startup, or $0.25 + 8 \times 10^{-7} = 0.25$ /year.

Transfer goes to Fault Tree B-1.2, Solvent Fire, Mechanical Formation of Solvent Mist (C1).

Release via a Pulser Leg (C15) on B-1.2 is given by a pipe weld failure (all welded pipe is assumed in a contact maintained cell) and is assumed at 0.04/year⁽¹⁵⁾.

Combining Mechanical Solvent Mist (C1) and Transfer Leak (C16) gives a value of $0.25 + 0.04$ or 0.29/year for Release To The Cell (G2).

Given the foregoing, Fault Tree B-1.2, Solvent Fire Explosion, may be requantified as follows:

Failure of Solvent Boundary (G4) is therefore 0.29/year

Adequate Oxygen (C13) as before (3.1.1.3) is 1

Ignition (G5) as before (3.1.1.3) is 0.1

Therefore, Solvent Combustion (G2) is estimated to be $(0.29)(0.1)(1) = 0.29$ /year.

Blocked Floor Drain (C8) as before (3.1.1.3) is 0.1/challenge

Failure of Personnel Intervention (C14) as before (3.1.1.3) is 1

Failure of Fire Extinguishment (G3) as before (3.1.1.3) is 0.24/challenge

and combining, $(0.029) \times (0.1) \times (0.24) \times (1) = 7 \times 10^{-4}/\text{year}$.

The source term for this accident is assumed to be the quantity of mist and aerosol in the cell. The quantity of mist and aerosol corresponds to 100 mg/m^3 of aerosol as a 5 volume % mist. For a cell volume of 821 m^3 , 900 grams or approximately 1 liter of solvent is in the aerosol. Using a release fraction of 100% for the explosion, the resultant release to the cell exhaust is given in Table 3.2.15.

Table 3.2.15
RELEASES FROM SOLVENT MIST EXPLOSION
(Curies)

Species	Col. # 1	Col. # 7
Pu	0.3	26
Am	0.6	0.8
Cm	55	63
Total	56	90

3.2.1.8 Radiological Public Accident Risk

Tables 3.2.1.16 - 3.2.18 summarize the probabilities, consequences and the risks of the preceding accident analyses. These three tables, respectively, analyze the effects with none, one and two HEPA failures based on HEPA failure rates presented in Appendix A.6. No consideration was given to a deep bed HEPA failure. It will be noted that the criticality dominates the risk.

Table 3.2.16
SUMMARY OF FRP-WTF ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(No Failed HEPA Filters)

Accident Section	Description	Probability (/Plant-Yr)	Curie Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.4.1.2	Criticality in Process Cell	3×10^{-5}	9×10^4	2	2×10^{-4}	6×10^{-5}	8×10^{-9}
3.4.1.3	Hydrogen Explosion in Actinide Bearing Vessel	a. 3×10^{-4}	1×10^{-8}	4×10^{-8}	3×10^{-10}	1×10^{-11}	9×10^{-14}
		b. 3×10^{-4}	3×10^{-8}	9×10^{-9}	2×10^{-10}	3×10^{-12}	6×10^{-14}
	Hydrogen Explosion in Vessel Vent System	8×10^{-4}	8×10^{-8}	2×10^{-8}	4×10^{-10}	2×10^{-11}	3×10^{-13}
3.4.1.4	Nitrated Organic Explosion in Am Product Concentrator	2×10^{-3}	2×10^{-8}	5×10^{-9}	1×10^{-10}	1×10^{-11}	2×10^{-13}
3.4.1.5	"Red Oil" Explosion in Process Concentrator	2×10^{-4}	3×10^{-8}	9×10^{-9}	1.8×10^{-10}	1.8×10^{-12}	3×10^{-14}
3.4.1.6	Resin Explosion in CEC Unit	4×10^{-4}	3×10^{-6}	6×10^{-7}	2×10^{-8}	2×10^{-10}	8×10^{-12}
3.4.1.7	Facility Solvent Fire	2×10^{-4}	2×10^{-6}	1.6×10^{-6}	3×10^{-8}	3×10^{-10}	6×10^{-12}
	Solvent Mist Explosion	7×10^{-4}	8×10^{-8}	9×10^{-8}	1.4×10^{-9}	6×10^{-12}	1×10^{-12}
Total Risk						6×10^{-5}	8×10^{-9}

Table 3.2.17

SUMMARY OF FRP-WTF ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(1 Failed HEPA Filter)

Accident Section	Description	Probability (/Plant-Yr)	Curie Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.4.1.2	Criticality in Process Cell	3×10^{-7}	9×10^4	2	2×10^{-4}	6×10^{-7}	8×10^{-11}
3.4.1.3	Hydrogen Explosion in Actinide Bearing Vessel	3×10^{-6}	1×10^{-4}	4×10^{-4}	3×10^{-7}	1×10^{-9}	9×10^{-12}
		3×10^{-6}	3×10^{-4}	9×10^{-5}	2×10^{-6}	3×10^{-10}	6×10^{-12}
	Hydrogen Explosion in Vessel Vent System	8×10^{-6}	8×10^{-4}	2×10^{-4}	4×10^{-6}	2×10^{-9}	3×10^{-11}
3.4.1.4	Nitrated Organic Explosion in Am Product Concentrator	2×10^{-5}	2×10^{-4}	5×10^{-5}	1×10^{-6}	1×10^{-9}	2×10^{-11}
3.4.1.5	"Red Oil" Explosion in Process Concentrator	2×10^{-6}	3×10^{-4}	9×10^{-5}	1.8×10^{-6}	1.8×10^{-10}	3×10^{-12}
3.4.1.6	Resin Explosion in CEC Unit	4×10^{-6}	3×10^{-2}	6×10^{-3}	2×10^{-4}	2×10^{-9}	8×10^{-11}
3.4.1.7	Facility Solvent Fire	2×10^{-6}	2×10^{-2}	1.6×10^{-2}	2.5×10^{-4}	3×10^{-8}	5×10^{-10}
	Solvent Mist Explosion	7×10^{-6}	5×10^{-4}	6×10^{-4}	1×10^{-5}	4×10^{-9}	7×10^{-11}
Total Risk						6.4×10^{-7}	8×10^{-10}

Table 3.2.18
SUMMARY OF FRP-WTF ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(2 Failed HEPA Filters)

Accident Section	Description	Probability (/Plant-Yr)	Curie Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.4.1.2	Criticality in Process Cell	9×10^{-12}	9×10^4	2	2×10^{-4}	2×10^{-11}	2×10^{-15}
3.4.1.3	Hydrogen Explosion in Actinide Bearing Vessel	9×10^{-11}	2.3	6	5×10^{-2}	5×10^{-10}	5×10^{-12}
		9×10^{-11}	5.7	1.5	3×10^{-2}	1×10^{-10}	2×10^{-12}
	Hydrogen Explosion in Vessel Vent System	2×10^{-10}	13	2.9	6×10^{-2}	6×10^{-10}	1×10^{-11}
3.4.1.4	Nitrated Organic Explosion in Am Product Concentrator	6×10^{-10}	3	9×10^{-1}	1.6×10^{-2}	5×10^{-10}	1×10^{-11}
3.4.1.5	"Red Oil" Explosion in Process Concentrator	6×10^{-11}	4	1	2×10^{-2}	6×10^{-11}	1×10^{-12}
3.4.1.6	Resin Explosion in OEC Unit	1×10^{-10}	5.5×10^2	1×10^2	3	1×10^{-8}	3×10^{-10}
3.4.1.7	Facility Solvent Fire	6×10^{-11}	2.4×10^2	3×10^2	8	1.8×10^{-8}	5×10^{-10}
	Solvent Mist Explosion	2×10^{-10}	9	11	0.2	2×10^{-9}	3×10^{-11}
Total Risk						3.2×10^{-8}	8.5×10^{-10}

The summarized risk is 6×10^{-5} person-rem/plant-year and 9.9×10^{-9} latent cancers/plant-year. In terms of power equivalence, this is 8×10^{-7} person-rem/GWe-year and 1.3×10^{-10} latent cancers/GWe-year.

3.2.2 Public Atmospheric Risk of FRP-WTF

Consequences to the public from routine radioactive effluents from the FRP-WTF are developed in Appendix G. Routine releases can be expected to result in a population dose of 5.3 person-rem/yr or 0.24 latent cancer/yr in terms of health effects. On an energy equivalence basis, this corresponds to 7×10^{-2} person-rem/GWe-year or 3.2×10^{-3} latent cancer/GWe-year.

3.2.3 Occupational Radiological Risk of FRP-WTF

The occupational exposures for the FRP-WTF were estimated on a basis comparable to that used in assuming the FRP occupational dose since the facilities are similar in complexity of operations, levels of activity, shielding design criteria (500 millirem per year), and maintenance philosophy. The calculations and assumptions made to arrive at the values in Table 3.2.19 are given in Appendix D. The summary of these values is 560 person-rem/plant-year. Using the BEIR report equivalence estimate of 1×10^{-4} fatalities/person-rem, this is 7.5 person-rem/GWe-year and 7.6×10^{-4} fatalities/GWe-year.

3.2.4 Nonradiological Risks of the FRP-WTF

The chemical and industrial risks developed in Appendix E are summarized in Table 3.2.20. The primary chemical hazards are associated with the fraction of site power plant effluents attributable to utility consumption by the FRP-WTF. The health effects associated with these effluents completely dominate the nonradiological risk.

Table 3.2.19

ESTIMATED OCCUPATIONAL EXPOSURE
FRP-WTF

Activity	Person-Rem/Year	Person-Rem/GWe-Yr	Fatalities** GWe-Year
Routine Operations	220	3	3×10^{-4}
Maintenance	220	3	3×10^{-4}
Abnormal Occurrences*	10	1×10^{-1}	1×10^{-5}
Accidents	2×10^{-2}	3×10^{-4}	3×10^{-8}
Decommissioning	83	1.1	1.1×10^{-4}
Total		7.1	7.1×10^{-4}

*Defined as personnel exposure exceeding limits set in 10CFR20

**From the BEIR report equivalence estimate of 1×10^{-4} fatalities per person-rem

Table 3.2.20

SUMMARY EFFECTS
NONRADIOLOGICAL CAUSES

Cause	Premature Deaths (Fatality/GWe-Yr)	Economic Deaths (Fatality/GWe-Yr)	Total (Fatality/GWe-Yr)
Effluents	0.09	0.28	0.37
Industrial Accidents	5.1×10^{-4}	2.3×10^{-2}	2.3×10^{-2}
Total			0.39

3.3 RISK OF THE FUEL FABRICATION PLANT (FFP)

3.3.1 Reference Plant Probabilities and Consequences

3.3.1.1 Overview and Preliminary Hazards Analysis

Appendix A.3 contains an overview discussion of the plant and the material quantities and flows. This plant is designed for fabricating Reference or PT fuel for the 5th recycle and differs from current mixed oxide (MOX) fabrication plant designs by the larger amount neutron shielding and by the use of remote processes and maintenance. It is also significantly larger (660 MTHM/year) than the Westinghouse Recycle Fuel Plants (RFP) (200 MTHM/year⁽²⁹⁾) which is the largest current design. The RFP was designed to accept separated uranium and plutonium as feedstock; the Reference FFP is designed for use with coprocessed fuel and, as such, accepts a uranium-actinide powder blend. Care must be taken in wet scrap recovery that column upsets do not result in plutonium separation from the other actinides thus defeating the coprocessing mode.

Because, with exception of wet scrap recovery, there are no operations that can change the chemical ratios in the FFP product, it is convenient to prepare a table of the isotopic curie quantities per kilogram of product and to analyze the accident consequences in terms of mass of material released. This is presented in Table 3.3.1.

While there are differences between the FFP and the Anderson plant used as a base for the Erdmann, et al.⁽⁵⁾ analysis, it is convenient to use their work as a point of departure.

Table 3.3.2 presents a preliminary hazards analysis (PHA) for the FFP plant. The purpose of the PHA is to aim a completeness without special consideration to accident significance. Through preliminary scoping accident calculations, the accidents presenting the greatest public risk are selected for detailed, fault tree analysis and accident consequence evaluations.

Table 3.3.1

SIGNIFICANT RADIOLOGICAL ISOTOPIC RISK
CONTRIBUTORS IN THE FFP PRODUCT
(curies/kg of powder blend)

Isotope	Reference Product	PT Product
^{239}Np	--	0.55
^{238}Pu	90.	76
^{239}Pu	1.5	1.60
^{240}Pu	3.8	4.1
^{241}Pu	980.	950.
^{241}Am	1.6	5.3
^{242}Am	--	0.21
^{242}Cm	--	90.
^{244}Cm	--	280.
Total Curies	1,078	1,408

Table 3.3.2
FFP PRELIMINARY HAZARD ANALYSIS

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Receiving/Powder Transfer Area	Container Leaking	Material Failure/ Fatigue	HEPA Filters Inward Air Flow	<8 μ m AED Powder	Highly Improbable - losses limited to 40 kg at most.
	Container Puncture During Receiving	Fork Truck Used for Maintenance			
	Container Puncture During Powder Transfer	Improper Indexing			
	Fire	Unauthorized Introduc- tion of Flammables Into Transfer Area			
	Reverse Air Flow Due to Tornado	Tornado Strike	Area Closure Doors Reverse Air Flow Dampers Design for Design Basis Tornado	Estimate 1 kg of Powder Accumulation on Surfaces	Highly Improbable - small source term.
	Severe Equipment Disruption	Earthquake	Area Closure Doors Reverse Air Flow Dampers Design for Design Basis Earthquake	Quantity for Release Could be Large (3 MTHM) but Lack of Dispersive Forces	Highly Improbable - small source term.
Pneumatic Powder Transfer	Failure of Tubing or Connectors	Wear and Usage	Vacuum Transfer Causes Inward Flow HEPA Filters	Surface Contamination	Very Low
	Failure of Exhaust Filters	Blockage, Binder Hardening	Quality Control HEPA Filters	Could Result in Disper- sion of Material Until Shutoff - estimate 10 kg	Fairly Probable
Storage Hoppers	Oxidation of UO_2	Powder Overheating	HEPA Filters Temperature Control Limited Oxygen Supply if no Pneumatic Trans- fer in Use	500 kg of 8 μ m AED Powder but Little Dis- persal Force - Possibly Air Entrainment	Possible but small release outside of plant.
	Disruption of Building and Storage Hopper	Air Crash	Building is Designed for DBE Earthquake so Difficult to Penetrate with Missiles or to Cause Collapse		Very unlikely but con- siderable outside release. Fir could accompany air crash to provide dispersion. Tornado winds could cause dispersion. No unusual winds.
		Tornado			
		Earthquake			

Table 3.3.2 (cont'd)

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Storage Hoppers	Criticality	Flooding	Siting for Design Basis Flood Strict Control Over Water in Plant 3% Equivalent Enriched Powder is Difficult to Go Critical HEPA Filters	Agglomerated Ceramic Powder Fission Gases Neutrons	Very low to the public.
Blending					
Blending	Pneumatic Transfer Failure	Filter, Failure	Quality Control Maintenance HEPA Filters	8 AED Powder 1 KG Dispersed Before Shutoff	Slight
Slugging/Pressing	Hydraulic Fluid Fire	Seal Failure, Improper Fluid, Excessive Temp.	HEPA Filters Little Dispersion Force	8 AED Powder 1 Pellet Amount	Very Slight
Fill Containers	Explosion	Residual Solvent	Administrative Control Over Solvents	8 AED Powder 1 kg Dispersed Inside of Cell	Slight
Hopper Loading and Storage	Failure of Transfer Line	Powder-Caused Abrasion Pipe Stress or Vibration	RAA, LAA Walls, 2 Stages of HEPA Filters Transfer Cutoff on Failure Detection	Transfer-In: In Batches of 2.2 kg Transfer-Out: Until Closure of Isolation Valve	Modest
	Collapse of Pu Hopper Supports or Hopper Container Body	Design or Fatigue	Quality Control Over Design	170 kg of Ceramic Grade Pu in One of 3 Vessels	
Blending and Storage	Pneumatic System Failure	Wear, Abrasion	3 HEPAs	Ceramic Powder	Very Low
Pressing	Hydraulic Fluid Fire	Seal Failure Improper Fluid	3 HEPAs	Coarse Powder	Very Low
Sintering	Hydrogen Explosion in the Sintering Furnace	Improper Oxygen Explo- sion Mixture Valve Failure	Same	Coarse Powder	Very Low
	Dust Spill	Mechanical Failure	Same	Coarse Powder	Very Low
	Criticality	Water-Cooled Furnace Moderation Extremely Difficult for 3-4% Enrichment to go Critical	Same	Neutrons, Iodine, Gamma Rays, and Nobles	Very Low
	Release During Maintenance	Improper Procedures	Same	Coarse	Very Low
Grinding	Abnormal Grinder Operations	Loss of Coolant	3 HEPAs	Dust-Like Powder	Very Low

Table 3.3.2 (cont'd)

SYSTEM/ SUBSYSTEM	POTENTIAL ACCIDENTS	CAUSAL FACTORS	BARRIERS	CHARACTERISTIC SOURCE TERMS	SIGNIFICANCE
Several	Solvent Fire	Procedural Error	Same	Coarse	Very Low
Rod Loading or Repair	Dropped Pellets	Error	Same	Coarse	Very Low
Improper Welding	Dispersed MOX	Error	Same	Dust-Like Powder	Very Low
Fuel Shipping	Dropped Rod	Error	1 HEPA	Coarse	Very Low
Clean Scrap Recovery	Criticality	Procedural Errors	3 HEPAs	Neutrons, Iodine, Nobles and Gamma	Very Low
	Hydrogen Explosion in ROR Reactors	Oxygen Release in Reaction Improper Ventilation	3 HEPAs	Coarse	Very Low
	Spill of MOX Scrap	Error	3 HEPAs	Coarse	Very Low
Dirty Scrap Recovery	Solvent Fire	Leak, Operating Errors	3 HEPAs	Coarse	Very Low
	Red Oil Explosion	Loss of Temperature Control	3 HEPAs	Wet Process	Low
	Resin Fire and Explosion	Administrative Control Nitration of Filter	3 HEPAs	Wet Process	Low
Analytical Service Facility	Solvent Fire in Glove Box	Procedural Error	3 HEPAs	Wet Process	Very Low
Final Filter Failure	Release of Plutonium Trapped in Final Filter	Unknown	None		

From scoping calculations, the following accidents (Table 3.3.3) are selected as the plant accidents that dominate the risk envelope.

Table 3.3.3
DOMINANT ACCIDENTS IN THE FFP

<u>Accident Section</u>	<u>Accident Description</u>
3.3.1.2	Aircraft Crash Into Powder Storage
3.3.1.3	Hydrogen Explosion in ROR Furnace
3.3.1.4	Hydrogen Explosion in Sintering Furnace
3.3.1.5	Hydrogen Explosion in Wet Scrap Recovery
3.3.1.6	Criticality Accident in Wet Scrap
3.3.1.7	Powder Shipping Container Spill
3.3.1.8	Exothermic Reaction in Powder Storage
3.3.1.9	Accident Summary

3.3.1.2 Air Crash Into Powder Storage

While the FFP has not been designed, it is estimated that the Category I wall will be about 2 feet in thickness of concrete in order to adequately shield the operating personnel from radiation. Chelapati, et al.⁽³⁰⁾ find that the probability of a nuclear plant wall of 2-foot thickness being penetrated by an aircraft is 5×10^{-7} /year. The ratio of target areas between the nuclear plant and the FFP is 27 so the inferred probability of aircraft penetration of the 2-foot shield wall is 2×10^{-8} /year. From the studies of the effects of aircraft crash into European Fuel Fabrication Plant^(30,31), the dominate release would result from a crash and fire in the powder storage area. From the FFP conceptual floor plan presented in Appendix A.3, the powder storage area is exposed on two sides; hence, a target area of 990 m^2 or 7.6% of the building to give an estimated probability of 1.5×10^{-9} /year.

Mishima⁽³¹⁾ calculates a fraction of inventory release of 0.2%. The maximum hopper size used in the FFP is 500 kg of which there are 56 (Table A.3.1); however, it is assumed that because of dividing walls (to prevent criticality) and other plant features, it is assumed that only the contents of one hopper are affected. The release is then 1 kg which is released into the atmosphere at a 200 m height (due to heat-caused plume effects⁽³¹⁾).

3.3.1.3 Hydrogen Explosion in the Reduction-Oxidation-Reduction Furnace

An explosion in the ROR could arise in three ways: improper mixing of the reducing gas, improper air purge prior to startup and oxygen evolution in the reduction process. This was analyzed in Section 3.1.1.10 for the FRP in which a probability of 9×10^{-4} /year was found. In the case of the FFP, however, there are 6 ROR furnaces; hence, the probability of a hydrogen explosion in an ROR furnace in the plant is 5×10^{-3} /year.

The effects of an accident are analyzed in reference 29 (p.14.5-10) in which it is postulated that the 40 kg of material in both reactors is ejected to challenge the ventilation system. This is overly conservative for a cell of 230 m^3 in which a powder aerosol of 1 g/m^3 could be formed⁽³⁾ for a total release to the ventilation of 230 g. It will be noted that this is a more dense aerosol than used in 3.3.1.2 because the aerosol used there was unstable.

3.3.1.4 Hydrogen Explosion in the Sintering Furnace

An explosion could occur in a sintering furnace in the same manner as in the ROR. There are 6 furnaces so the probability is the same as 3.3.1.3; hence, 5×10^{-3} /year.

The charge to each furnace is 2761 kg. If this whole gallery of 7000 m³ were filled with dust, 7 kg (1 g/m³) could be suspended and available to the ventilation.

3.3.1.5 Hydrogen Explosion in Wet Scrap Recovery

A hydrogen explosion could occur in wet scrap recovery in the acid dissolution of the scrap. The conditions for this occurrence are similar to that of 3.1.1.2 for the FRP, the probability for which is 3×10^{-4} /year. Should this occur, it is assumed that an aerosol 0.1g/m³ fills the 200 m³ cell. This is equivalent to 20 g of material, of which it is assumed that 50% is radioactive fuel-type material. Hence, the challenge to the air cleaning system is 10 g.

3.3.1.6 Criticality Accident in Wet Scrap Recovery

From an examination of the plant and considering the fact that except for the wet scrap recovery area, the fuel material is dry and of enrichment suitable for light water reactors, the most likely area for criticality to occur is in the wet scrap recovery.

The fault tree used to model this accident is presented in Figure B-3.1 which is a portion of a generic fault tree prepared in accordance with Regulatory Guide 3.33.

C4, Dilution Supply Failure, is assessed to fail from human error. reference 9 gives the frequency of procedural errors as 9.3×10^{-3} /man-year and will be used here.

C2 = C6, Failure of Manual Intervention, is 10^{-2} /task⁽¹⁵⁾.

C5, Control Failure, is assessed from Anyakora, et al.⁽¹⁰⁾ for a moderate environment as 0.57/hour. Hence G4, Failure of Dilution, is

$$G4 = (0.57 + 9.3 \times 10^{-3}) \times 10^{-2} = 5.7 \times 10^{-3}/\text{year}$$

$$C10 = C5$$

C1, Instrumentation Failure is taken from RSS⁽¹⁵⁾ as 8.8×10^{-3} /year.
Hence $G3 = (8.8 \times 10^{-3} + 5.7 \times 10^{-3}) \times 10^{-2} = 1.4 \times 10^{-4}$ /year.

C7, Overconcentration, is assumed a possibility (C7 = 1).

C8, Personnel Failure = 10^{-2} /task as before.

C9, Failure of Criticality Instrumentation, is determined as follows:
Reference 32 gives the failure rate of source range neutron instrumentation in reactors as 5.2×10^{-6} /D with one month maintenance interval. Hence $G7 = 10^{-2}$ /D and the probability of a criticality in a cell is assessed as 5.8×10^{-5} /year. This is somewhat smaller than the Selby, et al.⁽²¹⁾ estimate of 9×10^{-3} /year in fuel fabrication plants but similar to Cooperstein, et al.⁽⁴⁾ who estimate 3×10^{-5} /year.

This accident could result in 1×10^{19} fission in 8 hours, which is the regulatory position (R.G. 3.33). Physical damage is possible but has not generally occurred in past criticality accidents. In this instance, it is assumed that a rupture disc or seal is blown and the fission gases are released to the cell atmosphere. All of the noble gases and 25% of the iodine is assumed to be released to the atmosphere. The releases are presented in Table 3.3.4.

3.3.1.7. Powder Shipping Container Spill

The powder is conceptually received at the FFP in one cylinder for the Reference case or 6 cylinders for the PT case. In these cylinders the powder is compartmentalized into 40 kg lots. A cylinder is remotely removed by hooking an eyelet on the end of the cylinder and pulling it onto a roller conveyor that moves it inside of the unloading cell where the powder is pneumatically transferred through automatically sealing fittings permanently attached to each compartment. It is possible for the cylinder to fall from the conveyor or be punctured by the pneumatic transfer. It is possible that unauthorized equipment in the cell will damage the transfer cylinder. These events are diagrammed in the fault tree shown in Figure B-3.2.

Fall from Conveyor (C1). Since there is no experience with the transfer equipment, this cannot be directly assessed. One way this might occur would be the inappropriate placing of objects or equipment on the conveyor. Considering that this is a remote operation, this would be done with a manipulator. Assuming this is due to a procedural deficiency, reference 18 gives this as: $C1 = 9 \times 10^{-3}$ /year.

Table 3.3.4

ESTIMATED RADIONUCLIDE RELEASES FOR A
CRITICALITY ACCIDENT IN THE WET SCRAP RECOVERY CELL

Species	Curies Released in 8 Hours
Kr-83m	37
Kr-85m	170
Kr-87	1,500
Kr-88	660
Kr-89	41,000
Xe-133m	.55
Xe-133	13
Xe-135m	110
Xe-135	170
Xe-137	39,000
Xe-138	11,000
I-131	2
I-132	7
I-133	35
I-134	480
I-135	120
TOTAL CURIES	93,805

Fall Damages Container (C2) depends upon the specifications to which the container is constructed. Because low weight is not a requirement, it seems reasonable to expect the container to satisfy the requirements of 10 CFR 70 for Type B containers; i.e., the 15-foot drop test onto an unyielding surface, since the canister should be no higher than about 8 feet from the floor (to allow for removal from the upper portions of the PT cask). To model this accident, it is assumed that the container has a design safety factor of 2; i.e., it will stand a 30-foot drop. Furthermore, the variance about this value is 20% referring to Haugen⁽³³⁾ and modeling the stress and strength as normal distributions, a failure probability $C2 = 6 \times 10^{-7}$ is found.

Fall from Conveyor and Damage (G1) is thus assessed from these probabilities as: $G1 = 3 \times 10^{-3} \times 6 \times 10^{-7} = 2 \times 10^{-9}/\text{year}$.

Indexing Failure (C3). Without an actual design, this cannot be precisely analyzed; however, to be acceptable for remote applications, the failure rate should be less than about 0.5/year.

Interlock Failure (C4) is taken to be due to a single microswitch failure. Balfanz⁽¹⁶⁾ provides 4 values, the geometric mean of which gives $C4 = 0.8/10^6$ hours. Assuming monthly test and maintenance, the probability of interlock failure is $C4 = 0.8 \times 10^{-6} \times 1/24^* = 3 \times 10^{-4}/\text{demand}$.

Pneumatic Transfer Tube Does Not Buckle (C5). It may be possible if the alignment is very precise that the pneumatic transfer tube will not buckle when it impacts the powder container. No data are available, but estimates are that the probability of not buckling is $<0.1/\text{impact}$.

Thus the probability of Pneumatic Tube Puncture of the Powder Vessel (G2) is estimated to be

$$G2 = 0.5 \times 3 \times 10^{-4} \times 0.1 = 2 \times 10^{-5}/\text{year}$$

Attempt to Use Unauthorized Equipment (C6) is taken to be a procedural deficiency⁽¹⁸⁾ so $C6 = 9 \times 10^{-3}/\text{year}$.

*In the constant failure rate model, the test period is multiplied by 2 to obtain the average exposure time.

Unauthorized Equipment Damages Powder Container (C8) is estimated to be 0.1 because there must be good reasons to exclude certain items such as a fork truck.

Failure of Supervision (C7) is estimated from reference 15 as a general omission failure $C7 = 10^{-2}/D$.

Unauthorized Equipment Caused Damage (G3) is thus $G3 = 9 \times 10^{-3} \times 0.1 \times 10^{-2} = 9 \times 10^{-6}/\text{year}$.

Powder Container Spill (G4) is the combination $G4 = G1 + G2 + G3 = 2 \times 10^{-9} + 2 \times 10^{-5} + 9 \times 10^{-6}/\text{year} = 3 \times 10^{-5}/\text{year}$.

Should this event occur, the maximum amount of material for dispersal is 40 kg, but the presence of the compressed air for the vacuum intake of the pneumatic transfer could disperse the powder about the 200 m^3 room. Again using $1 \text{ gm}/\text{m}^3$ as the aerosol concentration, 200 gm are estimated to challenge the air cleaning system.

3.3.1.9 Exothermic Reaction in Powder Storage

The heat production rate from radioactivity is 3.2 W/kg (Reference 34) and 12.8 W/kg (PT) for 500 kg hoppers. This is a heat source of 1.6 kW and 6.4 kW, respectively. A possible conceptual design for a tank is 5 cm thick and 90 x 90 cm on each side. (The tanks are chosen to be thin to limit temperature gradients.) Using handbook values for a vertical slab in free air⁽³⁵⁾, the equilibrium temperature is 570°F and 1900°F and a rate of rise of $79^\circ\text{F}/\text{hour}$ and $315^\circ\text{F}/\text{hour}$, respectively. This is hot enough to result in oxidation of the UO_2 to U_3O_8 if no cooling is provided (an inert atmosphere could be used but would be complicated with pneumatic transfer of powder).

To control the temperature of the powder hoppers, it is assumed that closed-loop water cooling channels are affixed to the exterior surfaces of the hoppers. This demineralized, pressurized water is circulated through a heat exchanger connected with the main plant water. A fault tree model of this system is presented in Figure B-3.2.

Pump Failure (C1) Balfanz⁽¹⁶⁾ gives several values for medium sized pumps from a GE internal report. The geometric mean: $(14 \times 40 \times 19 \times 23 \times 14 \times 15 \times 68 \times 63)^{1/8} = 26 \times 10^{-6}/\text{hour}$; $C1 = 0.22/\text{year}$.

Check Valve Failure (C2). Reference 15 gives $9 \times 10^{-4}/\text{year}$ as the failure rate for reverse leaks, so $C2 = 9 \times 10^{-4}/\text{year}$.

Line Clogging (C3) for noncorroding systems with demineralized water is estimated to be very small: $C3 = \epsilon$.

Insufficient Loop Water (C4) is for a closed system. It might be necessary to add water every five years. This would be a general error of omission which from reference 15 is 10^{-2} /task, so $C4 = 2 \times 10^{-3}$ /year.

Heat Sink Failure (C5). This is taken to be very unlikely because of water supply system redundancies; hence, $C5 = \epsilon$.

Cooling System 1 Failure (G2) is the combination of these rates: $G2 = 0.2$ /year.

Cooling System 2 Failure (G3) is the same as system 1, however, must be placed on a demand basis by considering the probability of failing at the same time as system 1. Assuming a 3-hour repair time and immediate access, this is $G3 = 0.2 \times 3/8760 = 7 \times 10^{-5}/D$; however, since systems 1 and 2 may fail in any order, this number is doubled: $G3 = 1.4 \times 10^{-4}/D$.

Loss of Hopper Cooling (G1). This is found by combining G2 and G3 as $G1 = 0.2 \times 1.4 \times 10^{-4} = 3 \times 10^{-5}$ /year. The probability is not the same for PT and for Reference because of the much slower heating of the Reference material. The time to reach reaction temperature is about 1.5 hours for PT and more than 6 hours for the Reference cycle. The estimated probability for establishing cooling during this longer time for PT is taken from reference 7 is a factor of 5, thus $G1 = 3 \times 10^{-5}$ /year (PT cycle) and 1.5×10^{-6} /year (Reference cycle).

Whether this accident can proceed or not depends on the air exposure to the powder. Here it is assumed that the design is such that air is exposed to the powder and can be drawn in as the reaction proceeds.

UO_2 exposed to air at temperatures above $572^{\circ}F$ can oxidize to U_3O_8 . The reaction occurring at $1110^{\circ}F$ produces fine powders but at $1290^{\circ}F$ the powder was found to be mixed with lumps.

It is assumed that when this reaction begins at a hot area, it propagates throughout the powder resulting in expansion and heating of the hopper sufficiently that structural failure occurs and the powder spills onto the floor of the cell. Following Mishima's analysis⁽³¹⁾, approximately 11 w/o of the material is less than $10 \mu m$ AED and less than 0.4 w/o less than $5 \mu m$ AED and the assessed release is 0.2 w/o or 1 kg released to challenge the air cleaning system.

3.3.1.9 Radiological Public Accident Risk

The following tables summarize these accidents and releases using the HEPA filter penetration fraction data from Appendix A.6. In the FFP, all accident releases except for the earthquake and air crash are normally filtered through 3 HEPA filters (1 cell filter and a double final filter). All accidents from dry processes use aerosol sizes characteristic of dry processes and similarly wet processes assume the appropriate aerosol distribution.

Table 3.3.5 assumes no filters have failed and use directly the accident probabilities just determined. Table 3.3.6 assumes one failed filter and adjusts the accident probability according to the probability of a failed filter and Table 3.3.7 assumes 2 failed filters with suitable probability adjustment as given in Appendix A.6. The summarized risks for combining these results are: Reference cycle, 3.1×10^{-5} person-rem/plant-year or 1.5×10^{-8} latent cancers/plant-year; PT cycle, 3.3×10^{-5} person-rem/plant year or 3.4×10^{-8} latent cancers/plant-year. The risks in energy equivalents are: Reference cycle, 4×10^{-7} person-rem/GWe-year or 2×10^{-10} latent cancer/GWe-year; PT cycle, 4.3×10^{-7} person-rem/GWe-year or 4.5×10^{-10} latent cancer/GWe-year.

3.3.2 Relative Public Atmospheric Risk of PT and Reference FFP

Consequences to the public from routine radioactive effluents from the PT and Reference FFPs are developed in Appendix G. The results are presented in Table 3.3.8.

Table 3.3.5a

SUMMARY OF REFERENCE CYCLE FFP ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(No Failed HEPA Filters)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem	Total Latent Cancer
						Plant-Yr	Plant-Yr
3.3.1.2	Air Crash	1.5×10^{-9}	1×10^3	5×10^2	8	8×10^{-7}	1.1×10^{-8}
3.3.1.3	H ₂ Explosion in ROR	5×10^{-3}	2×10^{-11}	1.1×10^{-11}	1.6×10^{-13}	5×10^{-14}	8×10^{-16}
3.3.1.4	H ₂ Explosion in Sintering	5×10^{-3}	7×10^{-10}	4×10^{-10}	5×10^{-12}	1.9×10^{-12}	3×10^{-14}
3.3.1.5	H ₂ Explosion in Wet Scrap	3×10^{-4}	2×10^{-11}	1.1×10^{-11}	1.6×10^{-13}	3×10^{-15}	5×10^{-17}
3.3.1.6	Criticality in Wet Scrap	6×10^{-5}	9×10^4	2	2×10^{-4}	3×10^{-5}	4×10^{-9}
3.3.1.7	Powder Shipping Container Spill	3×10^{-5}	2×10^{-11}	1.1×10^{-11}	1.6×10^{-13}	3×10^{-16}	5×10^{-18}
3.3.1.8	Exothermic Reaction in Powder Storage	1.5×10^{-6}	1×10^{-10}	5×10^{-11}	8×10^{-13}	8×10^{-17}	1.2×10^{-18}
Total Risk						3×10^{-5}	1.5×10^{-8}

Table 3.3.5b

SUMMARY OF PT CYCLE FFP ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(No Failed HEPA Filters)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem	Total Latent Cancer
						Plant-Yr	Plant-Yr
3.3.1.2	Air Crash	1.5×10^{-9}	1.4×10^3	1.1×10^3	2×10^1	1.6×10^{-6}	3×10^{-8}
3.3.1.3	H ₂ Explosion in ROR	5×10^{-3}	3×10^{-11}	3×10^{-11}	5×10^{-13}	1.4×10^{-13}	3×10^{-15}
3.3.1.4	H ₂ Explosion in Sintering	5×10^{-3}	1×10^{-9}	9×10^{-10}	1.8×10^{-4}	5×10^{-12}	9×10^{-14}
3.3.1.5	H ₂ Explosion in Wet Scrap	3×10^{-4}	3×10^{-11}	3×10^{-11}	5×10^{-13}	8×10^{-15}	1.6×10^{-16}
3.3.1.6	Criticality in Wet Scrap	6×10^{-5}	9×10^4	5×10^{-1}	6×10^{-5}	3×10^{-5}	4×10^{-9}
3.3.1.7	Powder Shipping Container Spill	3×10^{-5}	3×10^{-11}	3×10^{-11}	5×10^{-13}	8×10^{-16}	1.6×10^{-17}
3.3.1.8	Exothermic Reaction in Powder Storage	3×10^{-5}	1×10^{-11}	9×10^{-12}	1.8×10^{-13}	3×10^{-16}	5×10^{-18}
Total Risk						3.2×10^{-5}	3.4×10^{-8}

Table 3.3.6a

SUMMARY OF REFERENCE CYCLE FFP ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(1 Failed HEPA Filter)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem	Total Latent Cancer
						Plant-Yr	Plant-Yr
3.3.1.2	Air Crash	1.5×10^{-11}	1×10^3	5×10^2	8	8×10^{-9}	1.2×10^{-10}
3.3.1.3	H ₂ Explosion in ROR	5×10^{-5}	1×10^{-7}	5×10^{-8}	8×10^{-10}	3×10^{-12}	4×10^{-14}
3.3.1.4	H ₂ Explosion in Sintering	5×10^{-5}	4×10^{-6}	2×10^{-6}	3×10^{-8}	1×10^{-5}	1.5×10^{-12}
3.3.1.5	H ₂ Explosion in Wet Scrap	3×10^{-6}	6×10^{-8}	3×10^{-8}	5×10^{-10}	9×10^{-14}	1.5×10^{-15}
3.3.1.6	Criticality in Wet Scrap	6×10^{-7}	9×10^4	2	2×10^{-4}	1.2×10^{-6}	1.2×10^{-10}
3.3.1.7	Powder Shipping Container Spill	3×10^{-7}	1×10^{-7}	5×10^{-8}	8×10^{-10}	1.5×10^{-14}	2×10^{-16}
3.3.1.8	Exothermic Reaction in Powder Storage	1.5×10^{-8}	5×10^{-7}	3×10^{-7}	4×10^{-9}	5×10^{-15}	6×10^{-17}
Total Risk						1.2×10^{-6}	2.4×10^{-10}

Table 3.3.6b

SUMMARY OF PT CYCLE FFP ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(1 Failed HEPA Filter)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem	Total Latent Cancer
						Plant-Yr	Plant-Yr
3.3.1.2	Air Crash	1.5×10^{-11}	1.4×10^3	1.1×10^3	2×10^1	1.7×10^{-8}	3×10^{-10}
3.3.1.3	H ₂ Explosion in ROR	5×10^{-5}	2×10^{-7}	1.8×10^{-7}	4×10^{-9}	9×10^{-12}	2×10^{-13}
3.3.1.4	H ₂ Explosion in Sintering	5×10^{-5}	5×10^{-6}	4×10^{-6}	9×10^{-8}	2×10^{-10}	5×10^{-12}
3.3.1.5	H ₂ Explosion in Wet Scrap	3×10^{-6}	8×10^{-8}	7×10^{-8}	1.4×10^{-9}	2×10^{-13}	4×10^{-15}
3.3.1.6	Criticality in Wet Scrap	6×10^{-7}	9×10^4	2	2×10^{-4}	1.2×10^{-6}	1.2×10^{-10}
3.3.1.7	Powder Shipping Container Spill	3×10^{-7}	1×10^{-7}	9×10^{-8}	1.8×10^{-9}	3×10^{-14}	5×10^{-16}
3.3.1.8	Exothermic Reaction in Powder Storage	1.5×10^{-8}	7×10^{-7}	6×10^{-7}	1.3×10^{-8}	9×10^{-15}	2×10^{-16}
Total Risk						1.2×10^{-6}	3.1×10^{-10}

Table 3.3.7a

SUMMARY OF FFP REFERENCE ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(2 Failed HEPA Filters)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem	Total Latent Cancer
						Plant-Yr	Plant-Yr
3.3.1.2	Air Crash	4.5×10^{-16}	1×10^3	5×10^2	7	2×10^{-13}	3×10^{-15}
3.3.1.3	H ₂ Explosion in ROR	1.5×10^{-9}	0.01	5×10^{-3}	8×10^{-5}	8×10^{-12}	1.2×10^{-13}
3.3.1.4	H ₂ Explosion in Sintering	1.5×10^{-9}	0.4	2×10^{-1}	3×10^{-3}	3×10^{-10}	5×10^{-12}
3.3.1.5	H ₂ Explosion in Wet Scrap	9×10^{-11}	6×10^{-4}	3×10^{-4}	5×10^{-6}	3×10^{-14}	5×10^{-16}
3.3.1.6	Criticality in Wet Scrap	1.8×10^{-11}	9×10^4	5×10^{-1}	6×10^{-5}	9×10^{-10}	1.1×10^{-15}
3.3.1.7	Powder Shipping Container Spill	9×10^{-12}	0.01	5×10^{-3}	8×10^{-5}	5×10^{-14}	7×10^{-16}
3.3.1.8	Exothermic Reaction in Powder Storage	4.5×10^{-13}	0.06	3×10^{-2}	5×10^{-4}	1.4×10^{-14}	2×10^{-16}
Total Risk						1.2×10^{-9}	5.1×10^{-12}

Table 3.3.7b

SUMMARY OF PT CYCLE ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(2 Failed HEPA Filters)

Accident Section	Description	Probability (/Plant-Yr)	Curies Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem	Total Latent Cancer
						Plant-Yr	Plant-Yr
3.3.1.2	Air Crash	4.5×10^{-16}	1.4×10^3	1.1×10^3	20	5×10^{-13}	9×10^{-15}
3.3.1.3	H ₂ Explosion in ROR	1.5×10^{-9}	2×10^{-2}	1.8×10^{-2}	4×10^{-4}	3×10^{-11}	6×10^{-13}
3.3.1.4	H ₂ Explosion in Sintering	1.5×10^{-9}	0.6	5.5×10^{-1}	1.1×10^{-2}	8×10^{-8}	1.7×10^{-8}
3.3.1.5	H ₂ Explosion in Wet Scrap	9×10^{-11}	8×10^{-4}	7×10^{-4}	1.4×10^{-5}	7×10^{-14}	1.3×10^{-15}
3.3.1.6	Criticality in Wet Scrap	1.8×10^{-11}	9×10^4	5×10^{-1}	6×10^{-5}	9×10^{-10}	1.1×10^{-15}
3.3.1.7	Powder Shipping Container Spill	9×10^{-12}	2×10^{-2}	1.8×10^{-2}	4×10^{-4}	1.6×10^{-13}	4×10^{-15}
3.3.1.8	Exothermic Reaction in Powder Storage	4.5×10^{-13}	8×10^{-2}	7×10^{-2}	1.4×10^{-3}	3×10^{-14}	9×10^{-16}
Total Risk						8×10^{-8}	1.7×10^{-11}

Table 3.3.8

SUMMARY ROUTINE RADIOLOGICAL RISKS TO PUBLIC FOR THE FFP

Cycle	<u>Total Person-Rem</u> Plant-Yr	<u>Total Latent Cancer</u> Plant-Yr	<u>Total Person-Rem</u> GWe-Yr	<u>Total Latent Cancer</u> GWe-Yr
Reference	7×10^{-3}	1.9×10^{-4}	9.3×10^{-5}	2.5×10^{-6}
PT	1.7×10^{-2}	6.8×10^{-4}	2.3×10^{-4}	9.1×10^{-6}

3.3.3 Relative Occupational Radiological Risk of PT and Reference FFP

No difference in occupational dose between the Reference and PT FFPs is discerned. Both facilities are similarly designed, i.e., highly automated, hot cell type operations conducted remotely. Both facilities are designed to minimize routine worker exposure by shielding provisions to limit dose to 500 millirem per year. Further all activities will be conducted within ALARA constraints which in practice means controlling the environment within which the work is done as much as is possible as opposed to excessive individual exposure or utilization of transient work force to spread out exposure. The estimated exposure for the FFP is given in Table 3.3.9. The values derived are from the estimated work force size and the allowable exposure levels. The assumptions made in arriving at these values are given in Appendix D.

Table 3.3.9
ESTIMATED OCCUPATIONAL EXPOSURE -- FFP

Activity	Person-Rem/Year	Person-Rem/GWe-Year	Health Effects** Fatalities/GWe-Year
Routine Operations	230	3	3×10^{-4}
Maintenance	230	3	3×10^{-4}
Abnormal Occurrences*	10	0.1	1×10^{-5}
Accidents	5×10^{-2}	7×10^{-4}	7×10^{-8}
Decommissioning	14	0.19	2.5×10^{-5}
Total		6.3	6.3×10^{-4}

*Defined as personnel exposure exceeding limits set in 10CFR20

**From the BEIR report equivalence estimate of 1×10^{-4} fatalities per person-rem

3.3.4 Relative Nonradiological Risks of PT and Reference FFP

The chemical and industrial risks developed in Appendix E are summarized in Table 3.3.10. The primary chemical hazards are associated with the fraction of site power plant effluents attributable to utility consumption by the respective FFP. The health effects associated with these effluents completely dominate the nonradiological risk.

Table 3.3.10

SUMMARY EFFECTS NONRADIOLOGICAL CAUSES

Cause	Premature Deaths (Fatality/GWe-Yr)	Economic Deaths (Fatality/GWe-Yr)	Total (Fatality/GWe-Yr)
Effluents	2.7×10^{-3}	8×10^{-3}	1.1×10^{-2}
Industrial Accidents	2.3×10^{-4}	3.1×10^{-3}	3.3×10^{-3}
Total			1.4×10^{-2}

3.4 RISK OF THE FUELS FABRICATION PLANT - WASTE TREATMENT FACILITY (FFP-WTF)

3.4.1 FFP-WTF Dominant Accidents

3.4.1.1 Overview and Preliminary Hazards Analysis

The FFP-WTF is summarily described in Appendix A.4. The major facility differences between the FRP-WTF and the FFP-WTF are also listed. The FFP-WTF is smaller and simpler than the FRP-WTF. The major differences are fewer CEC units due to lower AmCm throughput, no TBP solvent extraction system, one less CMP cycle, and no provisions for treating dissolver solids and cladding hulls. The hazards previously listed in Table 3.2.1 are applicable to the FFP-WTF with the exception of those systems identified in the Table as specific to the FRP-WTF.

One significant difference is the degree of actinide production concentration in the FFP-WTF. The resultant product is about 6.9 times more concentrated in Am and Cm than the equivalent stream in the FRP-WTF. This results in greater risk from accidents affecting the integrity of the concentration or product receiver vessels.

The FFP-WTF accidents selected from analysis of Table 3.2.1 parallel those selected for equivalent operations in the FRP-WTF. These are listed in Table 3.4.1 below.

Table 3.4.1
FFP-WTF ACCIDENTS ANALYZED

<u>Accident Section</u>	<u>Accident Description</u>
3.4.1.2	Waste Treatment Facility Criticality
3.4.1.3	Radiolytic Hydrogen Explosion <ul style="list-style-type: none">● Vessel● Vessel offgas system
3.4.1.4	Resin Explosion in Actinide Product Concentrator
3.4.1.5	"Red Oil" Explosions in WTF Concentrators
3.4.1.6	Resin Explosion/Fire
3.4.1.7	Facility Fire <ul style="list-style-type: none">● Solvent fire● Solvent moist explosion
3.4.1.8	Accident Summary

3.4.1.2 Waste Treatment Facility Criticality

The criticality accident in 3.2.1.2 is equally applicable to the FFP-WTF. An accident frequency of 6.4×10^{-6} /cell-yr was previously determined. In the FFP-WTF there are four areas of risk for a value of 2.6×10^{-5} /plant yr. The release is assumed to consist of 93,805 curies of volatile fission products (see Table 3.1.9) over an eight hour period. Candidate vessels are listed in Table 3.4.2.

3.4.1.3 Radiolytic Hydrogen Explosion

In Section 3.2.1.3 the likelihood and consequences of radiolytic hydrogen explosions were determined for a vessel vent filter and for an actinide product vessel in the FRP-WTF. The following vessels are likely candidates for such an accident in the FFP-WTF.

Table 3.4.2 FFP-WTF Candidate Vessels for Nuclear Excursion

<u>Vessel</u>	<u>Plutonium (kgs)</u>	<u>H⁺ (m)</u>	<u>Event Sequence</u>
<u>Ceric Leach</u>			
HEPA Leacher	1.1	4.0	<ul style="list-style-type: none">● undissolved fissile material● omission of neutron poison● inadvertent fissile material precipitation
<u>Salt Waste</u>			
2 EHOH Contactor	2.4	3.5	<ul style="list-style-type: none">● fissile material transfer into solvent● inadvertent precipitation
Holdup Tank	4.2	3.5	
Alkaline Feed Concentrator	5.6	---	
<u>Product Concentration</u>			
TK-24	2.4	2.5	<ul style="list-style-type: none">● fissile material precipitation● deformation of slab tank to non-safe geometry
Concentrator #6	15	8.0	
TK-23	12	8.0	
<u>Sump Collection Tank</u>			
TK-43	≥ minimum critical mass	≥1.5	<ul style="list-style-type: none">● fissile material accumulation and precipitation

Table 3.4.3
FFP-WTF HYDROGEN EXPLOSION CANDIDATE VESSELS

Vessel	Function	w/l	Estimated H ₂ (l/hr)	Time* (minutes) (≥4%)
TK-16	CEC Recycle	20	7	≥156
TK-3**	Am/Cm Product #1 Concentrator	480	20	≥ 7
TK-4**	Am/Cm Feed #1 Concentrator	80	22	≥ 37
TK-14**	CEC Feed	29	32	≥ 30
TK-31**	CEC Feed	29	35	≥ 40

*Based on 50% void space
 **Special provisions are assumed for hydrogen dilution by purge air

The releases below are calculated for an explosion in TK-3 based on assumptions identical to those previously made for the FRP-WTF vessels (Section 3.2).

Table 3.4.4
RELEASES FOR HYDROGEN EXPLOSION IN TK-3 Am/Cm PRODUCT

Species	Inventory (curies)	Release to Air Tunnel (curies)
Am	7.2×10^2	3
Cm	5.3×10^5	2×10^2
Totals	5.3×10^5	2×10^2

An explosion in a vessel vent filter in the FFP-WTF would be identical to that previously analyzed for the FRP-WTF, namely 8×10^{-4} /yr for a combined release of 1.3×10^2 curies of Am/Cm.

3.4.1.4 Resin Explosion in Actinide Product Concentrator

The mechanism and analysis of resin introduction to, and violent reaction in a product concentrator were previously described in Section 3.2.1.4. The assessed frequency of 2×10^{-3} /yr is retained though there are only 5 CEC risks versus 50 in the FRP-WTF. The higher Am/Cm loading in the FFP-WTF increased the likelihood of solubilized resin. The attendant release is given in Table 3.4.5 below.

Table 3.4.5

RELEASES TO AIR TUNNEL ATTENDANT ON EXPLOSION/RUPTURE
OF PRODUCT CONCENTRATOR (curies)

Species	Concentrator #1	
	Inventory	Release
Pu	---	---
Am	6.1×10^4	3
Cm	4.5×10^6	2×10^2
<u>TOTAL</u>	<u>4.6×10^6</u>	<u>2×10^2</u>

3.4.1.5 "Red Oil" Explosions in WTF Concentrators

Candidate FFP-WTF concentrators are listed in Table 3.4.6. Estimated frequencies for accident occurrence are listed in Table 3.4.7. These values stem from the analysis presented in Section 3.2.

Table 3.4.6 FFP-WTF Concentrators

Vessel	Concentrator Purpose	Inventory (liters)	Decay Heat (w/t)	Solution Acidity (moles H ⁺)	U (kgs)	Pu (kgs)	Np (kgs)	Am (kgs)	Cm (kgs)	Fission Products (curies)	Comments
#1	Actinide product from cation exchange chromatography	300	478	2.5	---	---	---	47	58		Significant, radiolytic hydrogen generation rate. Potential introduction and accumulation of resin fragments and fines. Loss of coolant accident possible. Criticality possible, but remote.
#2	Uranium product strip solution from CMP solvent extraction	4000	---	0.3	22	---	---	---	---		Multiple decanting and steam stripping to preclude "red oil." Phase inversion most likely transfer mechanism.
#3	An/Ln product strip solution from CMP solvent extraction	4000	28	0.5	10	0.8	2.8	37	46		Criticality possible but remote. "Red oil" possible if phase inversion introduces solvent. Significant radiolytic hydrogen generation rate.
#4	Alkaline feed for salt waste treatment	4000	---	---	11	5.6	---	---	---		Possible criticality due to precipitated plutonium. Possible solids pluggage.
#5	Acid recovery--concentrator overheads and 2-EHOB strip	6000	---	4.0	---	---	---	---	---		"Red oil" type reaction possible (solids from 2-EHOB filtration).
#6	Leachate from alpha waste treatment, product from salt waste treatment	4000	51	8.0	84	15	9.6	0.2	0.3		Possible, ceric ion corrosion if not reduced. "Red oil" type reaction, particularly with high nitric acid and possible entrained degraded TBP or 2EHOB. Feed previously decanted and steam stripped. Possible criticality due to large plutonium inventory.
#7	Acidic feed to salt waste treatment	4000	---	4.0	8	0.7	1.6	0.4	0.4		Nitrated organic reactions possible by inadvertent introduction of degraded CMP or resin fines and fragments.
#8	Column #5 raffinate (CMP treated FPP IAW)	4000	---	<0.5	---	---	---	---	---		Fluoride induced corrosion from FPP dissolver. Potential criticality from precipitated plutonium with high Pu losses. Nitrated solvent reaction possible with failure to denitrate.

--- negligible

<u>Vessel</u>	<u>Concentrator Purpose</u>	<u>Inventory (liters)</u>	<u>Decay Heat (w/t)</u>	<u>Solution Acidity (moles H⁺)</u>	<u>U (kgs)</u>	<u>Pu (kgs)</u>	<u>Np (kgs)</u>	<u>Am (kgs)</u>	<u>Cm (kgs)</u>	<u>Fission Products (curies)</u>	<u>Comments</u>
#9	Acid overheads (2nd stage)	1000	---	10-12.0	---	---	---	---	---		Halide-acid corrosion likely, if purging not done.
TK-43	Sump waste collection and reflux tank (assume 0.01% of plant inventory)	4000	---	≥4.0	---	---	---	---	---		Loss of control of nitrated solvent reactions possible. Potential for criticality.

Table 3.4.7
FAULT TREE QUANTIFICATION

Concentration Type	FFP Conc.#	Solution in Feed $\frac{(G2)}{(/yr)}$	Heavy Metals/ Nitric $\frac{(C19)}{(/yr)}$	"Red Oil" Explosion $\frac{(G1)}{(/yr)}$
A.	2, 3	1×10^{-1}	1×10^{-2}	2×10^{-7}
B.	4	---	---	---
C.	5, 6	$2 \times 10^{-2}/\text{day}$	1	4×10^{-6}
D.	7	---	1	2×10^{-4}
E.	8	---	1×10^{-2}	4×10^{-8}
F.	9	---	1	---
G.	TK-43	1	1	2×10^{-4}

*The rationale for the individual vessels is as follows:

A. Concentrators

- Solvent in feed is determined by likelihood of CMP phase conversion due to over saturation
- Nitric acid concentration normally low; loss of formic acid (task performance error) most likely mechanism

B. Concentrators

- Bottoms solution always alkaline

C. Concentrators

- Solvent introduction governed by likelihood of misvalving or column flood transferring 2-EHOH with associated pump down of feed tank
- Heavy metals/nitric acid always present

D. Concentrators

- Solvent transfer is normal due to presence of solvent in wastes
- Heavy metals/nitric acid are present by virtue of estimated waste characteristics

E. Concentrators

- Reaction dependent on failure to denitrate

F. Concentrators: No solvent present

G. Sump Collection Tanks

- Solvent transfer normal
- Heavy metals/nitric acid are normally present

The product concentrator is omitted here, as a resin reaction has a greater likelihood than a red oil explosion (as can be seen by comparing accident frequencies).

A release fraction of 1×10^{-5} will be used as a conservative estimate since the volumes of the process cells vary. The air tunnel releases for each vessel may then be calculated based on the above release factor. No credit will be taken for the wire mesh pad used as a roughing filter for the flow channel venting the cell to the air tunnel. The air tunnel releases are tabulated in Table 3.4.8. It should be noted that this accident is more serious than its FRP-WTF counterpart by virtue of the greater Am/Cm concentrations in the concentrator.

Table 3.4.8
AIR TUNNEL RELEASES: RED OIL EXPLOSION IN
FFP-WTF CONCENTRATORS (curies)

<u>Species</u>	Concentrator			
	#3	#6	#7	TK-43*
Pu	0.2	0.5	2×10^{-2}	8×10^{-3}
Am	0.8	4×10^{-3}	9×10^{-3}	2×10^{-3}
Cm	60	0.4	0.5	0.1
TOTALS	60	0.9	0.6	0.1

*Assume 1×10^{-8} fraction of facility inventory

3.4.1.6 Resin Explosion/Fire

The likelihood of a violent resin reaction in a CEC bed in the FFP-WTF is something less than that for the FRP-WTF due to the significant reduction in the number of units (5 versus 50). For the FFP-WTF this means an assessed frequency of 4×10^{-5} /yr. The consequences of a violent resin reaction are somewhat enhanced in the FFP-WTF in contrast to the FRP-WTF by virtue of the greater bed loading. The releases to the air tunnel are given below (Table 3.4.9) based on 10% resin combustion. Based on a combustion soot loading of 3000 mg/m^3 , a release factor of 0.1 is reasonable for the cell size.

3.4.1.7 Facility Fire

FFP-WTF major inventories of flammable and combustible materials include process solvents, ion exchange resins, and contaminated solid wastes from in-cell activities. This analysis is similar to the one for the FRP-WTF (3.2.1.7) in that a solvent fire and a solvent mist explosion are examined. The resin fire was included on the previous resin explosion. Solid waste fires are of limited impact outside the facility.

Table 3.4.9
AIR TUNNEL RELEASES FROM RESIN EXPLOSION

Species	FFP-WTF (curies)
Am	1.9×10^2
Cm	1.3×10^4
TOTAL	1.3×10^4

The columns containing significant quantities of actinide bearing solvents are listed in Table 3.4.10. Also listed are the inventories of actinides of concern.

Solvent Fire

The frequency of a significant solvent leak followed by ignition was previously estimated at 2×10^{-4} /year (3.1.1.3). Based on the same assumptions and analysis as presented in 3.2.1.7, the following constitutes the release to the cell exhaust.

Table 3.4.1?
RELEASE TERMS FOR SOLVENT FIRE (curies)

Species	Col. #5*
Pu	3
Am	5×10^1
Cm	5×10^3
TOTAL	5×10^3

*Solvent totally consumed

Table 3.4.10

FFP-WTF SX COLUMNS - SOURCE TERMS FOR SOLVENT FIRE

Vessel	Function	Organic	Organic Vol. (lbs)	U (kg)	Pu (kg)	We (kg)	Am (kg)	Cm (kg)
Col # 5	IAW Extraction	CMP	200	0.2	0.02	0.06	0.52	0.64
Col # 6	An Strip	CMP	100	0.1	0.01	0.03	0.26	0.32

Solvent Mist Explosion

This was previously estimated at 7×10^{-4} /yr (3.2.1.7). Assuming that the quantity of mist and aerosol constitutes the source term, the following releases to the exhaust ventilation tunnel result.

Table 3.4.12
RELEASES FROM SOLVENT MIST EXPLOSION (curies)

Species	Co1 #5
Pu	2
Am	3
Cm	2.5×10^2
TOTAL	2.5×10^2

3.4.1.8 Radiological Public Accident Risk

The following tables summarize these accidents and atmospheric releases using HEPA penetration fraction data from Appendix A.6 and an assumed df of 10^{-1} for the deep bed filter. All releases are assumed to correspond to the wet process fraction on A.6.

Table 3.4.13 assumes no filter failures and utilizes the accident probabilities directly. Tables 3.4.14 and 3.4.15 assume single and double HEPA failures (accident independent) and adjusts probabilities accordingly.

The summarized risk is 6×10^{-5} person-rem/plant-yr or 7×10^{-9} latent cancer/plant-yr. In terms of power equivalent this is 8×10^{-7} person-rem/GWe-yr or 9×10^{-11} latent cancer/GWe-yr.

3.4.2 Public Atmospheric Risk of FFP-WTF

Consequences to the public from routine radioactive effluents from the FFP-WTF are developed in Appendix G. Routine releases can be expected to result in a population dose of 0.55 person-rem/plant-yr or 0.12 latent cancer/plant-yr in terms of health effects. On an energy equivalence basis this corresponds to 7.3×10^{-3} person-rem/GWe-yr or 3.7×10^{-4} latent cancer/GWe-yr.

Table 3.4.13

SUMMARY OF FFP-WTF ACCIDENT PROBABILITIES, CONSEQUENCES AND RISK
(No Failed HEPA Filters)

Accident Section	Description	Probability (/Plant-Yr)	Curie Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.4.1.2	Criticality in Process Cell	3×10^{-5}	9×10^4	2.	2×10^{-4}	6×10^{-5}	6×10^{-9}
3.4.1.3	Hydrogen Explosion in Actinide Bearing Vessel	3×10^{-4}	1×10^{-7}	3×10^{-8}	6×10^{-10}	9×10^{-12}	1.8×10^{-13}
	Hydrogen Explosion in Vessel Vent System	8×10^{-4}	8×10^{-8}	2×10^{-8}	4×10^{-10}	1.6×10^{-11}	3×10^{-13}
3.4.1.4	Nitrated Organic Explosion in Am Product Concentrator	2×10^{-3}	1×10^{-7}	3×10^{-8}	6×10^{-10}	6×10^{-11}	1.2×10^{-12}
3.4.1.5	"Red Oil" Explosion in Process Concentrator	2×10^{-4}	4×10^{-5}	8×10^{-9}	2×10^{-10}	1.6×10^{-12}	4×10^{-14}
3.4.1.6	Resin Explosion in CEC Unit	4×10^{-5}	8×10^{-6}	2×10^{-6}	4×10^{-8}	6×10^{-11}	1.2×10^{-12}
3.4.1.7	Facility Solvent Fire	2×10^{-4}	3×10^{-7}	3×10^{-7}	5×10^{-9}	6×10^{-11}	1×10^{-12}
	Solvent Mist Explosion	7×10^{-4}	1.5×10^{-8}	1.6×10^{-8}	3×10^{-10}	1.1×10^{-11}	2×10^{-13}
Total Risk						6×10^{-5}	6×10^{-9}

Table 3.4.14

SUMMARY OF FFP-WTF ACCIDENT PROBABILITIES, CONSEQUENCES AND RISKS
(1 Failed HEPA Filter)

Accident Section	Description	Probability (/Plant-Yr)	Curie Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.4.1.2	Criticality in Process Cell	3×10^{-7}	9×10^4	2.	2×10^{-4}	6×10^{-7}	6×10^{-11}
3.4.1.3	Hydrogen Explosion in Actinide Bearing Vessel	3×10^{-6}	1×10^{-4}	3×10^{-5}	6×10^{-7}	9×10^{-11}	2×10^{-12}
	Hydrogen Explosion in Vessel Vent System	8×10^{-6}	8×10^{-5}	2×10^{-5}	4×10^{-7}	1.6×10^{-10}	3×10^{-12}
3.4.1.4	Nitrated Organic Explosion in Am Product Concentrator	2×10^{-5}	1×10^{-4}	3×10^{-5}	6×10^{-7}	6×10^{-10}	1.2×10^{-11}
3.4.1.5	"Red Oil" Explosion in Process Concentrator	2×10^{-6}	4×10^{-5}	8×10^{-6}	2×10^{-7}	1.6×10^{-11}	4×10^{-13}
3.4.1.6	Resin Explosion in CEC Unit	3×10^{-7}	8×10^{-3}	2×10^{-3}	2×10^{-5}	6×10^{-10}	6×10^{-12}
3.4.1.7	Facility Solvent Fire	2×10^{-6}	3×10^{-3}	2×10^{-3}	5×10^{-5}	4×10^{-9}	1×10^{-10}
	Solvent Mist Explosion	7×10^{-6}	1.5×10^{-4}	1.6×10^{-4}	3×10^{-6}	1.1×10^{-9}	2×10^{-11}
Total Risk						6×10^{-7}	2×10^{-10}

Table 3.4.15

SUMMARY OF FFP-WTF ACCIDENT PROBABILITIES, CONSEQUENCES AND RISK
(2 Failed HEPA Filters)

Accident Section	Description	Probability (/Plant-Yr)	Curie Release to Atmosphere	Consequences		Risk (p x c)	
				Total Person-Rem	Total Latent Cancer	Total Person-Rem Plant-Yr	Total Latent Cancer Plant-Yr
3.4.1.2	Criticality in Process Cell	9×10^{-12}	9×10^4	2.	2×10^{-4}	1.8×10^{-11}	1.8×10^{-15}
3.4.1.3	Hydrogen Explosion in Actinide Bearing Vessel	9×10^{-11}	20	5	0.1	5×10^{-10}	9×10^{-11}
	Hydrogen Explosion in Vessel Vent System	2×10^{-10}	13	3	6×10^{-2}	6×10^{-10}	1.2×10^{-12}
3.4.1.4	Nitrated Organic Explosion in Am Product Concentrator	6×10^{-10}	20	5	0.1	3×10^{-9}	6×10^{-11}
3.4.1.5	"Red Oil" Explosion in Process Concentrator	6×10^{-11}	6	1.4	3×10^{-2}	8×10^{-11}	1.8×10^{-13}
3.4.1.6	Resin Explosion in CEC Unit	9×10^{-12}	1300	3×10^2	6	3×10^{-9}	5×10^{-11}
3.4.1.7	Facility Solvent Fire	6×10^{-11}	5×10^2	4×10^2	8.	2×10^{-8}	5×10^{-10}
	Solvent Mist Explosion	2×10^{-10}	25	30	0.4	6×10^{-9}	8×10^{-11}
Total Risk						3.3×10^{-8}	7.8×10^{-10}

3.4.3 Occupational Radiological Risk of FFP-WTF

The estimated occupational exposure for the FFP-WTF is given in Table 3.4.17. The assumptions made in assessing these values are presented in Appendix D. The exposure is estimated to 40% of that for the FRP-WTF. The differences are attributable to reduced crew size and significant reduction in scope and complexity of operation.

Table 3.4.17
ESTIMATED OCCUPATIONAL EXPOSURE -- FFP-WTF

Activity	Person-Rem/Year
Routine Operations	90
Maintenance	90
Abnormal Occurrences*	3
Accidents	7×10^{-3}
Decommissioning	14
TOTAL	197

*Defined as personnel exposure exceeding limits set in 10CFR20

The summary of these values is 201 person-rem/plant-yr. Using the BEIR report equivalence of 1×10^{-4} fatalities/person-rem this is 3 person-rem/GWe-yr or 3×10^{-4} latent fatality/GWe-year.

3.4.4 Nonradiological Risks of the FFP-WTF

The chemical and industrial risks developed in Appendix D are summarized in Table 3.4.18. The primary chemical hazards are associated with the fraction of site power plant effluents attributable to utility consumption by the FFP-WTF. The health effects associated with these effluents completely dominate the non-radiological risk.

Table 3.4.18
SUMMARY EFFECTS -- NONRADIOLOGICAL CAUSES

Cause	Premature Deaths (Fatality/GWe-yr)	Economic Deaths (Fatality/GWe-yr)	Total (Fatality/GWe-yr)
Effluents	0.04	0.12	0.16
Industrial Accidents	1.4×10^{-4}	5.1×10^{-3}	5.2×10^{-3}
TOTAL			<hr/> 0.17

3.5 RISKS OF TRANSPORTATION

3.5.1 Risk of the Reference Cycle Spent Fuel Transportation

3.5.1.1 Overview and Preliminary Hazards Analysis

Transportation links a geographically dispersed fuel cycle to function as an integrated whole. While there is more transportation usage in the front end (prereactor), it poses very little radiological risk and is not investigated here. The steps that are treated are: (1) spent fuel transportation from power plant to FRP, (2) fuel powder transportation between FRP and FFP, (3) fresh fuel transport from FFP to nuclear power plants, and (4) solidified high level waste (HLW) cladding hulls and TRU wastes to a waste repository.

Table 3.5.1 shows the backend of the fuel cycle transportation links necessary to support 75 GWe of nuclear LWR power. The distances are nominal and taken to be the same for both the Reference and PT cycles.

The general approach of this section draws heavily on reference 36 and is similar to that of the preceding sections. It uses Preliminary Hazards Analyses to identify major risk areas, fault tree analyses to assist in calculating accident probabilities, physico-chemical calculations of material releases, and the use of the CRAC code (Appendix C) to determine doses and health effects. In the previous sections, fault trees were used to model individual accidents. This section adopts a slightly different approach by defining release categories based on the barriers that must fail before radiological effects can affect the public. These categories are: (1) failure of the neutron shield, (2) exposure of the inner fuel-containing cavity, (3) exposure of the inner cavity and 50% fuel damage and (4) exposure of the inner cavity, 50% fuel damage and fire. Fault trees are constructed to model each of these categories. Details of the barriers and of the National Lead Industries NLI 10/24 cask used to ship fresh fuel, spent fuel, and powder between the FRP and the FFP are presented in Appendix A.5.

The Preliminary Hazards Analyses for this cask are presented in Table 3.5.2. From these, dominant accident sequences are selected for fault tree modeling and presented in the following sections along with probabilistic quantification of the fault trees and estimated radiological releases for each release category.

Table 3.5.1

TRANSPORTATION LINKS IN THIS STUDY TO
SUPPORT A 75 GWe NUCLEAR POWER INDUSTRY*

Fuel Cycle	Shipping Point	Receiving Point	Package - Mode (Material)	Assumed Distance (miles)	Quantity/ year	Shipment/ year	Thousand Shipment miles/GWe-year
Reference	Power Plant	FRP	NLI 10/24 - Rail Spent Fuel	1,000	2,000 MT	440	5.9
	FFP	Power Plant	NLI 10/24 - Rail Fresh Fuel	1,000	660 MT	145	1.9
	FRP	FFP	NLI 10/24 - Rail Fuel Powder	0.63	660 MT	220	1.9×10^{-3}
	FRP	Repository	ORNL - Rail HLW	2,000	$113m^3$	20	0.5
	FRP	Repository	ORNL - Rail Clad Hulls	2,000	$534m^3$	460	12.3
	FRP	Repository	Super Tiger - Truck TRU	2,000	$1 \times 10^4 m^3$	1,190	32
PT	Power Plant	FRP	ORNL - Rail PT Spent Fuel	1,000	660 MT	245	3.3
	Power Plant	FRP	NLI 10/24 - Rail U Spent Fuel	1,000	1,333 MT	489	6.5
	FFP	Power Plant	ORNL - Rail Fresh Fuel	1,000	660 MT	242	3.2
	FRP	FFP	ORNL - Rail Fuel Powder	0.63	660 MT	367	3.1×10^{-3}
	FRP	Repository	ORNL - Rail HLW	2,000	$113m^3$	20	0.5
	FRP	Repository	ORNL - Rail Clad Hulls	2,000	$534m^3$	460	12.3
	FRP	Repository	Super Tiger - Truck TRU	2,000	$1 \times 10^4 m^3$	1,190	32

*Each fuel element contains 0.454 MTHM, 64 assemblies discharged/year-GWe, 33% recycle

Table 3.5.2

PRELIMINARY HAZARDS ANALYSIS REFERENCE CYCLE
SPENT FUEL TRANSPORT

Potential Accident	Cause	Barriers	Likelihood	Consequences	Risk Contribution	Effect of P-T Fuel Cycle
loss of neutron shield	failure to fill shield	radiological surveying of cask, inspections, cask body	possible	neutron source	direct dose limited to nearby vicinity of cask, hazard to fire, police and ambulance personnel	neutron moderator of P-T cask is in solid form, is kept in water tight segments and is protected by several outer structural layers, therefore, this hazard is essentially nonexistent
	shield water leak	capping of fill valving, inspection, cask body	has occurred once			
	loss of shield water due to deformation of cask during accident	structural outer rigidity of steel bracing between inner and outer shells, innershell & shielding	low (Sandia crash tests have not used water jacketed casks)			
	loss of shield water due to missile or stud penetration of outer shell during accident	heavy gauge steel of outer shell, inner shell and shielding	low			
opening to inner cavity	loss of containment integrity of head	inner and outer closure heads, used, no penetration from outside to cask cavity, multiple bolts and balsa impact limiters, intact fuel cladding	low	release of gaseous fission products present in cask cavity	hazard to public from inhalation	since closure head assemblies are assumed to be the same for both casks this event is equally likely
	cask deformation in accident	structural and flexural strength of cask design, intact fuel claddings	very low			higher strength of P-T cask makes this accident less likely in a P-T fuel cycle, smaller inventory reduces consequences

Table 3.5.2 (cont'd)

Potential Accident	Cause	Barriers	Likelihood	Consequences	Risk Contribution	Effect of P-T Fuel Cycle
gross failure of fuel mechanically	failure of cask head or body due to missile or stub contact as a result of accident	inner and outer containment shells, intact fuel claddings, limitation in the size of penetration	very low	release of gaseous fission products in cask cavity and fuel, some release of particulates	hazard to the public, particulates cause public exposure from inhalation and contamination	since cask has an extra layer of stainless steel the amount of protection against puncture and deformation is greater for the P-T cask. Also the volume of material inside is less. However the presence of added amounts of actinides renders specific inventories higher
	large scale cask deformation as a result of accident	structural rigidity of cask inner containment, vibration and impact resistant nature of assembly arrays, intact fuel cladding	very low			
gross failure of fuel with fuel overheating	penetration of cask cavity by missiles or stubs with subsequent immersion in a thermal environment	inner & outer containment shells, limitation in the size of penetration, plating out of semi-volatiles released from fuel claddings	extremely low	release of gaseous fission products in cask cavity and fuel, release of particulates, release of semi-volatiles from broken fuel	highest hazard to the public from inhalation and contamination	given a cask loss of integrity and explosive to a fire both casks are equally susceptible to
	large scale deformation of cask body or head as a result of accident with subsequent immersion in thermal environment	structural rigidity of cask, plating out of semi-volatiles released from fuel claddings	extremely low			

In these probabilistic calculations, one probability is used repeatedly; namely, the probability of a train car accident per mile of travel. This is given by Clarke, et al.⁽³⁷⁾ as 1.5×10^{-6} /car-mile. It should be noted that this is not the same as the probability of a train accident which is given by both references 38 and 39 as 10^{-5} /train-mile. The difference is that not all cars are affected in a train accident.

Each of the categories release a fraction of the radiation or radioactivity present in the cask. Therefore, it is convenient to tabulate the curie quantity of radioactivity present in the reference cask. This is presented in Table 3.5.3

3.5.1.2 Reference Spent Fuel Transportation Category 1: Loss of Neutron Shielding

Appendix B.4.1 shows the top of the Category 1 fault tree. This shows four major failure modes, three of which are developed in subfault trees. Because of the minor impact of Category 1 and the fact that the subfault trees are presented in reference 36, the fault trees will not be presented but only the results from updating the fault trees.

Leak from Metal Fracture (A2G). This could result from excessive pressure which may be due to intersection of events of double failure of active cooling, failure of pressure relief and insufficient heat transfer due to fire or air flow blockage. These are assessed as 1.4×10^{-13} /c-mi* and, hence, this is an unlikely cause of the loss of neutron shielding. Another cause could be cyclic fatigue of the neutron shield, which is assessed from boiler data⁽⁴⁰⁾ to be 4×10^{-9} /c-mi. Water freezing is an unlikely cause because of antifreeze and the internal heat generation.

Missile Penetration (A3G). This is analyzed as the union of missile or stub penetration of the water jacket. Appendix F, Table F.1, gives this probability as 1.5×10^{-8} /c-mi.

Water Jacket Boils Dry (A4G). The most likely cause of this would be an accident involving fire estimated from train experience to be 5×10^{-9} /c-mi.

* c-mi indicates container or cask mile

Table 3.5.3*

PRINCIPAL ISOTOPIC CURIE QUANTITIES IN
THE REFERENCE SPENT FUEL CASK

(Fifth Recycle Fuel Burned to 33,000 MWD/MT
with 150-Day Cooling)

Isotope	Curies	Isotope	Curies
^3H	3.3×10^3	^{238}Pu	1.6×10^5
^{85}Kr	3.1×10^4	^{239}Pu	2.2×10^3
^{90}Sr	2.5×10^5	^{240}Pu	6.3×10^3
^{90}Y	2.5×10^5	^{241}Pu	1.6×10^6
^{106}Ru	1.2×10^6		
^{106}Rh	1.2×10^6	^{241}Am	7.7×10^3
^{129}I	1.4×10^{-1}	^{242}Am	2.0×10^2
^{134}Cs	3.8×10^5		
^{137}Cs	4.3×10^5	^{242}Cm	9.1×10^4
^{144}Ce	1.3×10^6	^{244}Cm	1.0×10^5
^{144}Pr	1.3×10^6		
^{147}Pm	4.2×10^5		
^{154}Eu	4.1×10^4		
^{155}Eu	2.7×10^4		

*This table is prepared from ORIGEN data provided by A. Cross, ORNL, using 0.454 MTHM per PWR fuel element and 2.24×10^9 gm HM per 2,000 MTHM of annual FRP throughput for the blended fuel. (35)

Failure to Fill the Water Jacket (C4). While there is a case of a water jacket that emptied due to improper closure, this is believed to be very improbable (ϵ) because of the independent checkout procedures combined with the requirements to show that the cask satisfied transportation requirements in its bill of lading before shipping.

Category 1 summary is that the probability of losing neutron shielding is 2.4×10^{-8} /c-mi. Should this occur, neutrons will escape the cask. The ORIGEN calculations indicate the neutron source strength of the spent fuel in a reference cask is 1.6×10^{10} n/sec. Assuming no neutron absorption in the cask, this would result in an exposure of 0.13 rem/hr at 10 m and 1.3 mrem/hr at 100 m. Using 255 people/km² (the geometric mean of the three densities in reference 41) and one hour exposure time, this integrates to 8×10^{-7} person-rem.

3.5.1.3 Reference Spent Fuel Transportation Category 2: Exposure of the Inner Spent Fuel Containing Cavity

Appendix B.4.2 shows the top of the Category 2 fault tree. This shows 5 failure modes, two of which were developed in subfault trees but are omitted for brevity.

Fire-Caused Metal Fracture (C1). From thermal and stress analyses fire-caused metal fracture does not seem possible; hence, $C1 = \epsilon$.

Train Wreck-Caused Metal Fracture (C2). Casks have been impact tested up to speeds of 82 mph without fracture⁽⁴²⁾. Conservatively assuming that 80 mph was the threshold of fracture, the probability of impacting above 80 mph is the probability of fracture. From data in Appendix F, this probability appears to range from 2×10^{-2} to 2×10^{-7} , the geometric mean of which is 6×10^{-5} given an accident and 5×10^{-11} /c-mi.

Cyclic Fatigue (C3). This is evaluated using pressure vessel data. WASH-1318⁽⁴⁰⁾ gives a 99% confidence noncritical failure probability of 4×10^{-4} /vessel-year. Because of the ductility of the stainless steel used in the cask, it is estimated that for this application this should be 4×10^{-6} /trip or 4×10^{-9} /c-mi.

Improper Manufacture (C4). This was included in the cyclic fatigue data so $C4 = 0$.

Missile Penetration of Both Boundaries (B3G). This was evaluated in a subfault tree; however, the dominant contributor is due to a stub from wreckage debris impaling the cask. Appendix F gives the probability of puncturing all the way to the cask interior as 3×10^{-9} /c-mi.

Failure of Closure Seal (B6G). This subtree is also not presented; however, the dominant failure mode is that of improper installation which under low stress is 1×10^{-4} (reference 18). Assuming one task per trip, this is $1 \times 10^{-7}/\text{c-mi}$. However, there are redundant seals and the probability of a common mode error is 10^{-2} (reference 15) reducing the combined probability to $1 \times 10^{-9}/\text{c-mi}$, but these mistakes should be found when the cask is leak tested. The probability of failing to do so is estimated as 10^{-2} for a probability of $1 \times 10^{-11}/\text{c-mi}$.

This may also be estimated from gasket failure data as $1.6 \times 10^{-6}/\text{hr}$ ⁽⁴³⁾ corresponding to $1.1 \times 10^{-9}/\text{c-mi}$; however, there are two seals and the combined probability is estimated as the geometric mean of the independent and dependent failure rate or 3.6×10^{-11} .

The combined probability of these two failure modes is $B6G = 5 \times 10^{-11}/\text{c-mi}$.

The Category 2 summarized probability is the union of the preceding or $9 \times 10^{-9}/\text{c-mi}$.

Should this accident occur, it is estimated to release gases contained within the cavity from the leaking of 1% of the rods. This is estimated using Gaussian diffusion as resulting in releasing the following curie quantities: $^3\text{H} - 1.1 \times 10^{-4}$, $^{85}\text{Kr} - 5 \times 10^{-2}$, and $^{129}\text{I} - 1.4 \times 10^{-8}$. The population dose and health effects of these and the other accidents associated with reference spent fuel transportation are presented in Table 3.5.4 in Section 3.5.1.6.

3.5.1.4 Reference Spent Fuel Transportation Category 3: Exposure of the Inner Fuel Containing Cavity and 50% Fuel Damage

This failure mode is similar to the Category 2 except that the probe that penetrates the cask or the cask crushing results in extensive fuel damage (50%). This is depicted in Figure B.4.3.

Tornado Missile (C1). Thom⁽⁴⁴⁾ estimated the probability of a tornado striking a point in Des Moines, Iowa (one of the most frequent areas for tornadoes), is $10^{-4}/\text{yr}$. For a 72-hour trip duration, this gives a probability of $8 \times 10^{-10}/\text{c-mi}$. However, a very severe tornado would be required to generate a missile to penetrate the cask. Doan⁽⁴⁵⁾ estimates 1% of tornadoes reaching the design basis; hence, $C1 = 8 \times 10^{-12}/\text{c-mi}$.

Extrain Explosion (C2). Napadensky, et al.⁽⁴⁶⁾ give the rate of cargo explosions as $4.2 \times 10^{-8}/\text{train-mile}$. Conservatively assuming a one-mile interaction distance, only 0.1% of the trip is at risk, hence $4 \times 10^{-11}/\text{train-mile}$ is estimated

Table 3.5.4

PROBABILITIES, CONSEQUENCES AND RISK OF REFERENCE SPENT FUEL TRANSPORTATION

Accident Category	Probability (p)		Total Curies Released	Consequences (c)		Risk (pxc)					
				Total Person Rem	Total Latent Cancer	Person Rem/			Latent Cancer/		
	/c-mile	/trip				c-mile	trip	GWe-yr	c-mile	trip	GWe-yr
1	2×10^{-8}	2×10^{-5}	--	8×10^{-7}	4×10^{-11}	1.9×10^{-14}	1.9×10^{-11}	1.1×10^{-10}	1×10^{-18}	1×10^{-15}	6×10^{-15}
2	9×10^{-9}	9×10^{-6}	5×10^{-2}	1.7×10^{-6}	3×10^{-10}	1.5×10^{-14}	1.5×10^{-11}	9×10^{-11}	3×10^{-18}	3×10^{-15}	1.8×10^{-14}
3	4×10^{-10}	4×10^{-7}	5000	0.5	3×10^{-3}	2.0×10^{-10}	2.0×10^{-7}	1.1×10^{-7}	1×10^{-12}	1×10^{-9}	6×10^{-9}
4	3×10^{-12}	3×10^{-9}	5100	1700	0.19	5×10^{-9}	5×10^{-6}	3×10^{-5}	6×10^{-13}	6×10^{-10}	4×10^{-9}
Risk Total								3×10^{-5}			1×10^{-8}

as the probability. The probability of an explosion generating a missile that penetrates the cask is even less, so $C2 < 4 \times 10^{-11}/\text{c-mi}$.

Missile from Hazardous Cargo (C3). Regulations prohibit type "B" material in association with hazardous cargo, but it could result from human error taken to have a probability of $10^{-4}/\text{task}^{(9)}$ under low stress conditions. Napadensky⁽⁴⁶⁾ estimates that 1% of TNT shipment accidents result in some type of explosion; hence, $C3 = 1.5 \times 10^{-12}/\text{c-mi}$. It should be noted that this probability is an upper bound estimate that assumes the availability of explosive cargo to be inadvertently shipped with the cask. In fact, only a fraction of trains contain explosives.

Missile from Train Wreck (C4). Missiles that are both projectiles and impalement stubs are considered in Appendix F. From the distribution of train speeds, estimates of the probability of missile/stub presence and penetration probability, this is estimated as $3 \times 10^{-10}/\text{c-mi}$. This estimate contains a probability that 10% of the missiles of this penetrability causes the specified fuel damage.

Gross Crushing of Cask (C5). Based on the Sandia cask tests⁽⁴²⁾ and the failure to achieve any fuel or cask damage at extreme speeds into abutments, it is concluded that this probability is negligible compared with the others. $C5 = \epsilon$.

The Category 3 summarized accident probability is $4 \times 10^{-10}/\text{c-mi}$.

Should this occur with half of the fuel damaged, the releases are estimated to be 11 Ci of ^3H , 5×10^3 Ci of ^{85}Kr , 1.4×10^{-3} Ci of ^{131}I , and 0.56 Ci of nonvolatile fires.

These estimates are based on no retardation for unconfined noble gases and tritium, 50% plate-out for iodine release fraction of 7×10^{-8} for fires obtained by geometric-mean averaging the fuel rod burst test data from reference 39.

3.5.1.5 Reference Spent Fuel Transportation Category 4: Exposure of Severe Fuel With Severe Damage and Fire

This accident is very similar to the preceding except for the possibility of a severe sustained fire accompanying an accident that results in cask puncture and extensive fuel damage. This event is depicted in the fault tree Figure B.4.4. (It will be noted that the fault trees become simpler as the accident severity increases because of circumvention of redundant barriers.) Combining the

probability of gross cask damage (Category 3) with extensive fire accompanying a train wreck accident - 10^{-2} (reference 37) gives an estimated probability for Category 4 as 3×10^{-12} /c-mi.

Should this accident occur, it is estimated that the following quantities of material will be released: 11 Ci of ^3He , 5×10^3 Ci of ^{85}Kr , 2.8×10^{-3} Ci of ^{131}I , 53 Ci of Cs and 0.56 Ci of nonvolatiles (essentially material fines). These estimates are prepared in the same manner as Category 3. The high temperature affects the Cs release by the temperature exceeding the boiling point of Cs metal. It is, however, conservative because Cs is chemically active and the temperature does not reach the boiling point of the compounds. This estimate also assumes no iodine plate-out because of the temperature.

3.5.1.6 Reference Spent Fuel Transportation Risk Summary

These results are summarized in Table 3.5.4.

3.5.2 Risks of PT Spent Fuel Transportation

3.5.2.1 Overview and Preliminary Hazards Analysis

In the previous reference case, it was assumed that the cask was one-third filled with spent recycled mixed-oxide fuel and two-thirds filled with spent uranium fuel (no recycle). In the present PT case, because of the very high neutron emission rate of the PT fuel, it is assumed that the PT MOX fuel is shipped in a specially designed cask called the PT Cask. The remaining two-thirds of the fuel that entered the reactor contained only uranium and is shipped in the Reference Cask (NLI10/24). In all cases cask shielding will be such that regulatory requirements⁽⁴¹⁾ such as 10 m-rem/hr 3 ft from an external surface will be met.

Information on the ORNL cask is presented in Appendix A.5.2. It will be noted that it contains no liquid moderator hence category 1 of the Reference cask is not possible for this case.

Table 3.5.5 presents a preliminary hazard analysis for spent fuel shipped in the PT cask. It will be noted that the failure modes are qualitatively similar to the Reference cask but quantitatively the PT cask is more accident resistant due to the 2-inch thick outer stainless steel jacket as well as the longer distance from the outside to the interior thereby increasing the collapse probability of penetrating probes.

The PT cask contains 6 PWR fuel elements as compared with the 10 fuel elements in the reference cask. This is reflected in Table 3.5.6 which presents the curie quantities of material in cask and in Table 3.5.7 which presents the curie quantities in the Reference cask.

3.5.2.2 PT Spent Fuel Transportation Category 1: Loss of Neutron Shielding

Because the cask uses lithium hydride as a moderator/absorber instead of water, the loss of the neutron shielding is very unlikely. This judgement is reinforced by the use of other types of neutron shielding (e.g., B_4C) which would be even more difficult to accidentally lose. Thus for PT MOX transportation, the probability of Category 1 accident is very small (ϵ).

Table 3.5.5

PRELIMINARY HAZARD ANALYSIS
PT CYCLE SPENT FUEL TRANSPORT

POTENTIAL ACCIDENT	CAUSE	BARRIERS	LIKELIHOOD	RADIOLOGICAL CONCERNS	CONSEQUENCE	EFFECT OF P-T FUEL CYCLE
opening to inner cavity	loss of containment integrity of head	inner and outer closure heads used, no penetration from outside to cask cavity, multiple bolts and balsa impact limiters, intact fuel cladding	low	release of gaseous fission products in cask cavity	hazard to public from inhalation	since closure head assemblies are assumed to be the same for both casks, this event is as likely for the P-T as for the reference MOX cycle
	cask deformation in accident	structural flexural strength of cask design, intact fuel claddings	very low			higher strength of P-T cask makes this accident less likely in a P-T fuel cycle, smaller inventory reduces consequences
gross failure of fuel mechanically	failure of cask head or body due to missile stub contact as a result of accident	inner and outer containment shells, intact fuel claddings, limitation in size of penetration	very low	release of gaseous fission products in cask cavity and fuel, some release of particulates	hazard to the public, particulates cause public exposure aside from inhalation	since cask has an extra layer of stainless steel, the amount of protection against puncture and deformation is greater for the P-T cask. Also, the volume of material inside is less. However, the presence of added amounts of actinides renders specific inventories higher
	large scale cask deformation as a result of accident	structural rigidity of cask inner containment, vibration and impact resistant nature of assembly arrays, intact fuel cladding	very low			
gross failure of fuel with fuel over-heating	penetration of cask cavity by missiles or stubs with subsequent immersion in a thermal environment	inner and outer containment shells, limitation in the size of penetration, plating out of semi-volatiles released from fuel claddings	extremely low	release of gaseous fission products in cask cavity and fuel, release of particulates, release of semi-volatiles from broken fuel	highest hazard to the public from inhalation and contamination	given a cask loss of integrity and explosive to a fire both casks are equally susceptible to
	large scale deformation of cask body or head as a result of accident with subsequent immersion in thermal environment	structural rigidity of cask, plating out of semi-volatiles released from fuel claddings	extremely low			

Table 3.5.6*

PRINCIPAL ISOTOPIC CURIE QUANTITIES
IN THE PT SPENT FUEL CASK

<u>Isotope</u>	<u>Curies</u>	<u>Isotope</u>	<u>Curies</u>
^3He	1.8×10^3	^{238}Pu	2.5×10^5
^{85}Kr	1.5×10^4	^{239}Pu	3.9×10^3
^{90}Sr	1.1×10^5	^{240}Pu	1.1×10^4
^{106}Ru	1.2×10^6	^{241}Pu	2.8×10^6
^{106}Rh	1.2×10^6	^{242}Pu	86
^{129}I	0.11		
^{134}Cs	3.0×10^5	^{241}Am	1.4×10^4
^{137}Cs	3.1×10^5		
^{144}Ce	8.7×10^5	^{242}Cm	2.4×10^5
^{144}Pu	8.7×10^5	^{244}Cm	1.0×10^6
^{147}Pm	2.7×10^5	^{245}Cm	3.9×10^2
^{154}Eu	4.0×10^4	^{246}Cm	2.9×10^2
^{155}Eu	3.0×10^4		

*This table is prepared from ORIGIN PT MOX data provided by A. Croff (ORNL) using 0.454 MTHM per PWR fuel element and 7.63×10^8 gm HM per 2,000 MTHM of annual FRP throughput.(35)

Table 3.5.7*

PRINCIPAL ISOTOPIC CURIE QUANTITIES
IN THE REFERENCE CASK CARRYING URANIUM SPENT FUEL

<u>Isotope</u>	<u>Curies</u>	<u>Isotope</u>	<u>Curies</u>
^3He	2.1×10^3	^{238}Pu	9.7×10^3
^{85}Kr	3.5×10^4	^{239}Pu	1.3×10^3
^{90}Sr	2.9×10^5	^{240}Pu	2.2×10^3
^{106}Ru	8.5×10^5	^{241}Pu	4.8×10^5
^{106}Rh	8.5×10^5	^{242}Pu	7.2
^{129}I	0.13		
^{134}Cs	3.9×10^5	^{241}Am	1.6×10^3
^{137}Cs	4.1×10^5		
^{144}Ce	1.4×10^6	^{242}Cm	1.6×10^4
^{144}Pr	1.4×10^6	^{244}Cm	5.9×10^3
^{147}Pm	3.8×10^5		
^{154}Eu	3.8×10^4		
^{155}Eu	2.1×10^4		

*This table is prepared from ORIGEN uranium fuel data provided by A. Croff (ORNL) using 0.454 MTHM per PWR fuel element and 1.48×10^9 gm HM per 2,000 MTHM of annual FRP throughput. (35)

This is not the case for the Reference Cask shipment of uranium fuel which in 3.5.1.3 was estimated to have a probability of 2.4×10^{-8} c-mi and results in exposure of a neutron strength of 4.2×10^7 n/sec and could result in 0.3 p-rem/hr at 10 m.

3.5.2.3 PT Spent Fuel Transportation Category 2: Exposure of the Inner Spent Fuel Containing Cavity

It was seen in Section 3.5.1.3 that the major failure modes of the Reference Cask were through puncture or metal fatigue. The PT cask is even more immune to secondary failure modes and the same probability of cyclic fatigue failure for the Reference Cask which is 4×10^{-9} /c-mi (event B3, Section 3.5.1.3) is used here. The probability of puncture is 1.1×10^{-9} /c-mi for a combined Category 2 probability of 5.1×10^{-9} /c-mi. Should this occur, the releases are estimated to be 1.1×10^{-8} Ci of I.

The Category 2 probability for the Reference Cask carrying uranium fuel is given in Section 2.5.1.3 as 9×10^{-9} /c-mi and would result in release of 1.1×10^{-4} Ci of ^3H , 5×10^{-2} Ci of ^{85}Kr , and 1.3×10^{-8} Ci of I.

3.5.2.4 PT Spent Fuel Transportation Category 3: Exposure of the Inner Fuel-Containing Cavity and 50% Fuel Damage

This evaluation proceeds in the same manner as 3.5.1.4 except that the penetration probability is modified according to Table F.1 in Appendix F. The probability of PT Category 3 is 1.2×10^{-10} /c-mi. Should this occur, the estimated releases are 9 Ci of ^3H , 2×10^3 Ci of ^{85}Kr , 1.1×10^{-3} of I, and 0.63 Ci of non-volatile particulates.

The probability of a Category 3 accident with a Reference Cask containing uranium fuel is given in Section 3.5.1.5 as 3×10^{-10} /c-mi and the estimated curie releases are 11 of ^3H , 5×10^3 of ^{85}Kr , 1.3×10^{-3} of I, and 0.45 of non-volatile particulates.

3.5.2.5 PT Spent Fuel Transportation Category 4 : Exposure of Fuel With Severe Damage and Fire

This probability is the previous value reduced by 0.01 to account for the probability of severe fires accompanying train wreckage. Thus the probability of PT Category 4 is 1.2×10^{-12} /c-mi and the releases are 9 Ci of ^3H , 2×10^3 Ci of ^{85}Kr , 2.2×10^{-3} Ci of I, 45 Ci of Cs and 0.63 Ci of non-volatile particulates.

The probability of a Category 4 Reference Cask accident involving uranium fuel is 3×10^{-12} /c-mi and the curie releases are 11 of ^3H , 5×10^3 of ^{85}Kr , 2.6×10^{-3} of I, 53 of Cs, and 0.45 of non-volatile particulates.

3.5.2.6 PT Spent Fuel Transportation Risk Summary

These results are summarized in Table 3.5.8.

Table 3.5.8

PROBABILITIES, CONSEQUENCES AND RISK OF PT SPENT FUEL TRANSPORTATION

							Risk (pxc)					
Transportation Step	Accident Category	Probability (r)		Curie Released	Consequences		Person-rem/			Latent cancer/		
		/c-mile	c/trip		Per-rem	Latent Cancer	c-mile	Trip	GWe/yr	c-mile	Trip	GWe/yr
Spent PT MOX	1	-	-	-	-	-	-	-	-	-	-	-
	2	5×10^{-9}	5×10^{-6}	2×10^{-2}	7×10^{-7}	1.4×10^{-10}	4×10^{-15}	4×10^{-12}	1.3×10^{-11}	7×10^{-11}	7×10^{-16}	2×10^{-15}
	3	1.2×10^{-10}	1.2×10^{-7}	2×10^3	0.5	6×10^{-3}	7×10^{-11}	7×10^{-8}	2×10^{-7}	7×10^{-13}	7×10^{-10}	2×10^{-9}
	4	1.2×10^{-12}	1.2×10^{-9}	2×10^3	1.4×10^3	0.16	1.7×10^{-9}	1.7×10^{-6}	6×10^{-6}	1.9×10^{-13}	1.9×10^{-10}	6×10^{-10}
Risk Sub-total									6×10^{-6}			2×10^{-9}
Spent PT Uranium Fuel	1	2×10^{-8}	2×10^{-5}	-	1.8×10^{-9}	9×10^{-14}	4×10^{-17}	4×10^{-14}	3×10^{-13}	2×10^{-21}	2×10^{-18}	1.3×10^{-17}
	2	9×10^{-9}	9×10^{-6}	5×10^{-2}	1.7×10^{-6}	3×10^{-10}	1.5×10^{-14}	1.5×10^{-11}	1.0×10^{-11}	2×10^{-18}	2×10^{-15}	1.3×10^{-14}
	3	3×10^{-10}	3×10^{-7}	5×10^3	0.4	6×10^{-4}	1.1×10^{-10}	1.1×10^{-7}	7×10^{-7}	1.9×10^{-13}	1.9×10^{-10}	1.2×10^{-9}
	4	3×10^{-12}	3×10^{-9}	5×10^3	1.7×10^3	0.18	5×10^{-9}	5×10^{-6}	3×10^{-5}	5×10^{-13}	5×10^{-10}	3×10^{-9}
Risk Sub-total									3×10^{-5}			4×10^{-9}
PT Risk Total									4×10^{-5}			6×10^{-9}

3.5.3 Risk of Powder Transportation

3.5.3.1 Risk of Reference Powder Transportation

The powder for use in fabricating new fuel rods is conceptually transported between the FRP to the FFP in the same cask used for shipping spent and fresh fuel. No casks have been designed for transporting coprocessed fifth cycle MOX or PT fuel powder. Hence, while this is perhaps overly conservative, it is nevertheless necessary to provide shielded transport. The Reference fuel powder cask is basically an NLI 10/24 spent fuel cask containing 3 MTHM of powder in an inner shell container that is subdivided into 40 kg compartments. The battery electric powdered railway car makes a 1 km trip/day at slow speeds between the two plants and is loaded and unloaded remotely in an atmospherically controlled environment at each plant.

The preliminary hazards analysis that is applicable is that of Table 3.5.2 except that the slow speed (< 10 mph) and absence of other traffic makes the events more improbable than in Section 3.5.1. Because of the on-site control of other transportation and even the ability to provide barriers along the track right of way, no site-related accident mode capable of releasing material has been found. The only remaining credible mode is an aircrash into the cask. Using data from 3.3.1.3 for a cross section of 4.6 m^2 , the probability of penetration is estimated to be 2×10^{-12} /year where it has been assumed that the cask only contains powder 0.25 of the time.

Should this accident occur, one of the 40 kg compartments could be damaged, thereby releasing the powder. Using the same 0.2% release fraction as was used in Section 3.3.1.3, it is estimated that 80 gm of material could be released.

To summarize, the probability of an accidental release of Reference fuel powder is 2×10^{-12} /year resulting in an 80 Ci release, causing an exposure of 120 person-rem, 1.6 latent cancers with a risk of 5×10^{-10} person-rem/year 6×10^{-12} latent cancer/year.

3.5.3.2 Risk of PT Powder Transportation

Again the aircrash into the cask seems to be the only accident that can result in sufficient force to damage the cask. (The same probability and release fraction as in Section 3.5.3, but the isotopic inventory is different (see Table 3.2.1)).

The probability is 2×10^{-12} /year with a 108 Ci release causing an exposure of 140 person-rem, 2 latent cancers, with a risk of 6×10^{-10} person-rem/year and 8×10^{-12} latent cancer/year.

3.5.3.3 Summary of Powder Transport Risk

Table 3.5.9 summarizes the risk from fuel powder transport for both the Reference and PT cycles.

3.5.4 Risk of Fresh Fuel Transportation

Methods for transporting radioactive fresh fuel from the fabrication plant to the power plant have not been devised. For purposes of this study the very conservative assumption is made that it is shipped in the same casks that ship spent fuel. Thus the Reference Cask is assumed to be used for fresh Reference fuel and the P-T cask for fresh P-T fuel. Because these casks traverse the same routes and experience the same accident environments as when they are carrying spent fuel, the accident analyses of Sections 3.5.1 and 3.5.2 are applicable except for difference in fuel composition. Notable differences are the absences of the fission products ^3H , ^{85}Kr , I, Cs, and various non-volatile particulates. Table 3.5.10 summarizes the major isotopic quantities of each cask.

3.5.4.1 Risk of Reference Fresh Fuel Transportation

The Category 1, loss of neutron shield probability is 2.4×10^{-8} /c-mi and release 1.1×10^9 n/sec resulting in 9 m-rem/hr at 10 m. and a population dose of 6×10^{-8} p-rem.

Category 2 does not present a significant risk for this fuel.

Category 3, severe fuel damage with release, has a probability of 3×10^{-10} /c-mi and could result in a release of 0.96 Ci of non-volatile particulates.

Table 3.5.9

PROBABILITIES, CONSEQUENCES AND RISK OF FUEL POWDER TRANSPORT

Transportation Step	Accident Category	Risk pxc)										
		Probability (r)		Curie Released	Consequences		Person-rem/			Latent Cancer/		
		/c-mile	c/trip		Per-rem	Latent Cancer	c-mile	Trip	GWe/yr	c-mile	Trip	GWe/yr
Reference Powder	1	5×10^{-15}	7×10^{-15}	80	1.2×10^2	1.6	6×10^{-13}	8×10^{-13}	3×10^{-12}	8×10^{-15}	1.1×10^{-14}	4×10^{-14}
Reference Powder Risk Total									3×10^{-12}			4×10^{-14}
PT Powder	1	5×10^{-15}	7×10^{-15}	108	1.4×10^2	2	7×10^{-13}	1.0×10^{-12}	4×10^{-12}	1×10^{-14}	1.4×10^{-14}	5×10^{-14}
PT Powder Risk Total									4×10^{-12}			5×10^{-14}

Table 3.5.10*

PRINCIPAL ISOTOPIC CURIE QUANTITIES OF
FRESH FUEL CONTAINED IN THE REFERENCE AND P-T CASKS

<u>ISOTOPE</u>	<u>REFERENCE CASK</u>	<u>P-T CASK</u>
^{220}Rn	0.14	9.0×10^{-2}
^{238}Pu	4.1×10^5	2.1×10^5
^{239}Pu	6.8×10^3	4.3×10^3
^{240}Pu	1.7×10^4	1.1×10^4
^{241}Pu	4.4×10^6	2.5×10^6
^{242}Pu	1.3×10^2	1.1×10^2
^{241}Am	7.3×10^3	1.4×10^4
^{242}Cm	-	2.5×10^5
^{244}Cm	-	7.7×10^5
TOTAL	4.8×10^6	3.7×10^6

* This table is prepared from Origins Reference and P-T Fresh Fuel Data provided by A. Croff (ORNL) using 0.454 MTHM per PWR fuel element and 7.92×10^8 gm HM per 2,000 THM of annual FRP throughput. (34)

Category 4, severe fuel damage with release and fire, has a probability of 3×10^{-12} /c-mi and the same release.

3.5.4.2 Risk of PT Fresh Fuel Transportation

Category 1 does not present a significant risk for this cask.

Category 2 does not present a significant risk for this fuel.

Category 3, severe fuel damage with release, has a probability of 1.2×10^{-10} /c-mi and could result in a release of 0.74 Ci of non-volatile particulates.

Category 4, severe damage with release and fire, has a probability of 1.2×10^{-12} /c-mi and could result in the same release.

3.5.4.3 Fresh Fuel Transportation Risk Summary

The probabilities, consequences, and risk associated with fresh fuel transportation for both cycles is presented in Table 3.5.11.

3.5.5 Transportation of Fuel Element Structural Materials and Cladding Hulls

Structural material and cladding hulls from the FRP must be shipped to the waste repository in a shielded cask. Presently there are no licensed casks for this purpose and in this absence the ORNL conceptual design will be used as was done in Reference 36. This cask has a 2-inch stainless steel outer shell encasing 4.9 inches of lead with a 0.24-inch thick stainless steel liner for handling purposes. The volume of this container is 0.43 m^3 , corresponding to the processing of 2 MTHM of fuel.

Table 3.5.12 presents a preliminary hazards analysis for the rail shipment of SM in this conceptual cask. It should be noted that the cask has no neutron shielding that can be lost; however, it may be possible to lose the lead shielding in a sustained fire. The worst accident would result from a zirconium fire that released the semivolatiles and particulates that are bound to the structural materials. From the preliminary hazards analysis, four accident categories are defined: Category 1 is a fire-caused loss of lead shielding, Category 2 is a penetration to the interior resulting in a release of fines,

Table 3.5.11
PROBABILITIES, CONSEQUENCES AND RISK OF FRESH FUEL SHIPMENT

Transportation Step	Accident Category	Risk pxc)										
		Probability (r)		Curie Released	Consequences		Person-rem/			Latent Cancer/		
		/c-mile	c/trip		Per-rem	Latent Cancer	c-mile	Trip	GWe/yr	c-mile	Trip	GWe/yr
Reference Fresh Fuel Shipment	1	2×10^{-8}	2×10^{-5}	-	6×10^{-8}	3×10^{-12}	1.2×10^{-15}	1.2×10^{-12}	2×10^{-12}	6×10^{-20}	6×10^{-17}	1.1×10^{-16}
	2	-	-	-	-	-	-	-	-	-	-	-
	3	3×10^{-10}	3×10^{-7}	0.96	1.4	1.9×10^{-2}	4×10^{-10}	4×10^{-7}	8×10^{-7}	6×10^{-12}	6×10^{-9}	1.1×10^{-8}
	4	3×10^{-12}	3×10^{-9}	0.96	1.5	2×10^{-2}	5×10^{-12}	5×10^{-9}	1.0×10^{-8}	6×10^{-14}	6×10^{-11}	1.1×10^{-10}
Reference Fresh Fuel Shipment Risk									8×10^{-7}			1.1×10^{-8}
PT Fresh Fuel Shipment	1	-	-	-	-	-	-	-	-	-	-	-
	2	-	-	-	-	-	-	-	-	-	-	-
	3	1.2×10^{-10}	1.2×10^{-7}	0.74	1.0	1.4×10^{-2}	1.2×10^{-10}	1.2×10^{-7}	4×10^{-7}	1.7×10^{-12}	1.7×10^{-9}	5×10^{-9}
	4	1.2×10^{-12}	1.2×10^{-9}	0.74	0.9	1.3×10^{-2}	1.1×10^{-12}	1.1×10^{-9}	4×10^{-9}	1.6×10^{-14}	1.6×10^{-11}	5×10^{-11}
PT Fresh Fuel Shipment Risk									4×10^{-7}			5×10^{-9}

Table 3.5.12

PRELIMINARY HAZARDS ANALYSIS OF THE STRUCTURAL MATERIALS AND CLADDING HULLS CASK

Potential Accident	Cause	Barriers	Likelihood	Consequences	Effect of PT Fuel Cycle
Loss of shielding	Fire	Few fires are sufficiently sustained	Low	Gamma exposure of public and workers	About the same as Reference cycle
Venting of material to atmosphere	Collision puncture, metal fatigue	Cask plus inner liner	Low	Very low due to absence of volatiles	About the same as Reference
Release of fires to atmosphere	Collision, puncture, and fire ignition of zirconium	Cask plus inner liner zirconium is difficult to ignite	Very low	Low	Very considerable reduction of actinide risk contribution

Category 3 is a penetration to the interior resulting in extensive cannister and hull damage, and Category 4 is a penetration to the interior accompanied by fire to release material attached to the hulls. The principal isotopic content of the cask is presented in Table 3.5.13.

3.5.5.1 Reference Cycle Clad Transport Category 1: Loss of Lead Shielding

The probability of train accident is 1.5×10^{-6} /c-mi and severe fire accompanies 1% of these accidents⁽³⁷⁾; hence, the estimated Category 1 probability is 1.5×10^{-8} /c-mi. (This assumes that melting the lead will result in expansion sufficient to cause metal fracture with subsequent molten lead release.) Should this occur, neglecting the attenuation of the remaining cask materials, the exposed gamma flux would be 4×10^{11} MeV/sec in the energy range 0.015 to 11 MeV. Using the conversion factor for 600 MeV/cm²-sec photo energy flux as being approximately 1 p-rem/hour⁽⁴⁷⁾, the dose rate at 10 m is 53 p-rem/hour. With a population density of 255/km², the population dose is 0.034 person-rem/hour.

3.5.5.2 Reference Cycle Clad Transport Category 2: Release From the Cask Interior

The accident poses little risk because of the absence of volatile material in the cask.

3.5.5.3 Reference Cycle Clad Transport Category 3: Release From the Cask With Extensive Canister and Hull Damage

The probability of this type of puncture is 1×10^{-8} /c-mi however this must be reduced by 0.1 to account for the frequency of missiles of this penetrability. Hence the Clad Category 3 accident probability is 1×10^{-9} /c-mi.

Should this occur, the estimated release fraction is 7×10^{-8} -- the same as used in section 3.5.1.4 for an estimated non-volatile release of 5×10^{-5} curie.

3.5.5.4 Reference Cycle Clad Transport Category 3: Release From the Cask, Extensive Canister and Hull Damage and Fire

The probability of this event is the same as Category 3 reduced by the fire probability: 1×10^{-2} /wreck for an overall Category 4 probability of 1×10^{-11} /c-mi.

Table 3.5.13

PRINCIPAL ISOTOPIC CURIE QUANTITIES IN THE STRUCTURAL
MATERIALS AND CLADDING HULLS CASK

Isotope	Reference Cycle	PT Cycle
^{14}C	3×10^{-15}	3×10^{-15}
^{90}Sr	25	25
^{90}Y	25	25
^{95}Zr	2	2
^{95}Nb	4	4
^{106}Ru	117	124
^{106}Rh	117	124
^{125}Sb	5	5
$^{125\text{m}}\text{Tr}$	1	1
^{129}I	1.5×10^{-8}	1.5×10^{-8}
^{134}Cs	37	43
^{137}Cs	43	44
$^{137\text{m}}\text{Ba}$	41	42
^{144}Ce	133	137
^{144}Pr	133	137
$^{144\text{m}}\text{Pr}$	133	137
^{147}Pm	41	41
^{154}Eu	4	5
^{155}Eu	3	3
^{233}Pa	2	2
^{238}Pu	77	7
^{239}Pu	1	0.1
^{240}Pu	3	0.3
^{241}Pu	784	81
^{241}Am	4	0.4
^{242}Cm	45	7
^{244}Cm	49	29
Total	1,690	891

Should this occur, the non-volatile estimated release is 5×10^{-5} curie. The high temperature due to the fire could result in 50% release of cesium hence 40 Ci.

3.5.5.5 PT Cycle Clad Transportation Risk

The only difference between Reference and P-T Cycle clad transportation is in the clad inventory. The probabilities are the same.

The Category 1 consequences are the same (within the round-off error). Category 3 consequences are 6×10^{-5} Ci of non-volatiles. Category 4 consequences are 6×10^{-5} Ci of non-volatiles and 44 Ci of cesium.

3.5.5.6 Summary of Fuel Element Structural Material and Cladding Hulls Transport

The risks of this transportation mode are summarized in Table 3.5.14.

3.5.6 Risk of Solidified High Level Waste Transport

No casks have been licensed for solidified high level waste transport. For this study, it is assumed that the Reference Cask (NLI 10/24 - Appendix F) is modified for this purpose through the use of an aluminum cylinder filling the inner cavity. This cylinder is bored to accept 9 vitrified high level waste canisters. The analysis presented here closely follows that of Reference 36. It is assumed that each waste canister is composed of 25 w/o 10 year cooled high level waste (primarily fission product waste from the HA column) and 75 w/o borosilicate glass. The glass cylinders are contained within a 3/8 inch thick welded stainless steel wall. Each canister has a heat output of 3 kW and weighs 1257 lbs., contains the waste from processing 10 MT of fuel and occupies 0.63 m^3 .

Table 3.5.15 presents lists of the curie activity of the major waste constituents in the Reference and PT fuel cycle. The preliminary hazards analysis presented in Table 3.5.2 is applicable in this analysis.

3.5.6.1 Risk of Solidified High Level Reference Waste Transport

Category 1 as evaluated in Section 3.5.1.2 has a probability of 2.4×10^{-8} /c-mi and could result in neutron emission of 9×10^{11} n/sec resulting in an exposure of 7 rem/hr at 10 m and a population dose of 5×10^{-5} person-rem for one hour exposure.

Table 3.5.14

PROBABILITIES, CONSEQUENCES AND RISK OF FUEL ELEMENT STRUCTURAL MATERIAL AND CLADDING HULLS TRANSPORT BY RAIL

Transportation Step	Accident Category	Risk pxc)										
		Probability (r)		Curie Released	Consequences		Person-rem/			Latent Cancer/		
		/c-mile	c/trip		Per-rem	Latent Cancer	c-mile	Trip	GWe/yr	c-mile	Trip	GWe/yr
Reference Cycle Structural Material and Cladding Hulls	1	1.5×10^{-8}	3×10^{-5}	-	3×10^{-7}	1.6×10^{-11}	5×10^{-15}	9×10^{-12}	6×10^{-11}	2×10^{-19}	5×10^{-16}	2×10^{-15}
	2	-	-	-	-	-	-	-	-	-	-	-
	3	1×10^{-9}	2×10^{-6}	6×10^{-5}	6×10^{-5}	7×10^{-7}	6×10^{-14}	1.2×10^{-10}	7×10^{-10}	7×10^{-16}	1.4×10^{-12}	9×10^{-12}
	4	1×10^{-11}	2×10^{-8}	40	1.3×10^3	0.14	1.3×10^{-8}	3×10^{-5}	1.6×10^{-4}	1.4×10^{-12}	3×10^{-9}	1.7×10^{-8}
Reference Cycle Structural Material etc. Risk									1.6×10^{-4}			1.7×10^{-8}
PT Cycle Structural Material and Cladding Hulls	1	-	-	-	-	-	-	-	-	-	-	-
	2	-	-	-	-	-	-	-	-	-	-	-
	3	1×10^{-9}	2×10^{-6}	6×10^{-5}	3×10^{-5}	2×10^{-7}	3×10^{-16}	6×10^{-11}	4×10^{-12}	2×10^{-16}	4×10^{-13}	2×10^{-12}
	4	1×10^{-11}	2×10^{-8}	40	1.4×10^3	0.15	1.4×10^{-8}	3×10^{-5}	1.7×10^{-4}	1.5×10^{-12}	3×10^{-9}	1.8×10^{-8}
PT Cycle Structural Material etc. Risk									1.7×10^{-4}			1.8×10^{-8}

Table 3.5.15

CURIE ACTIVITY OF THE MAJOR ISOTOPIC CONSTITUENTS IN
A VITRIFIED HLW CANISTER (AGED 10 YRS)

<u>ISOTOPE</u>	<u>REFERENCE CYCLE</u>	<u>P-T CYCLE</u>
^{90}Sr	4.0×10^5	4.0×10^5
^{90}Y	4.0×10^5	4.0×10^5
^{106}Ru	2.0×10^5	2.0×10^5
^{106}Rh	2.0×10^5	2.0×10^5
^{125}Sb	4.0×10^4	4.0×10^4
$^{125\text{m}}\text{Te}$	2.0×10^4	2.0×10^4
^{134}Cs	3.0×10^5	3.0×10^5
^{137}Cs	1.0×10^6	1.0×10^6
$^{137\text{m}}\text{Ba}$	9.0×10^5	9.0×10^5
^{144}Cs	8.0×10^4	8.0×10^4
^{144}Pr	8.0×10^4	8.0×10^4
^{147}Pm	3.0×10^5	3.0×10^5
^{151}Sm	1.6×10^4	1.6×10^4
^{154}Eu	8.0×10^4	8.0×10^4
^{155}Eu	1.4×10^4	1.4×10^4
^{238}Pu	1.3×10^4	3.0×10^2
^{239}Pu	4.0×10^4	8.0×10^2
^{240}Pu	4.0×10^2	8
^{241}Pu	2.0×10^4	4.0×10^2
^{241}Am	7.0×10^3	0.7
^{243}Am	5.0×10^3	0.5
^{242}Cm		
^{244}Cm	7.0×10^5	70
TOTAL	4.9×10^6	4.1×10^6

Category 2 (release of the inner cavity atmosphere) does not present a significant risk for solidified waste because of the lack of volatile materials.

Category 3 (release and extensive canister damage) has a probability of 3×10^{-10} /c-mi (Section 3.5.1.4). Should this accident occur, the release from a single damaged canister is estimated to be 30 Ci of non volatile fines.

Category 4 (release, extensive canister damage and fire) has a probability of 3×10^{-12} /c-mi (Section 3.5.1.5). The release is estimated to be 30 Ci of non-volatile fines and 100 Ci of cesium.

3.5.6.2 Risk of Solidified High Level PT Waste Transport

Category 1 has an estimated probability of occurrence of 2.4×10^{-8} /c-mi but because of large reduction of actinides in the waste, the neutron emission is 9×10^7 n/sec resulting in an exposure of 0.7 m rem/hr at 10 m and a population dose of 5×10^{-9} person-rem for one hour exposure.

Category 2 does not present a significant risk for solidified waste because of the lack of volatile materials.

Category 3 has a probability of 3×10^{-10} /c-mi (Section 3.5.1.4) with an estimated release of 25 Ci of non-volatile fires.

Category 4 has a probability of 3×10^{-12} /c-mi (Section 3.5.1.5) with an estimated release of 25 Ci of non-volatile fires and 100 Ci of cesium.

3.5.6.3 Summary of Risks of High Level Waste Transport

Table 3.5.1.16 presents risk summaries of these steps.

3.5.7 Risks of Transportating Non-High Level Waste by Truck

Some wastes generated by the FRP, FFP, and the case of the PT cycle the WTF's do not radiate sufficiently to require shielded casks for transport. Some of these wastes are immobilized in concrete. Included in this category are ^{129}I , ^{14}C , and incinerated wastes amounting to $126 \text{ m}^3/\text{yr}$ (6%). The other category is non-immobilized TRU wastes amounting to $1954 \text{ m}^3/\text{year}$ (94%) and consisting of non-reusable equipment. These items are shipped in sealed 55 gallon drums which are not Transportation Class B. To satisfy shipping requirements, these are contained in an overpack which does satisfy Class B requirements. A description of the overpack is provided in Reference 36 which also provides the general form of the analysis presented here.

Table 3.5.16

PROBABILITIES, CONSEQUENCES AND RISK OF HIGH LEVEL WASTE TRANSPORT

Risk pxc)												
Transportation Step	Accident Category	Probability (r)		Curie Released	Consequences		Person-rem/			Latent Cancer/		
		/c-mile	c/trip		Per-rem	Latent Cancer	c-mile	Trip	GWe/yr	c-mile	Trip	GWe/yr
Reference Cycle HLW Transport	1	2×10^{-8}	5×10^{-5}	-	5×10^{-5}	3×10^{-9}	1×10^{-12}	3×10^{-9}	5×10^{-10}	6×10^{-17}	1.5×10^{-13}	3×10^{-14}
	2	-	-	-	-	-	-	-	-	-	-	-
	3	3×10^{-10}	6×10^{-7}	30	30	0.2	9×10^{-9}	1.8×10^{-5}	5×10^{-6}	6×10^{-11}	1.2×10^{-8}	3×10^{-8}
	4	3×10^{-12}	6×10^{-9}	130	3×10^3	0.6	9×10^9	1.8×10^{-5}	5×10^{-6}	1.8×10^{-11}	4×10^{-9}	9×10^{-9}
Reference Cycle HLW Risk Total									1×10^{-5}			4×10^{-8}
PT Cycle HLW Transport	1	2×10^{-8}	5×10^{-5}	-	5×10^{-9}	2.5×10^{-13}	1×10^{-16}	3×10^{-13}	5×10^{-14}	5×10^{-21}	1.3×10^{-17}	3×10^{-18}
	2	-	-	-	-	-	-	-	-	-	-	-
	3	3×10^{-10}	6×10^{-7}	30	20	3×10^{-2}	6×10^{-9}	1.2×10^{-5}	3×10^{-6}	9×10^{-12}	1.8×10^{-8}	5×10^{-9}
	4	3×10^{-12}	6×10^{-9}	130	3×10^3	0.4	9×10^{-9}	1.8×10^{-5}	5×10^{-6}	1.2×10^{-12}	2×10^{-9}	6×10^{-10}
PT Cycle HLW Risk Total									8×10^{-6}			6×10^{-9}

Table 3.5.17 presents a preliminary hazards analysis for overpack shipment of non-high level wastes. From this it seems that the risk of concrete-immobilized wastes is negligible and is not included in the analysis. The accident categories that are defined are: Category 1 -- loss and breakage of one drum outside of the overpack, Category 2 -- loss and breakage of half the overpack shipment (21 drums), and Category 3 -- loss, breakage of half the shipment with severe fire accompanying the accident. It should be recognized that the normal dunnage will tend to prevent the release of the drums from the overpack. Table 3.5.18 presents the isotopic inventory for both cycles.

3.5.7.1 Non-High Level Waste Transport: Category 1: One Drum Spillage

This is the fault tree diagrammed in Figure B.4.5. The probability of the inner and outer closure both opening (1L1 and 1L2) is assessed as human failure $10^{-2}/\text{task}^{(15)}$ with a common mode $^{(15)}$ or $10^{-3}/\text{shipment}$ and $2.5 \times 10^{-9}/\text{c-mi}$. The following assessments are based on Clarke, et al.⁽³⁷⁾. The probability of the truck overturning (1L3) is $2 \times 10^{-7}/\text{c-mi}$, Truck-Train accidents (1L5) have a probability of $1.8 \times 10^{-8}/\text{c-mi}$, collisions with a fixed object (1L7) have a probability of $5 \times 10^{-7}/\text{c-mi}$. Thus the probability of loss of containment (1LG2) is $3 \times 10^{-6}/\text{c-mi}$. The probability that a drum breaks open is estimated to be 0.1 for a Category 1 estimate of $3 \times 10^{-7}/\text{c-mi}$.

Should this occur the estimated release fraction (Section 3.5.1) is 7×10^{-8} and the non-immobilized release is 1.9×10^{-3} Ci.

3.5.7.2 Non-High Level Waste Transport: Category 2: 21 Drum Spillage

This is the same accident probability as before -- $3 \times 10^{-7}/\text{c-mi}$ conditioned by the probability of 21 drums spilling and releasing which is estimated in Reference 36 to be 0.003 for a Category 2 probability of $9 \times 10^{-10}/\text{c-mi}$.

Should this occur the estimated release is 0.9 Ci of non-immobilized wastes.

3.5.7.3 Non-High Level Waste Transport: Category 3: 21 Drum Spillage and Fire

This is the previous probability conditioned by the probability of fire⁽³⁷⁾ given an accident 0.1 for a Category 3 probability of $9 \times 10^{-11}/\text{c-mi}$.

Table 3.5.17

PRELIMINARY HAZARDS ANALYSIS FOR NON-HIGH LEVEL WASTE SHIPMENT
BY TRUCK

POTENTIAL HAZARD	CAUSE	BARRIERS	LIKELIHOOD	CONSEQUENCE	EFFECT OF P-T FUEL CYCLE
Fire	Truck Fuel	Diesel Fuel - Low Flammability Most fires are of short duration	1% of Accidents	Immobilized blocks may crack and break drums. Fire may mobilize nuclides on non-immobilized wastes.	50% Reduction for P-T
Overturn, Collision Possibly associated with Fire	Accident	Slow speed, Accident Avoidance Debris covering to prevent dispersion	High	Immobilized blocks may crack and drums break open. Fires may be lost from immobilized wastes	
Missile	Explosives on other transport	Material of low dispersibility	Low	Dispersion of immobilized material	
	Tornado				
Flood or Water Immersion	Flood, Bridge Overturn	Immobilization Drum breakage	Low	Some water dispersion	

Table 3.5.18
NON-IMMOBILIZED WASTE PRINCIPAL CURIE INVENTORY
(42 Drums/Overpack)

<u>ISOTOPE</u>	<u>REFERENCE CYCLE</u>	<u>P-T CYCLE</u>
⁹⁰ Sr	1.1×10^3	1.1×10^3
⁹⁰ Y	1.1×10^4	1.1×10^4
⁹⁵ Nb	1.7×10^3	1.8×10^3
¹⁰⁶ Ru	5.1×10^4	5.4×10^4
¹⁰⁶ Rh	5.1×10^4	5.4×10^4
¹²⁵ Sb	2.0×10^3	2.1×10^3
¹²⁹ I	6.3×10^{-6}	6.5×10^{-6}
¹³⁴ Cs	1.6×10^4	1.8×10^4
¹³⁷ Cs	1.8×10^4	1.9×10^4
^{137m} Ba	1.8×10^4	1.8×10^4
¹⁴⁴ Ce	5.8×10^4	6.0×10^4
¹⁴⁴ Pr	5.8×10^4	6.0×10^4
¹⁴⁷ Pm	1.8×10^4	1.7×10^4
¹⁵⁴ Eu	1.8×10^3	2.1×10^3
¹⁵⁵ Eu	1.2×10^3	1.3×10^3
²³⁸ Pu	7.4×10^4	5.1×10^3
²³⁹ Pu	1.2×10^3	1.0×10^2
²⁴⁰ Pu	3.0×10^3	2.6×10^2
²⁴¹ Pu	7.8×10^5	6.1×10^4
²⁴¹ Am	7.6×10^2	3.4×10^2
²⁴² Cm	-	5.8×10^3
²⁴⁴ Cm	-	1.9×10^4
TOTAL	1.1×10^6	4.1×10^5

Should this occur the non-immobilized estimated waste release is 0.9 Ci with a 1.5×10^{-4} fractional cesium release or 12 Ci total.

3.5.7.4 PT Non-High Level Waste Transport

The only difference between the Reference and PT cycles is the waste inventory. Category 1 has a probability of 3×10^{-7} /c-mi and a release of 7×10^{-4} Ci of non-immobilized waste. Category 2 has a probability of 9×10^{-10} /c-mi and 0.3 Ci and Category 3 has a probability of 9×10^{-11} /c-mi and a release of 0.3 Ci of non-immobilized waste and 13 Ci of cesium.

3.5.7.5 Summary of Reference and PT Non-High Level Waste Transport

A summary for both cycles is presented in Table 3.5.19.

3.5.8 Public Routine Radiological Risks from PT and Reference Transportation

3.5.8.1 Cask Shipments

All of the cask shipments are by train; it is also assumed that the casks meet minimum requirements for Class B Large Quantity shipments, namely the requirement that limits the radiation dose to 10 p-rem/hr at 1 meter. With these provisos, the difference between the Reference and PT cycles is in the number of shipments.

Mathematical models for estimating the public risk from transportation are presented in NUREG-0170⁽⁴¹⁾ and will not be repeated here. Using their formula D-8, the parameter for a spent fuel case ($K \cdot TI = 1000$ m-rem-ft²/hr) and the parameter for rail transport in their Table 4-9. The results are presented in Table 3.5.20.

3.5.8.2 Non-High Level Waste Shipment

The same formula is used with truck parameters from their Table 4-6 to give the results for this shipment mode that are presented in Table 3.5.20. For this case the product $K \cdot TI = 16$ m-rem-ft²/hr.

Table 3.5.19

PROBABILITIES, CONSEQUENCES AND RISK OF NON-HIGH LEVEL WASTE TRANSPORTATION

Transportation Step	Accident Category	Risk pxc)										
		Probability (r)		Curie Released	Consequences		Person-rem/			Latent Cancer/		
		/c-mile	c/trip		Per-rem	Latent Cancer	c-mile	Trip	GWe/yr	c-mile	Trip	GWe/yr
Reference Cycle Non-High Level Waste Transportation	1	3×10^{-7}	6×10^{-4}	1.9×10^{-3}	2×10^{-3}	3×10^{-5}	6×10^{-10}	1.2×10^{-6}	1.9×10^{-5}	9×10^{-12}	1.8×10^{-8}	3×10^{-7}
	2	8×10^{-10}	1.8×10^{-6}	0.9	1.0	1.4×10^{-2}	9×10^{-10}	1.8×10^{-6}	3×10^{-5}	1.3×10^{-11}	3×10^{-8}	4×10^{-7}
	3	9×10^{-11}	1.8×10^{-7}	13	4×10^2	5×10^{-2}	4×10^{-8}	7×10^{-5}	1.3×10^{-3}	5×10^{-12}	9×10^{-9}	1.6×10^{-7}
Referencing Cycle Non-High Level Waste Transport Risk Total									1.3×10^{-3}			9×10^{-7}
PT Cycle Non-High Level Waste Transportation	1	3×10^{-7}	6×10^{-4}	7×10^{-4}	2×10^{-3}	5×10^{-5}	6×10^{-10}	1.2×10^{-6}	1.9×10^{-5}	1.5×10^{-11}	3×10^{-8}	5×10^{-7}
	2	9×10^{-10}	1.8×10^{-6}	0.3	0.9	2×10^{-2}	8×10^{-10}	1.6×10^{-6}	3×10^{-5}	1.8×10^{-11}	4×10^{-8}	6×10^{-7}
	3	9×10^{-11}	1.8×10^{-7}	13	4×10^2	6×10^{-2}	4×10^{-8}	7×10^{-5}	1.3×10^{-3}	5×10^{-12}	1×10^{-8}	1.6×10^{-7}
PT Cycle Non-High Level Waste Transport Risk Total									1.3×10^{-3}			1.3×10^{-6}

Table 3.5.20. Public Routine Radiological Risks from PT and Reference Cycle Transportation

				Risk			
Cycle	Modes	Distance (mi)	Trips/GWe	Per-rem/trip	Per-rem/GWe	Latent Cancer/trip	Latent Cancer/GWe
Reference Cycle Routine Radiological Risk	Spent Fuel	1000	6	1.3×10^{-2}	8×10^{-2}	2.6×10^{-6}	1.6×10^{-5}
	Fresh Fuel	1000	2	1.3×10^{-2}	3×10^{-2}	2.6×10^{-6}	5×10^{-6}
	Cladding Hulls	2000	6	2.6×10^{-2}	0.16	5.2×10^{-6}	3×10^{-5}
	HLW	2000	0.27	2.6×10^{-2}	7×10^{-3}	5.2×10^{-6}	1.4×10^{-6}
	Non-HLW	2000	16	1.7×10^{-2}	0.3	3.4×10^{-6}	5×10^{-5}
Reference Total Public Routine Radiological Risk					0.6		1.0×10^{-4}
PT Cycle Routine Radiological Risk	Spent Fuel	1000	10	1.3×10^{-2}	0.13	2.6×10^{-6}	3×10^{-5}
	Fresh Fuel	1000	3	1.3×10^{-2}	0.1	2.6×10^{-6}	8×10^{-6}
	Cladding Hulls	2000	3	1.3×10^{-2}	0.1	2.6×10^{-6}	8×10^{-6}
	HLW	2000	0.27	2.6×10^{-2}	7×10^{-3}	5.2×10^{-6}	3×10^{-5}
	Non-HLW	2000	16	1.7×10^{-2}	0.3	3.4×10^{-6}	5×10^{-5}
Total					0.7		1.1×10^{-4}

3.5.9 Occupational Radiological Risk of PT and Reference Cycle Transportation

3.5.9.1 Cask Shipments

The same rationale is used here as was used in section 3.5.9. Using equation D-12 ⁽⁴¹⁾ of NUREG-0170 to calculate the train crew dose with the train parameters of Table 4-9, the data of Table 3.5.21 are obtained. This assumes a five-member train crew.

3.5.9.2 Non-High Level Waste Shipments

Equation D-12 ⁽⁴¹⁾ of NUREG-0170 is used to estimate the data presented in Table 3.5.21 using the truck parameter of Table 4-6. The assumed truck crew is 2 persons and the mean distance from driver to cargo is 50 ft.

3.5.10 Non-Radiological Risk of PT and Reference Cycle Transportation

3.5.10.1 Cask Shipments

Reference 48 indicates the following death statistics for 1974: 139 train, 1,692 train services and 77 non-train. These statistics include and are dominated by grade crossing accidents. The number of train miles for that year is 64×10^6 passenger train miles and 469×10^6 freight train miles. Hence correcting for the fraction of passenger trains provides freight train fatality estimates of 3.6×10^{-6} death/train-mile. Table 3.5.22 summarizes the non-radiological effects of cask shipments assuming one cask/train.

3.5.10.2 Non-High Level Waste Shipments

Reference 49 (pg. 56) indicates 0.03 fatalities/truck accident and (pg. 64) 3×10^{-6} accident/truck-mile to provide an estimate of 9×10^{-8} fatalities/truck-mile. These data are used in preparing Table 3.5.22 for non-high level waste shipments.

3.5.11 Summary Risk of Transportation

Table 3.5.23 provides a radiological summary of transportation accidents. Under the assumption that 50% of the latent cancers lead to death ⁽²⁾, Table 3.5.24 is prepared. It is seen that the non-radiological risk is dominant and the PT cycle about 33% higher risk primarily to the larger amount of transportation activities.

Table 3.5.21

Occupational Radiological Risk from PT and Reference Cycle Transportation

Cycle	Mode	Distance (mi)	Trips/GWe	Risk (PXC)			
				Per-rem/Trip	Per-rem/GWe	Latent Cancer/Trip	Latent Cancer/GWe
Reference Cycle Occupational Risk	Spent Fuel	1000	6	4×10^{-4}	2×10^{-3}	8×10^{-8}	5×10^{-7}
	Fresh Fuel	1000	2	4×10^{-4}	8×10^{-4}	8×10^{-8}	1.6×10^{-7}
	Cladding Hulls	2000	6	8×10^{-4}	3×10^{-3}	1.6×10^{-7}	1×10^{-6}
	HLW	2000	0.27	8×10^{-4}	2×10^{-4}	1.6×10^{-7}	4×10^{-8}
	Non-HLW	2000	16	0.02	0.37	4×10^{-6}	6×10^{-5}
Reference Cycle Total Occupational Transportation Risk					9.6		6.2×10^{-5}
P-T	Spent Fuel	1000	10	4×10^{-4}	4×10^{-3}	8×10^{-8}	8×10^{-7}
	Fresh Fuel	1000	3	4×10^{-4}	1.2×10^{-3}	8×10^{-8}	2×10^{-7}
	Cladding Hulls	2000	6	8×10^{-4}	3×10^{-3}	1.6×10^{-7}	1×10^{-6}
	HLW	2000	0.27	8×10^{-4}	2×10^{-4}	1.6×10^{-7}	4×10^{-8}
	Non-HLW	2000	16	0.02	0.37	4×10^{-6}	6×10^{-5}
P-T Cycle Total Occupational Transportation Risk					9.6		6.2×10^{-5}

Table 3.5.22

Non-Radiological Risk from PT and Reference Cycle Transportation

Cycle	Mode	Distance	Trips/GWe	Risk (PXC)	
				Fatalities/Trip	Fatalities/GWe
Reference	Spent Fuel	1000	6	3.6×10^{-3}	2.2×10^{-2}
	Fresh Fuel (MOX)	1000	2	3.6×10^{-3}	7.2×10^{-3}
	Fresh Fuel (U)	1000	4	3.6×10^{-3}	1.4×10^{-2}
	Cladding Hulls	2000	6	7.2×10^{-3}	4.3×10^{-2}
	HLW	2000	0.27	7.2×10^{-3}	1.9×10^{-3}
	Non-HLW	2000	16	1.8×10^{-4}	2.9×10^{-3}
Reference Cycle Total Non-Radiological Risk					9×10^{-2}
P-T	Spent Fuel	1000	10	3.6×10^{-3}	3.6×10^{-2}
	Fresh Fuel (MOX)	1000	3	3.6×10^{-3}	1.1×10^{-2}
	Fresh Fuel (U)	1000	7	3.6×10^{-3}	2.5×10^{-2}
	Cladding Hulls	2000	6	7.2×10^{-3}	4.3×10^{-2}
	HLW	2000	0.27	7.2×10^{-3}	1.9×10^{-3}
	Non-HLW	2000	16	1.8×10^{-4}	2.9×10^{-3}
P-T Cycle Total Non-Radiological Risk					0.12

Table 3.5.23. Summary Radiological Effects of Transportation

Transportation Step	Reference Cycle		PT-Cycle	
	p-rem/GWe	Latent/GWe	p-rem/GWe	Latent/GWe
Spent Fuel	3×10^{-5}	9×10^{-9}	4×10^{-5}	6×10^{-9}
Powder	3×10^{-12}	4×10^{-14}	4×10^{-12}	5×10^{-14}
Fresh Fuel	8×10^{-7}	1.1×10^{-8}	4×10^{-7}	5×10^{-9}
Cladding Hulls	1.6×10^{-4}	1.7×10^{-8}	1.7×10^{-4}	1.8×10^{-8}
HLW	1×10^{-5}	4×10^{-8}	8×10^{-6}	6×10^{-9}
Non-HLW	1.3×10^{-3}	9×10^{-7}	1.3×10^{-3}	1.3×10^{-6}
Accident Subtotal	1.4×10^{-3}	1×10^{-6}	1.4×10^{-3}	1.3×10^{-6}
Public Routine	0.6	1×10^{-4}	0.7	1.1×10^{-4}
Occupational Routine	9.6	6.2×10^{-5}	9.6	6.2×10^{-5}
Radiological Transportation Total Risk	10.2	1.6×10^{-4}	10.3	1.7×10^{-4}

Table 3.5.24. Risk of Fatalities/GWe from Transportation

	Reference Cycle	PT Cycle
Radiological	8×10^{-5}	8.5×10^{-5}
Non-radiological	9.1×10^{-2}	0.12
Total	9.1×10^{-2}	0.12

4. SUMMARIZED RISKS OF THE REFERENCE AND PT FUEL CYCLE

Table 4.1 provides an overall radiological and non-radiological risk comparison of the fuel cycles.

Table 4.1
SUMMARIZED COMPARISON OF THE FUEL CYCLES
(Fatalities/GWe capacity-year)

Risk Type	Reference Cycle	PT Cycle	Ratio $\left(\frac{\text{PT}}{\text{Reference}}\right)$
Radiological Accident - Public	5×10^{-7}	7×10^{-7}	1.4
Radiological Routine - Public	9×10^{-4}	4×10^{-3}	4.4
Radiological Occupational	1.4×10^{-3}	3×10^{-3}	1.8
Radiological Subtotal	2.3×10^{-3}	7.7×10^{-3}	3.3
Non-Radiological	0.36	0.95	2.6
Risk Total	0.36	0.96	2.7

The conclusions are that the PT cycle results in a risk increase to the affected population. Radiological risk is a small contributor to this risk. The following sections provide detailed breakdowns of the above summarized risks.

4.1 PUBLIC RADIOLOGICAL ACCIDENT RISK

Table 4.2 summarizes the radiological accidents that could affect the public. It will be noted that the risk is dominated by the earthquake exceeding design basis impacting the FFP. Should this occur the consequences are 3 times as severe in the case of the PT cycle as they are for the Reference cycle.

Table 4.2
SUMMARY ACCIDENTAL PUBLIC RADIOLOGICAL RISK
(Fatality/GWe-Year)

Operation	Reference Cycle	PT Cycle	Ratio $\left(\frac{PT}{Reference}\right)$
FRP	5×10^{-9}	5×10^{-9}	1
FRP-WTF	--	6×10^{-11}	--
FFP	1×10^{-10}	2×10^{-10}	2
FFP-WTF	--	5×10^{-11}	--
Transportation	5×10^{-7}	7×10^{-7}	1.4
Total	5×10^{-7}	7×10^{-7}	1.4

4.2 PUBLIC RADIOLOGICAL ROUTINE RISK

Table 4.3 summarizes the routine public radiological risk of the two cycles. The dominant risk contributors are the FRP and the WTF's. The net result is a factor of 4.5 increase by implementing PT as compared with the Reference cycle.

Table 4.3
SUMMARY ROUTINE PUBLIC RADIOLOGICAL RISK
(Fatality/GWe-Year)

Operation	Reference Cycle	PT Cycle	Ratio $\left(\frac{PT}{Reference}\right)$
FRP	8×10^{-4}	2×10^{-3}	2.5
FRP-WTF	--	1.6×10^{-3}	--
FFP	1.3×10^{-6}	5×10^{-6}	3.8
FFP-WTF	--	2×10^{-4}	--
Transporation	5×10^{-5}	6×10^{-5}	1.2
Total	8.5×10^{-4}	3.8×10^{-3}	4.5

4.3 OCCUPATIONAL RADIOLOGICAL RISK

Table 4.4 summarizes the occupational radiological risk of the two cycles. The factor of 1.8 increase for PT compared with the Reference cycle is due to the larger work force for the WTFs.

Table 4.4
SUMMARY OCCUPATIONAL RADIOLOGICAL RISK
(Fatality/GWe-Year)

Operation	Reference Cycle	PT Cycle	Ratio $\left(\frac{\text{PT}}{\text{Reference}}\right)$
FRP	6.5×10^{-4}	7×10^{-4}	1
FRP-WTF	--	8×10^{-4}	--
FFP	7×10^{-4}	7×10^{-4}	1
FFP-WTF	--	1.5×10^{-4}	--
Transportation	3.1×10^{-5}	3.1×10^{-5}	1
Total	1.3×10^{-3}	2.3×10^{-3}	1.8

4.4 NON-RADIOLOGICAL RISK

The summary non-radiological risks for the Reference and PT fuel cycles are presented in Table 4.5. The increase in the PT case is attributable to the inclusion of the two WTFs. A slight increase in the transportation risk results from the increased number of trips required to transport PT spent fuel.

As can be seen in Table 4.6, chemical effluents are the dominating factor in the occupational risk. The effluents of concern are NO_x , SO_2 , and CO from the oil fired plant operated on site to supply steam and power to the site facilities.

Table 4.5

SUMMARY NON-RADIOLOGICAL RISK BY OPERATION
(Fatalities/GWe-Year)

Operation	Reference Cycle	PT Cycle	Ratio $\left(\frac{PT}{Reference}\right)$
FRP	0.26	0.26	1
FRP-WTF	--	0.39	--
FFP	1.4×10^{-2}	1.4×10^{-2}	1
FFP-WTF	--	0.17	--
Transportation	9.1×10^{-2}	0.12	1.3
Total	0.36	0.95	2.61

Table 4.6

SUMMARY NON-RADIOLOGICAL RISK BY CAUSE
(Fatality/GWe-Year)

Cause	Reference Cycle	PT Cycle	Ratio $\left(\frac{PT}{Reference}\right)$
Chemical Effluents*	0.25	0.78	3.1
Industrial Hazards	0.11	0.17	1.5
Total	0.36	0.95	2.6

Table 4.7 summarizes the industrial risks for the Reference and PT cycles. Here, the incremental increase in PT operations is attributable to the inclusion of two WTFs, and to the increased number of trips required to handle PT spent fuel. For both cycles, the industrial risk is dominated by transportation.

Table 4.7
SUMMARY NON-RADIOLOGICAL RISK DUE TO INDUSTRIAL AND
TRANSPORATION ACCIDENTS
(Fatalities/GWe capacity-year)

Operation	Reference Cycle	PT Cycle	Ratio $\left(\frac{\text{PT}}{\text{Reference}}\right)$
FRP	2×10^{-2}	2×10^{-2}	1
FRP-WTF	--	2.3×10^{-2}	--
FFP	3.3×10^{-3}	3.3×10^{-3}	1
FFP-WTF	--	5.2×10^{-3}	--
Transportation	9.1×10^{-2}	0.12	1.3
Total	0.11	0.17	1.5

5. CONCLUSIONS AND RECOMMENDATIONS

Partitioning and transmutation of actinides increases the total risk of human fatality in the affected population by a factor of 2.8; the risk of death increases from 0.34/GWe-yr for the Reference cycle to 0.91/GWe-yr for the PT cycle. For a 75 GWe industry, this would be respectively 26 fatalities/yr and 69 fatalities/yr. However, the radiological aspects only constitute 0.6% of this risk. Thus the radiological fatality risk for a 75 GWe industry is 0.15/yr and 0.5/yr.

5.1 CONCLUSIONS

The principal conclusion from this study is that the increase in risk in going from the Reference to PT cycle is insufficient to constitute a substantial reason by itself for not doing so. However, the risk increase is substantial and must be offset by benefits that are not found here.

Some general conclusions are:

1. The gaseous effluents from burning of fossil fuels in oil-fired plant boilers presents the largest public fatality risk. Electric boilers supplied from nonfossil fuel electric power generators could result in a substantial risk reduction.
2. The radiological risk is about equally dominated by the exposure of the workers in performing their duties and public exposure from routine effluents.
3. The radiological risk of each WTF is about the same as the FRP.
4. The dominant radiological accident risk of the WTFs is due to the possibility of a criticality.
5. The dominant radiological accident risk of the FFP is due to the possibility of a wet scrap criticality, or aircraft crash into powder storage.
6. The radiological accident risk of the FRP is dominated by the possibility of an ^{85}Kr storage cylinder fracture. The radiological public risk of the FRP comes from ^{85}Kr routine release, thus there are tradeoffs between accident and routine risks.

7. The increase in transportation risk of the PT cycle over the Reference is primarily due to the larger amount of shipping. Moving the largest amount of material in a single shipment results in a risk reduction.
8. If the waste repository were colocated with the FRP, FFP and WTFs, the transportation risk would be reduced by a factor of 200.

5.2 SENSITIVITY ANALYSIS

These results and conclusions are based on many assumptions which could be modified as a result of changes in plant designs and procedures from those considered here. Furthermore, there may be errors in the data used to arrive at the conclusions of Table 4.1. Appendix G includes a detailed sensitivity analysis; however, here only the sensitivity of the major risk contributors is presented (Table 5.1) in terms of fractional change in risk for a fractional change in risk contributor. For example, a change that modified the Reference Radiological Operations Risk Affecting the Public by 100% would affect the radiological subtotal risk by 40% and the overall Reference cycle risk by 0.19%.

Table 5.1

SUMMARIZED RISKS FOR THE REFERENCE AND PT FUEL CYCLES

Risk Contributor	Reference		PT		Risk Ratio
	<u>Person-Rem</u> <u>GWe-year</u>	<u>Fatalities</u> <u>GWe-year</u>	<u>Person-Rem</u> <u>GWe-year</u>	<u>Fatalities</u> <u>GWe-year</u>	<u>Fatalities</u> <u>GWe-year</u> (PT)
					<u>Fatalities</u> <u>GWe-year</u> (Reference)
Non-Radiological Operations Affecting the Public		0.25		0.78	3.1
Non-Radiological Accidents Affecting the Public		0.09		0.13	1.4
Non-Radiological Accidents Affecting the Workers		5x10 ⁻⁴		1.1x10 ⁻³	2
Non-Radiological Subtotal		0.34		0.91	2.9
Radiological Operations Affecting the Workers	12	1.2x10 ⁻³	21	2x10 ⁻³	1.7
Radiological Operations Affecting the Public	9.1	8x10 ⁻⁴	9.8	4x10 ⁻³	5
Radiological Accidents Affecting the Public	1.5x10 ⁻³	5x10 ⁻⁷	1.5x10 ⁻³	7x10 ⁻⁷	1.4
Radiological Accidents Affecting the Workers	7x10 ⁻⁴	7x10 ⁻⁸	1x10 ⁻³	1x10 ⁻⁷	1.4
Radiological Subtotal	21	2x10 ⁻³	31	6x10 ⁻³	3
Risk Total		0.34		0.91	2.9

5.3 ERROR ANALYSIS

The errors associated with this analysis have been estimated based on the sources of data. The errors are combined in quadrature with each one weighed according to its sensitivity to the final result. This procedure is the statistically correct procedure for propagating variances in a linear system when there is statistical independence of each risk contributor. This results in error factors of 90% and 140%, thus the risk is expected to be between 0.8 to 0.2 and 2.8 to 0.5 fatalities/GWe for the Reference and PT cycles, respectively. The ratio of risks could be uncertain in the range of 2.5 to 3.6 with a central value of 2.9. It should be noted that in calculating the errors in the ratio, the dominating risks are nonradiological air pollution and traffic accidents. This ratio contains correlated errors because the same data were applied to both the Reference and PT assessments. Hence the error in the ratio was calculated with both numerator and denominator error correlated in the same direction.

Table 5.2

ESTIMATED ERROR IN THE RISK CONTRIBUTORS
AND THE ESTIMATED OVERALL ERROR

Risk Contributors or Overall Risk	Estimated Error Factors*
Non-Radiological Operations Affecting the Public	2
Non-Radiological Accidents Affecting the Public	0.2
Non-Radiological Accidents Affecting the Workers	0.2
Radiological Operations Affecting the Workers	0.5
Radiological Operations Affecting the Public	2
Radiological Accidents Affecting the Public	10
Radiological Accidents Affecting the Workers	10
Reference Cycle	2
PT Cycle	2
Reference Cycle Radiological	0.9
PT Cycle Radiological	1.4

*Multiplier or Divisor of the Quoted Value

5.4

RECOMMENDATIONS

The work summarized here considers a slice-in-time encompassing 5th recycle of Reference and PT fuel. In actual fact, the preceding recycles would both contribute risk and provide a learning experience to aid in better facility design. Such effects should be included in an integrated assessment of the cycles.

Some aspects of the facilities and processes for the Reference and PT cycles were adopted to facilitate the safety analysis and probably would not be incorporated in a design for construction purposes. As an example, a PT cycle might integrate reprocessing, fabrication and waste treatment in one facility in which case there is more shared usage, such as eliminating the FFP wet scrap recovery by processing it in the FRP. Similarly, separate WTFs for the FRP and FFP would not be needed in an integrated facility. However, a more realistic design would probably not greatly affect the conclusions of this study.

Some specific technical recommendations are:

1. The FRP accident analysis assumes ruthenium and iodine challenges the HEPA filters being carried on particulates. This may not always be true and a better model of iodine and ruthenium chemistry would improve the accuracy. This change would not alter the report conclusions however.
2. A more detailed analysis of the WTFs should be performed when greater facility and process details are available.
3. Pipeline transfers between facilities of self-boiling solutions and slurries are required by the flowsheets. Cask transfer or dilution before transfer would overcome many of the difficulties associated with pipeline transfer.
4. The actinide product solution in the FFP-WTF concentrator produces 150 kW of heat. This would prove to be unworkable in a production facility. It is recommended that this heat output be reduced by dilution.
5. The radioactivity of the concentrated actinide solutions in the WTFs are copious radiolytic hydrogen producers. This constitutes a significant hazard which could be mitigated by inert atmospheres in the critical areas.
6. The cation exchange chromatography process is questionable considering the complexity of operations and the rate of resin deterioration. Alternative processes should be investigated, possibly using inorganics.

6.0 REFERENCES

1. Fullwood, R. R., and R. Jackson, "Partitioning-Transmutation Program Final Report: VI. Short-Term Risk Analysis of Reprocessing, Refabrication and Transportation (Summary)," ORNL/TM-6986 September 1979
2. US ERDA, "Light Water Reactor Fuel Reprocessing and Recycling," ERDA-77-75, Energy Research and Development Administration (now DOE), July 1977
3. NAS/NRC, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," (BEIR Report) 1972
4. Cooperstein, R., R. C. Erdmann, and R. R. Fullwood, "Preliminary Environmental Analysis of a Generic Fuel Reprocessing Plant," EPA-520/3-75-003, May 1974
5. Erdmann, R. C., et al., "Status Report on the EPRI Fuel Cycle Accident Risk Assessment," EPRI NP-1128, July 1979
6. Liquid Metal Engineering Center, "Failure Data Handbook for Nuclear Power Facilities," LMEC-Memo-69-7, August 1969
7. Nicholson, F. H., and R. R. Fullwood, "An Evaluation of the Frequency of Externally-Caused Loss of Off-Site Power at the Clinch River Breeder Reactor Plant," SAI-042-77-PA, January 1977
8. ORNL Staff, "Siting of Fuel Reprocessing Plants and Waste Management Facilities," ORNL-4451, July 1970, pp 8-53
9. Cohen, S. C., and K. D. Dance, "Scoping Assessment of the Environmental Health Risk Associated with Accidents in the LWR Supporting Fuel Cycle," Environmental Protection Agency, Contract #68-01-2237, November 1975
10. Anyakora, S. N., "Some Data on the Reliability of Instruments in the Chemical Plant Environment," The Chemical Engineer, pp 396-402, November 1971
11. Antocisco, S., G. Tonaglia, and A. Valeri, "A Theoretical Reliability Assessment of a Fire Protection System," Proceedings of the Meeting of Specialists on the Reliability of Mechanical Components and Systems for Nuclear Safety, Risø, Denmark 1969, pp 21-41
12. Miller, M. J., "The Reliability of Fire Protection Systems," DICHE Loss Prevention Symposium, Philadelphia, Pennsylvania, November 11-15, 1973
13. M. J. Miller, Factory Mutual Company, private communication to R. R. Fullwood, January 5, 1977
14. Nichols, G. Starr, Decomposition of the Tributyl Phosphate--Nitrate Complexes, DP-526, Savannah River Laboratory, November 1960

15. Reactor Safety Study, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," USNRC report WASH-1400, October 1975
16. Balfanz, H. P. "Failure Rate Compilation," AEC-TR-7564, December 1973
17. Hodge, C. V., and A. A. Jarrett, "Transportation Accident Risks in the Nuclear Power Industry 1975-2000," EPA-520/3-75-023, April 1975
- 17A. Reliability Analysis Center, "Nonelectronic Parts Reliability Data", NPRD-1, Summer, 1978.
18. Fullwood, R. R., and K. J. Gilbert, "An Assessment of the Impact of Human Factors on the Operations of the CRBR SCRS," SAI-010-76-PA, August 1976
19. Current Events Power Reactors, USAEC-OOE p. 6, October 1974
20. Operating Experience Reactor Safety, USAEC ROE 74-18, p. 2, June 1974
21. Selby, J. M., et al., "Considerations in the Assessment of the Consequences of Effluents from Mixed-Oxide Fuel Fabrication Plants," BNWL-1697, Rev. 1, June 1975
22. Gangwer, T. E., et al., Radiation Effects on Ion Exchange Materials, BNWL-50781, Brookhaven National Laboratory, November 1977
23. Lewis, B., and G. VonElbe, "Combustion, Flames and Explosion of Gases," 2nd Edition, Academic Press, Inc., New York, 1961
24. Arnold, E. D., et al., "The Evaluation of Radioactive Releases from Chemical Plants," 7th AEC Air Cleaning Conference, October 10-12, 1961, USAEC Report ITD-7627, March 1962
25. Moeller, D. W., "Problems in Nuclear Air Cleaning Systems," Nuclear Safety 16 (4), July-August 1975
26. Gaskill, J.R., et al., "Preliminary Results of HEPA-Filter Smoke Plugging Tests Using the LLL Full-Scale Fire Test Facility," 14th ERDA Air Cleaning Conference, Volume 1, August 2-4, 1976
27. ANSI Standard N-287-1976, "Criteria for the Design of Plants for the Manufacture of Mixed Oxide (U-Pu) Fuels," December 30, 1976
28. Schneider, H., et al., A Study of the Feasibility of a Small Scale Reprocessing Plant for the Dresden Nuclear Power Station, IDO-14521
29. Westinghouse Nuclear Fuel Division, "Recycle Fuels Plant--License Applications," Docket 70-1432, July 1973
30. Chelapati, C. V., R. P. Kennedy, and I. B. Well, "Probabilistic Assessment of Aircraft Hazards for Nuclear Power Plants," Nuclear Engineering and Design 19, 333-364, 1972
31. Mishima, J., "Potential Airborne Releases of Uranium as a Result of the Impact of a Jet Aircraft on a Fuel Fabrication Plant," BNW/XN-187, May 1975

32. WCAP-7486-6 (Incomplete reference)
33. Haugen, E. B., Probabilistic Approaches to Design, John Wiley & Sons, New York, 1968
34. A. G. Croff to Distribution, "Origin Calculations for Reference and P-T Cases," Letter, October 27, 1978
35. Baumeister, T., and L. S. Marke, Standard Handbook for Mechanical Engineers, McGraw Hill, New York, New York, 1967, pp 4-103
36. Fullwood, R. R., et al., "Preliminary Risk Analysis of the Transportation of Nuclear Waste," SAI-032-76-PA, December 1976
37. Clarke, R. K., et al., "Severities of Transportation Accidents," SLA-74-0001, July 1976
38. Dennis, A. W., et al., "Severities of Transportation Accidents Involving Large Packages," SAND77-0001, May 1978
39. Bailey, E. S., et al., "Neutron Shielding Problems in Shipping of High Burnup Thermap Reactor Fuel," Nuclear Tech 17, pp 217-224, March 1973
40. AEC, "Analysis of Pressure Vessel Statistics from Fossil-Fueled Power Plant Service and Assessment of Reactor Vessel Reliability in Nuclear Power Plant Service," WASH-1318, U.S. Atomic Energy Commission (May 1974)
41. USNRC, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes," NUREG-0170, December 1977
42. Yoshimara, A. R., "Crash Testing of Spent-Nuclear Fuel Shipping Systems," SAND 78-0451C, April 1978
43. RADC, "Nonelectronic Reliability Notebook," Rome Air Development Center AD/A-005657, January 1975
44. Thom, H. C. S., "Tornado Probabilities," Monthly Weather Review 91, October-December 1963
45. Doan, P. L., "Tornado Considerations for Nuclear Power Plant Structures," Nuclear Safety, Vol 2, No. 4, pp 296, July-August 1970
46. Napadensky, R. Page, and A. Unione, "An Evaluation of the Sensitivity of TNT and the Probability of a TNT Explosion on the GM & O Railroad Line Past the Braidwood Reactor Site," IITRI Project J-9658, August 1974
47. Etherington, H., Nuclear Engineering Handbook, McGraw-Hill 1958.
48. Statistical Abstracts of the United States 1975, U.S. Government Printing Office
49. National Safety Council, "Safety Facts," 1976

APPENDIX A
PLANT AND CASK DESCRIPTIONS

A.1 FUEL REPROCESSING PLANT

A.1.1 Overview

The Fuel Reprocessing Plant (FRP) is located near Oak Ridge at the site of the proposed EXXON reprocessing plant. The Allied General Nuclear Services (AGNS) plant, located at Barnwell, South Carolina, is adapted to this study by increasing the reprocessing capacity to 2,000 MTHM/year. Other modifications from the Barnwell plant are: (1) coprocessing flow sheet, (2) fluorocarbon krypton recovery of the offgas, (3) carbon-14 recovery, (4) thermal codenitration of the product, and (5) modification of the exhaust gas treatment system prior to discharge such that the gas passes through a deep-bed fiberglass filter prior to two stages of HEPA filtration. (For purposes of analyzing releases, the deep-bed filter is assumed to be 90% efficient.) The increase in capacity is achieved through the addition of an additional receiving station and by increasing the shear, handling equipment, and process vessel capacities by 33%. This study assumes that shielding, criticality margins, and other process items are sized and designed such that either Reference or PT fuel may be processed; however, the processing of PT fuel requires the addition of the FRP-WTF discussed in Appendix A.2.

Figure A.1.1 shows a simplified coprocessing flowsheet. The fuel arrives in a cask (A), is stored underwater (B), withdrawn as needed and sheared into small pieces (C). It is dissolved in nitric acid thereby releasing gaseous fission products some of which, including krypton, are retained by the offgas system. The Purex process is used in step E to separate the heavy metal from the fission product waste (P,Q). This is followed by a partial partitioning that removes sufficient uranium to yield a product that is sufficiently fissionable to be used for reactor fuel. The U-Pu product is concentrated, stored (G) and converted into an oxide powder (H) for shipment to the colocated FFP. The uranium stream (L) is further decontaminated by the Purex processing, silica gel treatment (M) and converted to UF_6 for shipment to an enrichment plant.

Figure A.1.1. Fuels Reprocessing Plant - Coprocessing Flowsheet

Tables A.1.1 and A.1.2, respectively, show the daily quantities and activities of various materials entering the plant. It will be noted that, in general, the material does not differ markedly between the Reference and PT cycles thereby justifying the use of the same plant design for both safety analyses. Table A.1.3 compares FRP waste fractions for the Reference and PT cases.

The major reference to the plant details is the Final Safety Analysis Report for the Barnwell Nuclear Fuels Plant^{(1)*}, and the information presented here are conceptual modifications for processing either Reference or PT fuel under the assumptions of this report. The major system changes have already been mentioned; in addition, it is assumed that the plant shielding is increased and procedures are modified so that the maximum occupational dose is 500 mrem/year. It is also assumed that the 3.5/1.5 (PT/Reference) increase in α -activity will not result in solvent degradation that cannot be overcome by standard practice.

A.1.2 Conceptual Modifications

A.1.2.1 Cask Receiving Area

To handle 2,000 MTHM/year of spent fuel, it will be necessary to have 3 cask receiving stations with all three supplying fuel to the storage pool. The spent fuel storage is enlarged to have a storage capability of 7,000 MT, some fraction of which (nominally 50%) is used for solidified waste.

A.1.2.2 Remote Process Cell

This cell has a volume of 1920 m³ with 250 m³/min air exchange. It contains 3 dissolvers of 2,900 liters (1) volume, 3 dissolver screen pots of 270 l volume. Six batches of 1,100 kg of spent fuel are dissolved in 3,050 l of 2.7 M nitric acid per batch.

The shear that prepares the fuel for dissolution is sufficiently oversized to handle a 33% increase in throughput over the BNFP design and construction.

*References appear at the end of each Appendix section.

Table A.1.1

DAILY INPUT OF ACTINIDES TO FRP
(33,000 MWD/T fuel burnup, 5th recycle, 1.5 yr. fuel)

<u>Material</u>	Reference FRP		PT FRP	
	<u>kg/day</u>	<u>Ci/day</u>	<u>kg/day</u>	<u>Ci/day</u>
Spent Fuel	6,670		6,670	
U	6,260	15.1	6,220	12.3
Pu	156	2.7×10^6	178	2.8×10^6
Np	10.8	1,300	8.8	1,500
Am	10.1	1.3×10^4	11.3	1.6×10^4
Cm	1.9	3×10^5	23	1.1×10^6
^{241}Pu		2.6×10^6		2.7×10^6

Table A.1.2

DAILY INPUT OF FISSION PRODUCTS (FP)
AND α -EMITTING MATERIAL TO THE FRP
(33,000 MWD/T fuel burnup, 5th recycle, 1.5 yr. fuel)

<u>Material</u>	Reference FRP	PT FRP
	<u>Ci/day</u>	<u>Ci/day</u>
FP Total	1.15×10^7	1.20×10^7
Ru-Rh	3.4×10^6	3.9×10^6
Zr-Nb	8.9×10^4	9.3×10^4
Other FP	8.0×10^6	8.0×10^6
α -Activity	5.6×10^5	1.4×10^6

Table A.1.3(a)
FRP WASTE FRACTIONS

<u>Waste</u>	<u>Element</u>	<u>% Loss No PT</u>	<u>% Loss With PT</u>
HAW	U	0.5	0.01
	Pu	0.5	0.01
	Np	20.0	0.01
	Am	99.9	0.01
	Cm	99.9	0.01
Hulls	U	0.1	0.01
	Pu	0.1	0.01
	Np	0.1	0.01
	Am	0.1	0.01
	Cm	0.1	0.01
Non-Combustible Solids	U	0.4	0.02
	Pu	0.4	0.02
	Np	0.4	0.02
	Am	0.0	0.02
	Cm	0.0	0.02
Concreted-Wastes	U	1.0	0.06
	Pu	1.0	0.06
	Np	1.0	0.06
	Am	0.0	0.06
	Cm	0.0	0.06
Total Waste Fractions	U	2.0	0.1
	Pu	2.0	0.1
	Np	.40	0.1
	Am	100.0	0.1
	Cm	100.0	0.1

Table A.1.3(b)

Fission Product Losses to FRP Waste

High-level waste	99.96% (by difference)
Non-combustible waste	0.000868%
Concreted waste	0.02%
Structural material waste	1.02%

Also present in the cell is the HAW tank (15,700 l) and HLW concentration that holds the HA waste (fission products) after co-decontamination.

A.1.2.3 High Level Cell (HLC)

The HLC contains 1060 m³ with 120 m³/min air exchange rate. It contains the 11,640 l (6,730 l/batch in 2.7 M HNO₃) Feed Surge Tank 8,140 l Accountability Tank (7,020 l/batch in 2.6 M HNO₃), and the 8,500 l Feed Adjustment Tank. The other vessel volumes in this cell constitute relatively minor accident sources.

A.1.2.4 High Intermediate Level Cell (HILC)

This cell has 1580 m³ with 120 m³/min air exchange rate. It contains the centrifuge, the centrifugal HA contactor, and the electropulse column. The centrifuge of relatively small volume separates undissolved material and sends it to waste.

The HA centrifugal contactor separates the fission products from the fuel process stream by contacting the fuel-bearing acid with 30% tributyl phosphate (TBP) in a paraffin-base solvent resulting in the product leaving in the organic phase. This is stripped in the HS column with the uranium exiting in the organic stream and the U-actinides exiting in the aqueous. Table A.1.4 shows some of the material flows in HILC.

The column lengths are the same as in reference 1 but the diameters are increased by about 20% to handle the increased capacity. Even with this increase in size, the quantities of material in the columns is small compared with most of the tanks. The largest tank in the HILC is the solvent feed tank and has a volume of 24,000 l.

A.1.2.5 Fuel Product Cell (FPC)

This cell has a volume of 790 m³ and an air change rate of 127 m³/min. It contains concentrators, surge tanks and the denitrator. The concentration of the 27 g/l product solution to 400 g/l is performed in two stages. The first stage performs a 5X concentration resulting in an actinide concentration of 27.5 g/l (Reference case) and 26.3 g/l (PT case). To

Table A.1.4

MATERIAL FLOWS IN THE HILC

<u>Stream Identification</u>	<u>Liters/ Day</u>	<u>Reference Cycle (kg/day)</u>		<u>PT Cycle (kg/day)</u>		<u>Solution</u>
		<u>U</u>	<u>Actinides</u>	<u>U</u>	<u>Actinides</u>	
Codecontamination Feed (D - Fig. A.1.1)	31,000	6,400	183	6,400	210	3M HNO ₃
Codecontamination Solvent	54,000					30% TBP
Codecontamination Solvent (Recycle)	18,000	157	2.74	156	3.1	30% TBP 0.1M HNO ₃
Codecontamination (E)	29,000	31	0.78	30	0.9	2.5M HNO ₃
Codecontamination Product Solvent	74,000	6,500	185	6,500	212	30% TBP 0.28M HNO ₃
Product Solvent From Scrub Column (F)	75,000	6,400	157	6,400	180	30% TBP 0.15M HNO ₃
Aqueous to Partitioning Partial	34,000	22		1		1.3M HNO ₃
Depleted Uranium (L) Product	75,400	5,730		5,600		
Uranium Plutonium Product Stream	33,600	629	157	720	180	

achieve geometric safety from criticality, 11 parallel concentrators are used in the first stage and 2 in the second stage. The second stage produces a 3X increase in concentration. Dual slab surge tanks of 2,300 l each are used following the first stage and two smaller slab tanks (800 l each). These latter slab tanks receive and mix actinides returned from the WTF for the PT cycle.

A.1.3 Product Thermal Denitrator

The conceptual denitrator is patterned from the description of Souply and Neal⁽²⁾ and is a continuous process accepting the 2,240 l/day coming from the concentrated product storage tanks. This is metered to three 30 cm diameter fluidized bed reactors operating at 375°C. The reactor beds are composed of particles of U-actinides oxide and are fluidized by 0.37 m³/min of air forced through the beds. The solution is air-atomized into the beds where evaporation of H₂O and HNO₃ takes place. The product nitrates calcine onto the bed particles. As the amount of bed material continues to increase, it overflows to a product accumulator. From the accumulator, the product is ground, screened and stored in one of 75-40 kg shipping containers in which it is transferred to the FFP.

A.1.4 High Level Waste

The high level liquid waste which is the concentrated raffinate from the decontamination contactor is collected and stored in one of three 40,000 liter tanks. Two 1,130 liter HLLW routing tanks are provided. These in turn feed within the Waste Treatment Facility, in the PT case, or directly to waste calcination and vitrification in the reference case.

The actinide-free waste from the WTF is also received in one of these holding tanks. The waste is calcined in a fluidized bed to a granular free-flowing solid at a temperature of 400°C. The bed is internally heated by the combustion of kerosene.

The granular solids overflow the fluidized bed via a downcomer to a surge hopper which in turn blends the waste with glass flit. The blend is then fed to a continuous joule heated melter. The molten glass is poured into high integrity steel containers. The glass is solidified at a controlled cool down rate to assure annealing. The containers are sealed with a welded lid, decontaminated, and transferred to the Fuel Storage Pool for cooled storage.

A.1.5 Fluorocarbon Krypton Recovery System

Refrigerant-12 (dichlorodifluoromethene) demonstrates selective absorption qualities in the presence of off-gas impurities such as NO_x , H_2O , CO_2 , iodine and methyl iodide and, in fact, it has shown to be effective with the last three listed impurities⁽³⁾.

Figure A.1.3 shows a schematic of the FRP krypton removal system. The off-gas is compressed to the 27 atmosphere operating pressure of the system and cooled to -30°C and passed to the bottom of the absorber column and counter-currently contacted with the downward flowing solvent. Essentially, all the krypton, some bulk feed gas, Xe , N_2O , CO_2 and O_2 are dissolved according to their respective solubilities along with soluble sublimate solids. The decontaminated air is discharged to the main FRP air filtration system.

The loaded solvent passes to an expansion chamber at 4.4 atmospheres and a large portion of the least soluble gases desorbs and passes into the overhead condenser. The solvent containing the remaining gases is contacted with upward flowing, 50°F solvent vapor from the reboiler. This releases more of the less soluble gases and further enriches the krypton in the solvent. This goes to the final stripper with a lower liquid-to-vapor flow ratio at 2 atmospheres and the remainder of the absorbed gas (primarily krypton) is removed and collected as a product in compressed gas storage tanks which are stored at the FRP. (The solvent still is used to remove any remaining, less volatile constituents.)

Figure A.1.3. Fluorocarbon Krypton Recovery Schematic

A.2 FUEL REPROCESSING PLANT-WASTE TREATMENT FACILITY (FRP-WTF)

A.2.1 General WTF Description

The PT cycle requires that all wastes from the reprocessing and fuel fabrication plant be treated to remove, as far as possible, the actinides before the wastes are sent to the waste repository or burial ground. Because the actinides reside in several waste forms (Tables A.2.1 and A.2.2), various methods must be applied to their extraction. Furthermore, the actinides are chemically dissimilar and must be specifically treated (Figures A.2.1 and A.2.2). Solids are fluoride or in situ ceric acid leached. Actinide bearing solutions are processed by solvent extraction techniques to separate and purify the actinide fractions. The solvent primarily used is CMP (bidentate*) although in the FRP-WTF, a TBP** cycle is used to separate U, Pu and Np from the cerium recycle. A 2EHOH (2-ethyl, 1-hexanol) solvent extraction process is used to remove degraded solvent from solvent treatment wastes. Cation exchange chromatography separates actinides from lanthanides.

Both waste treatment facilities are structurally arranged as collections of hot cells in contrast to the open "canyon" concept. Personnel access zones include operating galleries and maintenance alcoves on various levels parallel to the hot cells. Access to the cells for contact maintenance on process equipment is via air locks and labyrinths. Control rooms, change rooms, office areas, and cold chemical storage and makeup areas are further separated from processing zones.

A.2.2 Waste Treatment Facility Operations

Most operations are common to both the FRP and FFP-WTFs and many of the remarks presented here equally apply to the FFP-WTF discussed in Section A.4. Common operations including a salt waste treatment with 2EHOH, actinide-lathanide separation by cation exchange chromatography, aqueous raffinate (from the parent plant) treatment with CMP, and ceric acid leaching of HEPA filters and ash from combustible solids incineration. Primary differences stem from respective waste stream quantities and actinide content. In addition, the FRP-WTF has process steps for treating HLLW, cladding hulls and dissolver solids. Unit operations

*dihexyl-N, N-diethyl carbamyl methylene phosphate

**tributyl phosphate

Table A.2.1

FRP WASTES/FRP-WTF FEED STREAMS

Waste	Description	Vol(l/day)	U (kgs/day)	Pu (kgs/day)	N _p (kgs/day)	Am (kgs/day)	Cm (kgs/day)
HLLW	acidic liquid with actinide and fission product nitrates	34,000	31.0	1.0	28.5	23.0	32.1
Dissolver Solids	dry solids; predominately noble metals	9.0	18.4	3.6	0.3	0.5	0.6
Cladding Hulls	zircalloy pieces;	2,000	6.2	0.2	<0.1	<0.1	<0.1
Solid Wastes For Ashing	HEPA Filter Media (fiber glass)	2,000	30.6	0.9	<0.1	0.1	0.2
Solid Wastes To Leach Tank	Incinerator ashes (misc.)	120	30.6	0.9	<0.1	0.1	0.2
Alkaline Salt Waste	Carbonate scrub from FRP solvent recovery. Laundry and other detergent wastes.	43,800	61.0	1.8	<0.1	<0.1	<0.1
Acid Salt Waste	Solvent from scrubber, misc. Plant and cask decontamination solutions; lab wastes	28,000	<3.5	<0.1	<0.1	<0.1	<0.1

Table A.2.2
FFP-WTF - FEED STREAM

Stream	Description	Vol (l/day)	U (kgs/day)	Pu (kgs/day)	Np (kgs/day)	Am (kgs/day)	Cm (kgs/day)
FFP IAW	Acidic aqueous raffinate from FFP wet scrap recovery	2,100	2.1	0.2	0.6	5.6	6.8
Carbon Contaminated Wastes	HEPA filter media from FFP and WTF, feed to ashing furnace	790	12.0	1.1	<0.1	0.2	0.2
Incinerator Ash	Residue from incineration of combustibles at onsite incinerator	330	12.0	1.1	<0.1	0.2	0.2
Alkaline Salt Waste	FFP Carbonate scrub solution from solvent recovery	26,700	8.8	4.0	3.5	0.6	0.8
Acid Salt Waste	FFP and FFP-WTF decontami- nation solutions	11,000	1.2	0.1	<0.1	<0.1	<0.1
			36.1	6.5	4.4	6.7	8.1

Figure A.2.1. FRP - Waste Treatment

Figure A.2.2. FFP - Waste Treatment

operations required include acid-fluoride leaching and a tributyl phosphate solvent extraction cycle (including solvent cleanup and recycle). Similarities and differences between the FRP and FFP-WTFs are summarized in Table A.2.3.

Table A.2.3
WASTE TREATMENT FACILITIES

	<u>FRP</u>	<u>FFP</u>
Feed	(1) HAW (2) Salt (ILLW) (3) HEPAs (4) TRU Ash (5) HE Leach/Hulls (6) Dissolver Solids	(1) IAW (2) Salt (ILLW) (3) HEPAs (4) TRU Ash
Salt waste (ethyl hexanol)	Yes	Yes
Cerium leach (w/ashing)	Yes (Yes)	Yes (Yes)
CMP SX TBP SX	Yes (HAW) Yes	Yes (IAW) No
CMP recovery TBP recovery	Yes, 2 cycles Yes	Yes, 1 cycle No
Acid/Water FP Sep. Pu	Yes HAW Yes, HE leach	Yes HAW ($\times 10^{-3}$) No
Cation Exchange Chromatography	Yes	Yes

A.2.2.1 Solid Alpha Waste Treatment

The actinide bearing solid wastes are subjected to leaching with boiling hexanitrate ceric and nitric acids. Gadolinium is added to the leachates as a soluble neutron poison to suppress any nuclear criticality.

Combustible wastes are incinerated at a separate facility and the ashes are transported to WTF. The incinerator ashes are automatically fed into a stirred tank for batch leaching in regenerated ceric acid (shown in Figure A.2.3). The

tank lid containing the electrodes is remotely replaceable as a unit for maintenance. Upon completion of leaching, the solids are separated from the leachate by centrifugation.

HEPA filter material by virtue of the quantity requiring treatment and the nature of the material (fiberglass pad) are treated separately. The filters are mechanically disassembled and the frames are shipped to the site incinerator facility. The fiberglass is shredded, and after low temperature ashing, is batch loaded in a steam-heated leach column which constitutes one portion of a recirculation loop. The remainder of the loop consists of a electrolytic oxidation tank (similar to the leach tank described above) and the inter-connecting piping (see Figure A.2.4). Circulation is accomplished by convection. Upon completion of leaching, the leachate is withdrawn and the filter media expelled from the leach column by air pressure.

The leached ashes and filter media are remotely packaged and transported to onsite assay and storage facilities for eventual shipment as waste. The ceric ion in the leachate is reduced to minimize corrosion and is routed to the FRP-WTF for TBP/CMP solvent extraction processing (see Section A.2.2.6) to recover U, Pu, and Np with TBP and for cerium recycle. A portion of the cerium recycle stream is routed through ion exchange (Section A.2.2.4) to recover americium and curium.

In FFP-WTF the cerium leachate is routed back through the wet scrap recovery system in the FFP. The result is recycle of the cerium via the FFP-IAW to actinide recovery and to the ion exchange unit.

A comparison of FRP and FFP-WTF equipment is given in Table A.2.4.

A.2.2.2 Salt Waste Treatment

Salt waste, as previously noted, refers to the aqueous alkaline streams generated by solvent treatment, and to various acidic wash solutions, decontamination solutions, and the like (acidic aqueous solutions). Both streams are generated within the respective FRP or FFP as well as by the WTF (solvent treatment and ion exchange processes).

Figure A.2.3. Stirred Tank for Electrolytic Oxidation and Actinide Leaching from Ash Wastes. Note that the gas discharge port and instrument port may be welded to the side of the tank rather than the lid.

Figure A.2.4. Thermosiphon Concept for Leaching HEPA Filter Waste. Circulation rate of the leachant is controlled by the steam temperature.

Solutions originating in the parent facility are transferred to the respective WTF via underground piping. In the WTF, the solutions are acidified (if necessary) with nitric acid while in intimate contact with 2EHOH which also removes solvent degradation products. The resultant aqueous raffinate is countercurrent stripped of acidic organic species with 2EHOH and is then routed to solvent extraction actinide recovery. In the FRP-WTF, this is the TBP-CMP system described below (Section 3.6.3). In the FFP-WTF, the solution is routed back to the wet scrap recovery area of the FFP.

The spent 2EHOH is then scrubbed with nitric acid, stripped with sodium carbonate, reacidified and recycled for reuse. The alkaline carbonate waste is routed to the respective parent facility as intermediate level liquid waste. A portion of the 2EHOH is drawn off and burned at the onsite solvent incineration facility. The system consists of two cycles; one for 2EHOH treatment of salt waste solution and one for 2EHOH recovery. The 2EHOH dissolved in the aqueous raffinates is presumed to be removed either by steam stripping or by selective absorption or extraction.

The salt waste treatment cycle consists of batch contractor/pump tanks, extractors, pulsed scrub column, pulsed strip column, and associated tankage. In-line filters are installed between the strip and acid columns. The 2EHOH is cleaned up by carbonate scrub and dilute acid stripping. Both the salt waste feed to the system and the aqueous waste from the 2EHOH cleanup are concentrated in thermal siphon reboilers. The treated waste is reconcentrated in forced circulation reboilers prior to transfer to the parent facility as ILLW.

A.2.2.3. Solvent Extraction Treatment of Parent Facility Aqueous Raffinate

The aqueous raffinate from the primary extraction column in the parent facility (FRP-HAW, FFP-IAW) is transferred to the respective WTF via buried piping and contacted with CMP (bidentate) to recover the actinides. The aqueous waste is routed back to the respective parent facility for treatment and disposal. The actinide bearing CMP is successively stripped of the lanthanides and transuranics, and then of the uranium. The strip solutions are concentrated and combined as feed for the ion exchange process. The spent CMP is recovered and recycled by a sequence of carbonate and acid washings. The resultant aqueous waste is routed to salt waste treatment (Section A.2.2.2). Periodically a portion of the CMP inventory is transferred to the onsite solvent incinerator.

CMP solvent extraction is accomplished with pulsed cartridge type columns, though final design of the FRP-WTF could use a centrifugal contactor for the initial extraction. Actinide solution concentration is accomplished by thermosiphon evaporators.

A.2.2.4 Cation Exchange Chromatography (CEC)

The combined purified actinide streams from salt waste treatment, solid waste treatment, and aqueous raffinate separations contain most of the original lanthanides. The separation of actinides from lanthanides is accomplished by ion exchange chromatography performed in pressurized cation exchange columns in series. Due to the large decay heat, the separation is performed in a large number of small batches. In the FRP-WTF, 50 CEC units are required and 5 are required in the FFP-WTF. Each unit is an integral package consisting of three columns connected in series by rigid piping. The package is a wall mounted assembly on a rack with inlet and outlet line connections made up in a manifold. Operations are automated by microcomputer.

The lanthanide fraction from ion exchange separation is combined with the high fission product bearing aqueous waste stream from the WTF and is transferred to the FRP for disposal. The actinide product is routed to the WTF product concentrator system. An actinide fraction is recycled as feed to the CEC system.

It is anticipated that a used rack resin inventory requires change out twice a day, 33 racks being changed out on the average, every day. This results in 2719 liters of spent resin being disposed per day. Table A.2.4 summarizes these features. The spent resin is shipped from the WTF to a collocated incinerator.

A.2.2.5 Acid/Water Recycle

In keeping with minimization of effluent releases, process nitrate acid and water are recycled in-process as much as is possible. The excess produced in the FRP-WTF is recycled to the FRP and is eventually discharged as vapor via the FRP stack.

Table A.2.4

CEC RESIN REPLACEMENT SCHEDULE AND RACKS REQUIREMENTS

<u>Plant</u>	<u>CEC Racks</u>	<u>Racks^a Changed/Day</u>	<u>Batches^b Per Rack</u>	<u>Batches Per Day</u>
FRP-WTF ^c	50	66	1	66
FFP-WTF ^d	5	6	1	6

^aApproximately 41.2 liters of resin waste are produced for each rack changed.

^bThis is the number of batches separated in a rack before it is necessary to change the resin.

^cAssumes 33 racks are always in use, ten racks are always being changed, and seven racks are spare.

^dAssumes three racks are always in use, one is being changed, and one is in spare.

Nitric acid recovery is accomplished by:

- Decomposing nitric acid in solution to NO_x with formic acid
- Recovering the NO_x as HNO_3 in scrub columns
- Concentrating the nitric acid by vacuum rectification

This entails a separate vent system collecting the off-gas from concentrators where formic acid denitrification is performed. Thus, NO_x off-gas system is vented through cooled tray atmospheric scrubbers prior to discharge to the main vessel vent.

A.2.2.6 Processes Specific to the FRP-WTF

FRP Hulls Treatment. The hulls from the FRP are received in casks at the WTF and unloaded using the basket inserts. The baskets are placed in a leach tank and contacted with nitric acid containing $0.05M F^-$ ion. Upon completion of leaching ANN is added to the solution to complex the remaining F^- and the solution is transferred to TBP extraction fuel makeup. The hulls, still in the baskets are transferred in casks to onsite packaging and storage facilities.

FRP Dissolver Solids. Solids removed from FRP dissolver solutions by clarification are slurried to the FRP-WTF. The solids are unloaded into a heated and agitated leach tank for digestion in boiling nitric acid containing $0.05 M F^-$. Gadolinium is added to the solution as a soluble neutron poison. After digestion ANN is added to the solution to complex the remaining fluoride. The solution is then centrifuged to remove residual noble metals which, in turn, are transported back to the FRP for incorporation into the HLW glass. The leachate is transferred to the TBP extraction feed makeup.

Cerium Recovery and Recycle. In the FRP-WTF the leachate from the Hulls/Dissolver solids treatment and from alpha waste treatment are combined with the salt waste actinide product stream. After clarification, the solution becomes aqueous feed to sequential TBP and CMP solvent extraction cycles. The CMP feed stream is the aqueous raffinate from the TBP extraction column containing the bulk of the americium and curium and part of the neptunium. The aqueous raffinate from the CMP extraction contains the complexed F^- and is routed to the WTF waste concentrator for acid recovery. The aqueous from CMP stripping contains the lathanides (mostly stable cerium and gadollinium, and americium/curium. The bulk of the solution is recycled to the cerium leach system for alpha waste. A portion of the stream feeds to the ion exchange system for Am/Cm recovery and separation from lathanides. This last step prevents the buildup of Am and Cm to excessive concentrations in the recycled reagent.

The spent TBP and CMP solvents are separately recovered by carbonate and dilute acid washes. The solvents are reused in the respective solvent extraction cycles, with portions periodically drawn off and discarded to the onsite solvent incineration facility. The aqueous wastes are combined and routed to the WTF salt waste treatment system.

WTF Off-Gas Treatment. The WTF off-gas exhaust stream consists of combined vessel vent and cell exhaust flows. The vessel vent system consists of four subsystems. One subsystem services vessels from which ruthenium may be released (ceric acid leach) and has provisions for ruthenium removal by silical gel absorption. A second subsystem services those vessels where 2EH0H vapors are present and is equipped with organic removal capability (knockout demisters or scrubbers). A third subsystem is provided to treat off-gas from concentrators and includes NO_x scrubbers and greater vacuum capability. A final subsystem services those remaining vessels not requiring any special treatment other than adequate vacuum.

The vent off-gases are combined but subsequent treatment differs for the FRP-WTF and the FFP-WTF. These differences are shown by comparing Tables A.2.4 and A.2.5. In the FRP-WTF off-gas treatment, additional ruthenium and iodine removal capability is desirable.

Table A.2.4

OFF-GAS TREATMENT IN THE FRP-WTF AFTER NO_x SCRUBBING

<u>Step</u>	<u>Equipment Requirements</u>
1. Mist de-entrainment	
2. Heating	
3. Prefiltration	1 roughing + 1 HEPA
4. Ruthenium adsorber	Silica gel
5. Iodine adsorber	Silver zeolite
6. Preheating	
7. Final filtration	2 HEPAs
8. Blowing	125% of flowsheet capacity at $\Delta P = 200$ inches H ₂ O
9. Heating	
10. Catalytic NO _x destruction	Reaction with ammonia over a zeolite catalyst bed at 350°C
11. Cooling	Heat transferred to NO _x destruction feed
12. Discharge	

Table A.2.5

OFF-GAS TREATMENT IN THE FFP-WTF AFTER NO_x SCRUBBING

<u>Step</u>	<u>Equipment Requirements</u>
1. Heating	
2. Prefiltration	1 roughing + 1 HEPA
3. Heating	
4. Final filtration	2 HEPAs
5. Blowing	125% of flowsheet capacity at $\Delta P = 200$ inches H ₂ O
6. Heating	
7. Catalytic NO _x destruction	Reaction with ammonia over a zeolite catalyst at 350°C
8. Cooling	Heat transferred to NO _x destruction feed
9. Discharge	

A.3 FUEL FABRICATION PLANT

The Fuel Fabrication Plant (FFP) is colocated about 1 km from the Fuel Reprocessing Plant (FRP) at the Oak Ridge site. For purposes of this study, the FFP used for reference fuel is the same as that used for PT. The distinction results from the addition of the separate WTF for PT processing. This means that both plants employ thick shielding (0.5 - 1 m), remote fabrication and assembly and remote maintenance.

The plant capacity is 660 MTHM/year operating 263 days/year with 20% recycle giving a daily throughput of 3 MTHM.

It should be emphasized that the work presented here is conceptual in nature and many details remain for resolution. To indicate some areas of uncertainty, actinide migration under high temperature ceramic processing may result in nonuniform fuel pellets. The very large amounts of powder that must be moved in a plant of this size will result in HEPA filter loading in a very short time. (Some form of recirculatory pneumatic transfer may be devised to circumvent this problem.) Cooling of the powder in the hoppers and possibly at other times may be a problem. In this concept, water loops affixed to the outside of the hoppers are assumed.

The conceptual plant layout is shown in Figures A.3.1 through A.3.3 in the respective order of second, first and basement floors which contain the 126,000 sq ft area of the process building. Associated with but not shown or discussed is an administrative and security building. The process building is of conventional pillar and block external construction but encloses cells on the three floors designed to utilize gravity material transfer. These cells for the FFP are sized to provide radiation shielding sufficient to achieve less than 500 mrem/year for any plant worker. They are assumed to comply with category I seismic criteria as well as tornado and flooding criteria. The plant contains emergency electric power units and has access to emergency water supplies.

Figure A.3.4 presents the plant main process flow diagram and Table A.3.1 presents the throughput and in-process material inventory.

Figure A.3.1. Fuel Refabrication Plant

Figure A.3.2. Fuel Refabrication Plant

Figure A.3.3. Fuel Refabrication Plant

A-31

Figure A.3.4a. Block Diagram - Powder Receipt Through Pellet Preparation

Figure A.3.4b. Block Flow Diagram - Powder Storage Through Pellet Inspection

Figure A.3.4c. Block Flow Diagram - Pellet Outgas Through Fuel Shipment

Table A.3.1a
FFP THROUGHPUT AND IN-PROCESS INVENTORY

UNIT PROCESS	PERCENT RECYCLE	CYCLE TIME	INVENTORY HM	DAILY THROUGHPUT							
				20% Pu				5% Pu			
				PROCESS UNITS	Pu-Kg	U-Kg	TOTAL HM - Kg	PROCESS UNITS	Pu-Kg	U-Kg	HM - Kg
Unload Master Blended Cani- sters	20	0.5 Hours	240 Kg	6 Units	602	2410	3012	6 Units		2385 UO ₂ Feed	
Store Master Blended Powder in Hoppers	20	5 days	15,060 Kg	32-500 Kg Hop- pers	602	2410	3012	30 2000 Kg Hoppers		2861 (57,220 Kg U 20 days)	
Weigh Powder Aliquots	20	0.5 Hours	360 Kg	6 Units	602	2410	3012	6 Units	151	2861	3012
Blend MOX Powder	20	16 Hours	1,500 Kg	6-500 Kg Blenders	602	2410	3012	6-500 Kg Blenders	151	2861	3012
Store Blended MOX Powder	20	3 days	9,036	18-500 Kg Hoppers	602	2410	3012	6-500 Kg Hoppers	151	2861	3012

Table A.3.1b

UNIT PROCESS	PERCENT RECYCLE	CYCLE TIME	INVENTORY HM	DAILY THROUGHPUT								
				20% Pu				5% Pu				
				PROCESS UNITS	Pu-Kg	U-Kg	TOTAL HM - KG	PROCESS UNITS	Pu-Kg	U-Kg	TOTAL HM - Kg	PELLETS
Prep MOX Powder Mill, Slug Granulate, Size	10	1 Hour	1200 Kg	6 Prep Lines	552	2209	2761	6 Prep Lines	138	2623	2761	
Fill Contain- ers (40 Kg Each)	10	0.5 Hours	240 Kg	6 Units	552	2209	2761	6 Units	138	2623	2761	
Store Prepped Powder (40 Kg Cannister)	10	Storage (20 Days)	55,220 Kg	1380 Canis.	552	2209	2761	1380 Units	138	2623	2761	
Blend Additive	10	1 Hour	240 Kg	6 Mixers	552	2209	2761	6 Units	138	2623	2761	
Press Pellets 10 g/Pellet	10	Cont.	240 Kg	6 Presses	552	2209	2761	6 Presses	138	2623	2761	276,100
Load Boat (5 Kg of 500 Pellets per Boat)	10	15 min.	60 Kg	6 Units	552	2209	2761 (276,100 Pellets)	6 Units	138	2623	2761	276,100
Sinter Pellets	10	24 Hour	2761 Kg	6 Furnaces	552	2209	2761	6 Furnaces	138	2623	2761	276,100
Unload Boat	10	15 min.	60 Kg	6 Stations	552	2209	2761	6 Stations	138	2623	2761	276,100
Grind Pellets	10	Cont.		6 Grinders	552	2209	2761	6 Grinders	138	2623	2761	276,100
Dry Sludge	8	Cont.	25 Kg	1 Dryer	44	177	221	1 Dryer	11	210	221	

Table A.3.1c

FFP THROUGHPUT AND IN-PROCESS INVENTORY

UNIT PROCESS	PERCENT RECYCLE	CYCLE TIME	INVENTORY HM	DAILY THROUGHPUT										
				20% Pu						5% Pu				
				PROCESS UNITS	Pu-Kg	U-Kg	TOTAL HM - Kg	PELLETS	FUEL RODS	Pu-Kg	U-Kg	TOTAL HM - Kg	PELLETS	FUEL RODS
Dry and In-spect Pellets	10	Cont.	180 Kg	6 Stations	552	2209	2761	276,100		138	2623	2761	276,100	
Load Pellet Trays (5 Kg-500 Pellets/Tray)	10	15 min.	60 Kg	6 Stations	552	2209	2761	276,100		138	2623	2761	276,100	
Store Fin-ished Pellets	10	Storage 20 Days	55,220 Kg 5,522,000	11,044 Trays	552	2209	2761	276,100		138	2623	2761	276,100	
Outgas Pellets	3	2 Hours	189 Kg	6 Stations	517	2068	2585	258,500		129	2456	2585	258,500	
Load Fuel Rods	3	0.1 Hours	600 Kg 300 Rods	6 Stations	517	2068	2585	258,500	1293	129	2456	2585	258,500	1293
Decontaminate, Weld, Fill, and Seal Fuel Rods	3	9 Hours	600 Kg 300 Rods	6 Stations	517	2068	2585		1293	129	2456	2585		1293
Leak Test Inspect and Gamma Scan Fuel Rods	3	8 Hours	1200 Kg 600 Rods	6 Stations	517	2068	2585		1293	129	2456	2585		1293
Store Rods	3	Storage 20 Days	51,700 Kg 25,850 Rods		517	2068	2585		1293	129	2456	2585		1293
Fabricate and Pack Fuel Assembly	1	18 Hours	2535 Kg	6 Stations	507	2028	2535			127	2408	2535		
Store Fuel Assembly	1	Storage 20 Days	50,700 Kg 140 Assemblies		507	2028	2535			127	2408	2535		

Process material arrives at the plant in a rail cask (modified NLI 10/24 or ORNL conceptual design for Reference or PT fuel, respectively) containing 3 MTHM of powder in 75 shipping containers each containing 40 kg. The rail cask enters the enclosed receiving area unloading dock where the atmosphere is controlled and HEPA filtered before exhausting. Using the special cask design features for remote or shadow-shielded handling, the head is removed and the containers are transferred to the unloading area where each powder container is emptied using pneumatic transfer. The empty container is returned to the transfer cask and the fuel powder is stored in large hoppers (500 kg capacity) where aliquots are taken to determine the powder composition. Using this information, the MOX is blended with natural UO_2 powder to produce the required LWR fuel reactivity. This adjusted powder is stored and as required, slugged in one of six lines; the slugs are ground and sieved. Oversized particles are returned for regrinding and undersized particles are returned for reslugging. A press lubricant is added to the acceptably sized particles and the powder is pressed into pellets slightly larger than fuel pellets. Following a density inspection, the accepted pellets are loaded into molybdenum boats and fired at $1,700^\circ\text{C}$ in a 6% H_2 , 94% N_2 atmosphere. The reject pellets are sent to clean scrap recovery. These sintered pellets are ground to the correct size, dimensionally checked and inspected. Oversized pellets are reground; undersized pellets go to clean scrap recovery and acceptable pellets are loaded in pellet trays and stored. As needed, a tray of pellets is removed using a computer controlled transfer device and aligned with the rod loader which inserts the pellets into a presealed cladding tube already containing the bottom hardware. The open end is cleaned, the hardware is inserted and the tube is pressurized and sealed by welding. After leak testing, inspection and gamma scanning, the rods are temporarily stored. As required, the rods are removed from storage by the automated rod handler and assembled to form a new fuel assembly. Still using remote handling devices, the assemblies are stored. When required by a customer, a number of assemblies are remotely removed and loaded into the shipping cask. All of these steps, including the final loading, are performed in a controlled, filtered-before-exhaust atmosphere. This completes the main process description.

Scrap containing only fuel materials, i.e., clean scrap, is transferred to the second floor where it is classified, and that suitable for recovery is stored and as needed is moved to Dry Scrap Recovery where it is

roll-crushed and charged to a reduction-oxidation-reduction reactor. Oxidation causes uranium to form U_3O_8 but plutonium remains PuO_2 . The oxide formation results in expansion and material pulverization. Following oxidation, the material is reduced in a 6% H_2 - 94% N_2 atmosphere thereby reducing U_3O_8 to UO_2 . This cycle is repeated to assure product suitability. The recovered powder is air-transferred to the blending hopper to be incorporated into the main process stream.

Scrap that is contaminated (dirty scrap) with nonfuel material must be chemically separated using reprocessing techniques. Examples of dirty scrap are:

1. Contaminated fuel powder, pellets, chips
2. Sweepings
3. Analytical and QC samples
4. Liquid wastes from the Analytical Services Facility (ASF)

These wastes are collected at the point of origin and transferred and classified at the Scrap Receiving and Classification Area. The scrap classified as dirty is transferred by gravity and conveyor to the Wet Scrap Recovery Area (located in the basement). After accountability analysis, the scrap is dissolved in HNO_3 -ceric acids followed by filtration for particulate removal. This is followed by feed adjustment and the solution is contacted with tributyl phosphate (TBP) in a paraffin-base solvent (the Purex process) in a pulsed column. The product is first extracted by the organic and then stripped from the organic into the aqueous phase. The next step is thermal denitration of the product to produce a fuel powder for recycle into the main process at the blender.

Waste containing so little fuel materials as to make recovery by either wet or dry scrap recovery or by the FFP-WTF (in the case of the PT cycle) impractical is packaged for offsite burial in 55 gallon drums. Combustibles are oxidized, liquids are concentrated and mixed with concrete and solids are compacted. Table A.3.2 summarizes the waste characteristics for the Reference and PT plants.

Table A.3.2
Fuel Fabrication Plant Losses

Waste	Element	% loss No P-T	% loss with P-T
Concrete (scrap recovery, raf- finate, ashes)	U	1.5	0.10
	Pu	1.5	0.10
	Np	4.0	0.10
	Am	20.0	0.10
	Cm	20.0	0.10
Non-comb. solids (HEPA, failed equipment, etc.)	U	0.5	0.05
	Pu	0.5	0.05
	Np	0.5	0.05
	Am	0.5	0.05
	Cm	0.5	0.05
Total losses	U	2.0	0.15
	Pu	2.0	0.15
	Np	4.5	0.15
	Am	20.5	0.15
	Cm	20.5	0.15

From this description of the FFP, it is seen that there are no routine liquid effluents but there are gaseous and filtered particulate effluents that are discharged with the ventilating air.

Air enters the plant through roughing filters to remove dust. (Blow-back dampers are used to prevent reverse air flow in case of explosion or tornado). The slightly compressed air flows from areas of lower to higher radiation. Air entering a powder process cell passes through an entrance HEPA filter and exits through an exhaust filter for that cell. Air from all of the cells is collected in the main exhaust system, passes through two independent banks of HEPA filters and is exhausted from a stack 50 m above the ground surface. The air flow rate is $1.4 \times 10^5 \text{ m}^3/\text{min}$ with 96 filters arranged as 24 in a bank.

A.4 FUELS FABRICATION PLANT-WASTE TREATMENT FACILITY (FFP-WTF)

A.4.1 Functional Description

The FFP-WTF receives and processes wastes from the fuels fabrication plant in a fashion similar to the FRP-WTF previously described (Section A.3). The wastes, characterized in Table A.4.1, once the actinides are removed, are returned to the FRP for assay and disposal. The actinide product stream is returned to the FFP and combined with the aqueous product stream from the wet scrap recovery process.

A.4.2 FFP-WTF Unit Operations

Unit operations in the FFP-WTF are similar to those previously described for the FRP-WTF. The process flows are shown in Figure A.4.2. Differences stem from the lower waste flow rates and the absence of waste forms requiring hydrofluoric acid leaching.

A.4.3 Design Philosophy

The siting requirements, regulatory requirements and design bases are similar to those discussed for the FRP-WTF.

Table A.4.1
FPP-WTF - FEED STREAM

<u>Stream</u>	<u>Description</u>	<u>Vol(l/day)</u>	<u>U (kgs/day)</u>	<u>Pu (kgs/day)</u>	<u>Np (kgs/day)</u>	<u>Am (kgs/day)</u>	<u>Cm (kgs/day)</u>
FPP IAW	Acidic aqueous raffinate from FPP wet scrap recovery	2,100	2.1	0.2	0.6	5.6	6.8
Carbon Contaminated Wastes	HEPA filter media from FPP and WTF, feed to ashing furnace	790	12.0	1.1	<0.1	0.2	0.2
Incinerator Ash	Residue from incineration of combustibles at onsite incinerator	330	12.0	1.1	<0.1	0.2	0.2
Alkaline Salt Waste	FFP Carbonate scrub solution from solvent recovery	26,700	8.8	4.0	3.5	0.6	0.8
Acid Salt Waste	FFP and FFP-WTF decontami- nation solutions	11,000	1.2	0.1	<0.1	<0.1	<0.1
			36.1	6.5	4.4	6.7	8.1

A-41

Figure A.4.2. FPP - Waste Treatment

A.5 SPENT FUEL SHIPPING CASKS

A.5.1 Reference Fuel Shipping Cask

The cask for the Reference fuel is the NLI 10/24⁽⁴⁾ rail cask designed for 10 PWR elements or 24 BWR elements. (0.454 MT/element and 0.197 MT/element, respectively, although the Reference fuel is PWR only.) It is described as a dry cask because the fuel is not immersed in water but is in a helium atmosphere at 81 psig and 690°F. The inner, stainless steel fuel-containing cavity has a wall thickness of 0.75 inches about which are two independent water channels for auxiliary heat removal during transfer and for cooldown at the reprocessing plant. These channels are connected to auxiliary heat exchangers during transport but should auxiliary cooling fail, the fuel will be limited to 900°F (150 day fuel) by passive cooling. Outside of the cooling channels is a 6-inch thick lead gamma shield contained in a 2-inch stainless steel jacket. Outside of this is a 9-inch thick neutron water shield contained within a 3/4 inch stainless steel outer jacket and the cask is sealed by inner and outer closure heads of comparable material. Balsa impact limiters are used on both the ends and the side. Table A.5.1 summarizes some of the cask dimensions, weights and temperature.

A.5.2 PT Fuel Shipping Cask

In contrast with the Reference cask, the ORNL conceptual-design PT cask⁽⁵⁾ has no optional active cooling systems. This design is not presently complete and it is assumed that the cask inner and outer closure head designs are similar to those of the Reference cask. Furthermore, it has been assumed that the cask will meet the appropriate regulatory criteria.

The PT spent fuel cask is about 3.6 m long and 2.9 m diameter and weighs 83 MT. The fuel is contained within 6 helium-filled compartments bored in an aluminum cylinder chosen for good heat transfer. (Refer to Figure A.5.1.) This is surrounded by a 5 cm stainless steel shell which is contained in an 10 cm copper-boron carbide cylinder for neutron and gamma shielding. This is surrounded by a 3.8 cm shell of depleted uranium for gamma shielding which is surrounded by a 18.7 cm of lithium hydride neutron shielding which is radially and longitudinally penetrated by copperstrips for heat transfer. Surrounding this is a

Table A.5.1
REFERENCE SPENT FUEL CASK CHARACTERISTICS

External Dimensions (mm)

Cask Length, Overall	5,194
Cask Diameter, Overall	2,235

Internal Dimensions

Cavity Length (mm)	4,559
Cavity Diameter (mm)	1,143
Number of Compartments	10
Compartment Cross Section (mm)	218 x 218
Compartment Length (mm)	4,356

Mass (kg)

Fuel	7,000
Cask Body	73,120
Spacer	65
Basket	7,161
Inner Lid	3,692
Outer Lid	1,176
Total (Empty)	85,214
Total (Loaded)	92,914

<u>Temperature (°F)</u>	<u>Normal Auxiliary Cooling</u>	<u>No Auxiliary Cooling</u>
Fuel (Average)		659
Fuel (Maximum)		900
Outer Surface	227	324
Cavity Surface	268	418

- ① Spent fuel cavity
- ② Anodized Al surface
- ③ Al fuel basket
- ④ SS
- ⑤ Boron carbide (70 v %) + copper (30 v %)
- ⑥ Uranium
- ⑦ Lithium hydride
- ⑧ Cu fins
- ⑨ SS
- ⑩ Boron carbide
- ⑪ SS
- ⑫ Circumferential Al fins
- ⑬ ZnO coating

Figure A.5.1. PT Spent Fuel Cask Cross Section

2.5 cm shell of stainless steel and 22.5 cm of boron carbide for moderated neutron capture. This is contained by a 5 cm outer jacket of stainless steel onto which is affixed aluminum radiating fins for heat transfer.

Since active cooling is not required and since no water is used for shielding purposes, the use of pressure relief valves is not anticipated.

Preliminary heat transfer calculations have been performed for the cask. When the temperature at the surface of the canisters is 700°F , the surface of the stainless steel at the base of the aluminum fins is about 220°F .

A.6 HIGH EFFICIENCY PARTICULATE FILTERS

Plant ventilation is required in all plants for human health and comfort, and for dilution of potentially explosive mixtures. It is standard practice to establish a pressure gradient that results in air flow from areas of lower to progressively higher radiation levels. Because of the particulates that result from the routine processes and from potential accident occurrences, this air is highly filtered prior to discharge. These filters provide the final boundary before air discharge into the environment. This places stringent requirements on their efficiency and integrity. To a very large extent, whether the outcome of accidents occurring at fuel cycle plants may be considered negligible in terms of effects on public, depends on the performance of these filters.

This appendix presents the data used in this report on the fractional aerosol penetration of tested, cascaded banks of high efficiency particulate air (HEPA) filters and on experience relating to the probability of their failing in use.

A.6.1 Filter Efficiency

The efficiency of HEPA filters is usually quoted for the particle size with the highest probability of penetrating the filter. However, the aerosols which challenge the filters are polydisperse and the passage through filters shifts the distribution of particle sizes so that the fraction penetrating several filters cannot be estimated by simply multiplying the fractional penetrability of one filter challenged by the incident aerosol. Gonzales, et al.⁽⁶⁾ also measured the aerosol size distributions that challenge air filtration systems and found a very fine aerosol in chemical processing plants of $0.5 \mu\text{m}$ amad with a geometric standard deviation (σ_g) of 2.2 (0.1-2.4 μm range of size). Dry process fabrication plants exhibit a much larger aerosol of $3.3 \mu\text{m}$ amad with $2.0 \sigma_g$.

Erdmann, et al.⁽⁷⁾ use these data and the experimentally measured fractional penetration factors as a function of number of filters to estimate the fractional penetration of filters employed for wet or dry processes. These are presented in Table A.6.1. The right column presents data from Gonzales, et al. on the penetrability of a polydisperse $^{238}\text{PuO}_2$ aerosol having a size range 0.22-1.6 μm .

Table A.6.1

COMPARISON OF DISTRIBUTION-WEIGHTED-AVERAGE FRACTIONAL
PENETRATION OF HEPA FILTERS WITH POLYDISPERSE AEROSOLS

HEPA No.	Wet Process Fractional Penetration	Dry Process Fractional Penetration	Estimated Error Factor	Reference 4 Mean Fractional Penetration
1	6×10^{-5}	6×10^{-5}	3	3.2×10^{-5}
2	6×10^{-9}	5×10^{-10}	10	1.4×10^{-10}
3	2×10^{-12}	1×10^{-13}	30	1×10^{-12}

A.6.2 HEPA Filter Reliability

HEPA filters are a high-reliability, Category I component used to allow air passage but to suppress the passage of radioactive particulates. In spite of the high quality and stringent test requirements, some failures have occurred. Moeller⁽⁸⁾ has compiled data on air cleaning failures for the 8-year period 1966-1974 that have occurred at nuclear facilities.

It should be noted that this time interval encompasses the implementation of 10 CFR Appendix B on quality control, as well as the normal design rectification that occurs when failure modes are exhibited. Hence, the failures reported in Moeller may not reoccur. New ones could occur but would be less probable or else they would have been exhibited. Erdmann, et al.⁽⁷⁾ estimate the probability of HEPA filter failure in a wet process application as: 1×10^{-4} /HEPA-year and in a dry process application as the same based on the Moeller data.

With 400 parallel filters arranged 100 per bank in the FRP and the FRP-WTF, the probability of failure in either of these two plants is 1×10^{-2} /year. For the FFP and the FFP-WTF with 96 filters, the probability is 2×10^{-3} /year. Assuming one day to isolate a failed filter, the probability of a filter failing when one was already failed for the FRP or FRP-WTF is:

$$1 \times 10^{-2} \times \frac{1}{365} \times 10^{-2} = 2.7 \times 10^{-7} / \text{year}.$$

With the same assumptions, FFP or FFP-WTF:

$$2 \times 10^{-3} \times \frac{1}{365} \times 2 \times 10^{-3} = 1 \times 10^{-8} / \text{year}.$$

This is based on the assumption that the failure of one filter does not affect the failure probability of the second (no common-mode coupling). Plants are designed to assure against this by separating filter banks, the use of bends in the ducting to degrade shockwave propagation, and the use of screens over the filters to suppress flying objects.

There is one accident for which the preceding accidental failure of HEPAs is not appropriate; it is a failure of a final HEPA with a significant loading. For this case, the probability is that of a single filter failing given a significant loading of radioactive material.

$$\text{FRP or FRP-WTF: } 10^{-2} \times \frac{1}{2} = 5 \times 10^{-3} / \text{year}$$

$$\text{FFP or FFP-WTF: } 2 \times 10^{-3} \times \frac{1}{2} = 1 \times 10^{-3} / \text{year}$$

where it is assumed that only during the last 6 months of usage is the filter significantly loaded with radioactive material.

Appendix A References

1. Allied-Gulf Nuclear Services, Inc., "Safety Analysis Report - Barnwell Nuclear Fuel Plant," USNRC Docket 50-332, October 1973.
2. K. R. Souply and D. H. Neal, "Fluid Bed Direct Denitration Process for Plutonium Nitrate to Oxide Conversion," Rockwell International - Rocky Flats Plant, RFP-2600, June 1977.
3. M. J. Stevenson, R. S. Eby and J. H. Pashley, "Fluorocarbon Absorption Process for the Recovery of Krypton from the Off-Gas of Fuel Reprocessing Plants," Oak Ridge Gaseous Diffusion Plant K-GD-1390, January 1976.
4. C.E. Williams, H.G. Clarke Jr, and G.L. Stukenbroeker, "Design and Licensing Considerations of the NLI 10/24 Rail Transportation System for Shipping Spent Nuclear Fuel," National Lead Industries, Inc., May 1973.
5. C. Alexander, Oak Ridge National Laboratories, private communication.
6. M. Gonzales, et al., "Performance of Multiple HEPA Filters Against Plutonium Aerosols," LA-6546, November 1976.
7. R. C. Erdmann, et al., "Status Report on the EPRI Fuel Cycle Accident Risk Assessment," EPRI NP-1128, July 1979.
8. D. W. Moeller, "Problems in Nuclear Air Cleaning Systems," Nuclear Safety 16 (4), July-August 1975.

APPENDIX B
FAULT TREES

Figure B-1.1

RADIOLYTIC HYDROGEN EXPLOSION

Figure B-1.2
SOLVENT FIRE/EXPLOSION

Figure B-1.3

RED OIL EXPLOSION IN CONCENTRATOR

Figure B-1.4
HLW SOLIDIFIER EXPLOSION
B-5

Figure B-1.5
RED OIL EXPLOSION IN FUEL DENITRATOR

Figure B-1.6
CRITICALITY IN FRP CELL

Figure B-1.7
FAULT TREE FOR HYDROGEN EXPLOSION IN A REDUCING FURNACE

Figure B-2.1
HYDROGEN EXPLOSION IN VOG

Figure B-2.2
NITRATED RESIN EXPLOSION

Figure B-2.3
Resin Explosion

Figure B-2.4
MECHANICAL GENERATION OF SOLVENT MIST

Figure B-3.1
 FAULT TREE FOR CRITICALITY ACCIDENT IN WET SCRAP RECOVERY

Figure B-3.2

FAULT TREE FOR POWDER CONTAINER SPILL

Figure B-3.3
FAULT TREE OF THE HOPPER HEAT REMOVAL SYSTEM

Figure B-4.1
NLI 10/24 TOP OF THE FAULT TREE CATEGORY 1
LOSS OF NEUTRON SHIELD

Figure B-4.2

TOP OF THE FAULT TREE CATEGORY 2 - LOSS OF DOUBLE CONTAINMENT

Figure B-4.3

REFERENCE CASK CATEGORY 3 - LOSS OF DOUBLE CONTAINMENT AND GROSS CLAD DAMAGE

Figure B-4.4

REFERENCE CATEGORY 4 - GROSS FUEL DAMAGE WITH ACCOMPANYING FIRE

Figure B-4.5

TRUCK SUPER TIGER RELEASE

APPENDIX C

CALCULATION OF RADIOLOGICAL DOSE AND HEALTH EFFECTS

Individual dose and health effects, and population dose and health effects were calculated using a slightly modified form of CRAC (Consequences of Reactor Accidents Code). This is the code that was used by the Reactor Safety Study (RSS) for the Final Report⁽¹⁾ and is described there. It is further described by Wall, et al.⁽²⁾; hence, further description is unnecessary. For the accident analysis, the primary effort was in modifying the CRAC Code input file (File 21). For the routine radiological effects, CRAC was modified by changes of units, conversion factors and by removal of the nonlinear health effects model that is in the accident form of the code.

C.1 INPUT DATA PREPARATION FOR ACCIDENT ANALYSIS

The 54 isotopes in the original reactor accident form of CRAC are not all appropriate or inclusive of isotopes of significance for the present work, hence it was necessary to add other isotopes.

Figure C.1 shows a block diagram of the steps involved in using the CRAC Code. The preparation of the inhalation and ingestion dose conversion factors for the new isotopes was performed using the CONVOLX Code⁽³⁾.

The respiratory tract model adopted by the CONVOLX Code is the same as the one used in the RSS (shown in Figure C.2). Three major regions of the respiratory portion that are considered are: the nasopharyngeal (NP), the tracheobronchial (TB), and the pulmonary (PUL). Deposition rates corresponding to these regions are set as the initial condition for each run based on the particle size and clearance model. Eleven translocation pathways are identified in the model. However, the model is expandable by adding some extra organs at the location of other organs. Values for D, W, Y representing the clearance class of a particle are presented in Table C.1. The classification of an isotope's clearance class⁽¹⁾ is summarized and shown in Table C.2.

Figure C.1. CRAC Code Executing Process

Figure C.2. Respiratory Tract Model

Table C.1
RESPIRATORY MODEL PARAMETERS

	D	W	Y
N-P(a)	0.15	0.030	0.003
N-P(b)	0.15	0.270	0.297
T-B(c)	0.076	0.040	0.0003
T-B(d)	0.004	0.040	0.0792
PUL(e)	0.200	0.0375	0.0125
PUL(f)	--	0.100	0.100
PUL(g)	--	0.100	0.100
PUL(h)	0.050	0.0125	0.0375

The organ mass values are based on a "phantom" consisting of idealized "mathematical" models for the total body and organs. This model is in close agreement with the International Commission on Radiological Protection (ICRP) publication, "Reference Man." The lung mass value of 540 grams (blood-free lung) is to be used for alpha particles. The total marrow mass is 3,000 grams and skeleton mass is 5,000 grams. For the lower large intestine, the mass of contents (136.8 grams) is used for the calculation.

The radioactive decay chain and radioactive half-lives are obtained from a chart of the nuclides.

The retention model for all the actinides is

$$K e^{-0.693t/L}$$

where the K, L values of each organ is respectively: bone - 0.45, 36,500; liver - 0.45, 14,600; kidney - 0.01, 14,600; testes - 0.0001, 36,500; ovaries - 0.00001, 36,500; other tissue - 0.05, 36,500.

Table C.2

DEPOSITION AND RETENTION MODELS

*Pulmonary clearance classification of inorganic compounds***Class Y**—Avid retention: cleared slowly (years)

Carbides—actinides, lanthanides, Zr, Y, Mn

Sulfides—none

Sulfates—none

Carbonates—none

Phosphates—none

Oxides and hydroxides—lanthanides, actinides Groups 8 (V and VI), 1b, 2b (IV and V), 3b except Sc^{3+} , and 6b.

Halides—lanthanide fluorides

Nitrates—none

Class W—Moderate retention: intermediate clearance rates (weeks)

Carbides—Cations of all Class W hydroxides except those listed as Class Y carbides.

Sulfides—Groups 2a (V + VI), 4a (IV-VI), 5a (IV-VI), 1b, 2b and 6b (V + VI).

Sulfates—Groups 2a (IV-VII), and 5a (IV-VI)

Carbonates—lanthanides, Bi^{3+} and Group 2a (IV-VII)Phosphates— Zn^{2+} , Sn^{3+} , Mg^{2+} , Fe^{3+} , Bi^{3+} and lanthanidesOxides and hydroxides—Groups 2a (II-VII), 3a (III-VI), 4a (III-VI), 5a (IV-VI), 6a (IV-VI), 8, 2b (VI), 4b, 5b, and 7b Sc^{3+}

Halides—lanthanides (except fluorides), Groups 2a, 3a (III-VI), 4a (IV-VI), 5a (IV-VI), 8, 1b, 2b, 3b (IV-V), 4b, 5b, 6b and 7b

Nitrates—all cations whose hydroxides are Class Y and W

Class D—Minimal retention: rapid clearance (days)

Carbides—see hydroxides

Sulfides—all except Class W

Sulfates—all except Class W

Carbonates—all except Class W

Phosphates—all except Class W

Oxides and Hydroxides—Groups 1a, 3a (II), 4a (II), 5a (II, III), 6a (III).

Halides—Groups 1a and 7a

Nitrates—all except Class W

Noble Gases—Group 0

Note: Where reference made from one chemical form to another, it implies that an *in vivo* conversion occurs, e.g. hydrolysis reaction.

The following periodic table of the elements is used with the foregoing classification.

Group																		
Period	1a	2a	3b	4b	5b	6b	7b	8			1b	2b	3a	4a	5a	6a	7a	0
I	H																	He
II	Li	Be											B	C	N	O	F	Ne
III	Na	Mg											Al	Si	P	S	Cl	Ar
IV	K	Ca	Sc	Ti	V	Cr	Mn	Fe	Co	Ni	Cu	Zn	Ga	Ge	As	Se	Br	Kr
V	Rb	Sr	Y	Zr	Nb	Mo	Tc	Ru	Rh	Pd	Ag	Cd	In	Sn	Sb	Te	I	Xe
VI	Cs	Ba	La*	Hf	Ta	W	Re	Os	Ir	Pt	Au	Hg	Tl	Pb	Bi	Po	At	Rn
VII	Fr	Ra	Ac†															

* Lanthanides	Ce	Pr	Nd	Pm	Sm	Eu	Gd	Tb	Dy	Ho	Er	Tm	Yb	Lu
† Actinides	Th	Pa	U	Np	Pu	Am	Cm	Bk	Cf	Es	Fm	Md	No	Lw

The integrated activity in the four portions of the GI tract can be calculated from the total input to the GI system and the mean residence times, which are 1 hour, 4 hours, 13 hours, and 24 hours for stomach, small intestine, upper large intestine, and lower large intestine, respectively. The absorption rate from small intestine to blood is:

$$\lambda_a = \frac{1}{T_{si}} \ln \frac{1}{1-f_i} = \frac{1}{T_{sx}} * f_i \quad (\text{for small } f_i).$$

So the input rate from small intestine to upper large intestine is:

$$\lambda_b = \frac{1}{T_{sx}} (1-f_i)$$

The value of f_i can be found in publication ICRP-2⁽⁵⁾.

The values of biological half-lives and regional fractions applicable to the TGLM are shown in Table C.3. The pathways indicated in the table correspond to those shown in the inhalation model (Figure C.1).

Table C.3
TRANSFER FACTORS FOR THE LUNG MODEL

Pathway	D		W		Y	
	f	λ	f	λ	f	λ
(a)	0.5	8.023×10^{-4}	0.1	8.023×10^{-4}	0.01	8.023×10^{-4}
(b)	0.5	8.023×10^{-4}	0.9	8.914×10^{-6}	0.99	2.006×10^{-5}
(c)	0.95	8.023×10^{-4}	0.5	8.023×10^{-4}	0.01	8.023×10^{-4}
(d)	0.05	4.011×10^{-5}	0.5	4.011×10^{-5}	0.99	4.011×10^{-5}
(e)	0.8	1.605×10^{-5}	0.15	1.605×10^{-7}	0.05	1.605×10^{-8}
(f)	--	--	0.4	8.023×10^{-6}	0.4	8.023×10^{-6}
(g)	--	--	0.4	1.605×10^{-7}	0.4	1.605×10^{-3}
(h)	0.2	1.605×10^{-5}	0.05	1.605×10^{-7}	0.15	1.605×10^{-8}
(i)	1.0	1.605×10^{-5}	1.0	1.605×10^{-7}	0.9	8.023×10^{-9}

Based on this information, the CONVOLX Code is able to calculate the time integrated activity per unit activity inhaled ($\mu\text{Ci-days/Ci}$ inhaled) for radionuclides. The dose conversion factors can be calculated by using the following expressions:

$$\text{F-factor}(\text{REM/Ci inhaled}) = \frac{512}{m} \text{Eeff U (all tissues except GI tract)}$$

$$\text{F-factor}(\text{REM/Ci inhaled}) = \frac{512}{2m_c} \text{Eeff U (GI tract only)}$$

where

m = mass of organs

m_c = mass of the contents in GI tract

U = time integrated activity deposited in the organ

Eeff = effective energy per disintegration for the radionuclide in the organ (MeV/dis)

The value of effective energy for all the isotopes can be found in both ICRP-2⁽⁴⁾ and ORNL-4992⁽⁵⁾. The missing values for some organs are assumed to be zero.

Some assumptions are required before calculating the dose conversion factor:

1. The exposure to skeleton is 80% of the exposure to bone
2. The exposure to total marrow is 20% of the exposure to bone
3. The whole body exposure is equivalent to the exposure on other tissues
4. The exposure to lung is the summation of all the exposure at four sections of PUL LUNG
5. The lack of krypton absorption in the biological processes results in an assumed zero conversion factor for ^{85}Kr

In addition to the respiratory model, it is also necessary to model the effects of radioisotopes that become incorporated in the body. This results in chronic exposure for which six exposure pathways affected by up to 10 isotopes (in CONVOLX) may be considered. The six exposure pathways are:

1. Inhalation of resuspended particles
2. Ingestion of exposed crops
3. Ingestion of milk products
4. Ingestion of milk
5. Ingestion of crops contaminated via root uptake
6. Exposure to contaminated ground

The importance of each isotope to pathways 2 to 5 can be defined based on the dose conversion factor in the critical organ. In general, pathway 2 contains cesium; pathway 3, strontium; pathway 4, iodine; and pathway 5 is the combination of pathways 2 and 3. Its value is 10 years delayed from the other two pathways; i.e., the first time period value in pathway 5 corresponds to the second time period value of pathways 2 and 3.

The dose conversion factors for pathways 1 and 6 are prepared using the isotope name followed by the value of concentration factor relating the ground contamination level to the total intake of the isotope from crops by man and milk by man. These two concentration factors can be calculated by the following equations:

$$T_{eff} = T_w T_r / (T_w + T_r)$$

$$CF = 0.0361 T_{eff} \quad \text{from crops}$$

$$CF = \frac{r_{cm} r_{yc} r_{rg}}{\lambda_1 \lambda_2 \lambda_3} \quad \text{from milk}$$

where

T_{eff} = effective residence time on herbage

T_w = weathering half-life (14 days)

T_r = radioactive half-life

λ_R = radioactive decay constant - $0.693/T_r$

r_{cm} = average daily milk consumption (0.7 l/day)

r_{rg} = uptake coefficient per year by pasture grass

r_{gc} = fraction of a total intake of a radionuclide by a cow that is transferred to the udder per liter of milk in the udder

$$\lambda_1 = \lambda_R + r_{rg} + r_{rd}$$

$$\lambda_2 = \lambda_R + V/AD$$

$$\lambda_3 = \lambda_R + r_{milk}$$

r_{milk} = loss of milk from the cow's udder (2.0/day)
 r_{rd} = loss of availability of elements besides radioactive decay
 V/AD = loss term for the pasture grass compartment due to grass consumption by a cow

The concentration factors calculated from these equations for certain isotopes are shown in Table C.4.

The concentration factors of the isotopes in pathway 2 to 4 can be referred to the data shown in Table VI E-5 of reference 1 and Table VI E-7 for pathway 5.

The ingestion dose conversion factor is obtained from CONVOLX Code calculation, but most of the parameters are set to be the same as the inhalation model with only the intake rate to each organ being modified. For the ingestion model, the total dose is taken from the GI tract, so the intake rate to lung is set at zero. The required time periods are slightly different from the inhalation model. The six time periods are: 0-10, 10-20, 20-30, 30-40, 40-50, >50 years.

In addition to inhalation and ingestion, the public may be exposed to radiation as a result of ground contamination or cloud contamination. Most of these data can be found in ORNL-4992; however, some isotopes must be calculated from the basic formulae. The CRAC Code requires the ground contamination dose conversion factor for three time periods; i.e., 8 hours, 7 days and 1 year. If the radioactive half-life is a large number (>365 days), it is reasonable to assume that the ground contamination dose conversion factor is a linear function. Thus,

$$\begin{aligned}
 DF(T \text{ DAYS}) &= DF(1 \text{ YEAR}) * 7/365 \\
 DF(8 \text{ HOURS}) &= DF(1 \text{ YEAR})/(365*3)
 \end{aligned}$$

However, for short-lived isotopes or isotopes for which data are unavailable, it is necessary to calculate the contamination factors from basic equations. From WASH-1400, the dose rate due to radionuclide i as a function of time is found as

$$\begin{aligned}
 RD^i(k) &= (SF)(DC^i)(SD_0^i)[0.63 \exp(-1.13 t) + 0.37 \exp \\
 &\quad (0.0075t)] \exp(-0.693 t/T_{1/2}^i)
 \end{aligned}$$

Table C.4
CONCENTRATION FACTOR CALCULATED FROM EQUATION

	r_{rg}	r_{rd}	r_{gc}	r_{cm}	f_m
Pu	2.74×10^{-5}	1.096×10^{-4}		1.0 l/day	2.00×10^{-6}
Am	2.74×10^{-5}	1.096×10^{-4}		1.0 l/day	5.00×10^{-6}
Cm	2.74×10^{-5}	1.096×10^{-4}		1.0 l/day	5.00×10^{-6}
U	2.74×10^{-5}	1.096×10^{-4}		1.0 l/day	5.00×10^{-4}
Np	2.74×10^{-5}	1.096×10^{-4}		1.0 l/day	5.00×10^{-6}
I	2.74×10^{-5}	1.096×10^{-4}		1.0 l/day	6.93×10^{-3}
Cs	2.74×10^{-5}	1.096×10^{-4}		1.0 l/day	1.4×10^{-3}
Sr	2.74×10^{-5}	1.096×10^{-4}		1.0 l/day	2.0×10^{-3}

CF Isotope	From Crops $0.0361 * \frac{14Tr}{14tTr}$	From Milk	Tr
I ¹²⁹	0.505	4.123	$0.62 \times 10^{+10}$
Pu ²³⁶	0.499	1.811×10^{-4}	$0.104 \times 10^{+4}$
Pu ²⁴²	0.505	1.190×10^{-3}	$0.138 \times 10^{+9}$
Am ^{242M}	0.505	2.722×10^{-3}	$0.56 \times 10^{+5}$
Am ²⁶³	0.505	2.973×10^{-3}	$0.29 \times 10^{+7}$
Cm ²⁴³	0.505	2.055×10^{-3}	$0.117 \times 10^{+5}$
Cm ²⁴⁵	0.505	2.973×10^{-3}	$0.73 \times 10^{+7}$
Cm ²⁴⁶	0.505	2.969×10^{-3}	$0.24 \times 10^{+7}$
Cf ²⁴⁹	0.505	2.887×10^{-3}	$0.17 \times 10^{+6}$
Cf ²⁵⁰	0.503	1.216×10^{-3}	$0.37 \times 10^{+4}$
Cf ²⁵²	0.497	3.526×10^{-4}	$0.803 \times 10^{+3}$
U ²³⁴	0.505	1.880×10^{-1}	$0.902 \times 10^{+4}$
U ²³⁵	0.505	2.975×10^{-1}	$0.87 \times 10^{+10}$
U ²³⁶	0.505	2.975×10^{-1}	$0.87 \times 10^{+10}$
U ²³⁷	0.164	1.835×10^{-5}	$0.675 \times 10^{+1}$
U ²³⁸	0.505	2.875×10^{-1}	$0.164 \times 10^{+13}$
Np ²³⁷	0.505	2.975×10^{-3}	$0.781 \times 10^{+9}$
Cm ²⁴⁴	0.5043	1.633×10^{-3}	$0.6424 \times 10^{+4}$
Cm ²⁴²	0.4653	5.226×10^{-5}	$0.1625 \times 10^{+3}$
Am ²⁴¹	0.5054	2.885×10^{-3}	$0.167 \times 10^{+6}$
Pu ²⁴¹	0.5039	5.647×10^{-4}	$0.48 \times 10^{+4}$
Pu ²⁴⁰	0.5054	1.187×10^{-3}	$0.24 \times 10^{+7}$
Pu ²³⁹	0.5054	1.189×10^{-3}	$0.89 \times 10^{+7}$
Pu ²³⁸	0.5052	1.028×10^{-3}	$0.33 \times 10^{+5}$
Ru ¹⁰⁶	0.4867	2.981×10^{-5}	$0.365 \times 10^{+3}$
Sr ⁹⁰	0.5047	5.7805	$0.1 \times 10^{+5}$
Cs ¹³⁷	0.5048	5.642	$0.11 \times 10^{+5}$

where

SF = the shielding factor to account for the protection afforded by houses (set to 1)

SD_0^i = initial surface deposition level (in curies/m²)

DC^i = dose conversion factor to relate surface deposition level to dose rate in the organ of interest (from EXREM-III Program)(6)

The calculation of this equation is included in CRAC Code; the input CRAC Code takes is the dose conversion factor DC^i . The equations of calculating DC^i are:

$$DC^i = D_{i\beta s} + D_{irs}$$

$$D_{i\beta s} = 1.07 \sum_{n=1}^{N_{\beta i}} \{ v\alpha (F[c, v, a(x)] + \exp[1-v\alpha(x)]) f_{\beta in} \bar{E}_{\beta in} \}$$

$$D_{irs} = 827B_s \sum_{n=1}^{N_{ri}} [\sigma_{in} E_1(\sigma_{in} x) f_{rin} E_{rin}]$$

where

n = particle or photon index

$N_{\beta i}$ = number of beta particles emitted by ith radionuclide

N_{ri} = number of photons emitted by ith radionuclide

$f_{\beta in}$ = abundance of the nth beta particle emitted by ith radionuclide

f_{rin} = abundance of the nth photon emitted by ith radionuclide

$\bar{E}_{\beta in}$ = average energy (MeV) of the nth beta particle emitted by the ith radionuclide

E_{rin} = energy (MeV) of nth photon emitted by the ith radionuclide

$E_i(x)$ is the exponential integral of the first kind.

From this the dose received by an individual over a period of time t' (yr) from the contaminated ground is

$$D = \int_0^{t'} RD^i(t) dt$$

The cloud contamination is calculated similarly using the relationship

$$DF = 8.87 \times 10^{-7} * (\bar{E}_\beta + \bar{E}_\gamma) \frac{\text{mREM/HR}}{\text{pCi/m}^3}$$

where $\bar{E}_\beta$ and $\bar{E}_\gamma$ are the average β and γ energy, respectively.

Besides file-21, the CRAC Code requires another 15 groups of input data. They are: spatial, site, population, topography, economic, isotope, leakage, dispersion, evacuation, latent, acute, chronic, scale factors, and options. For all fuel cycle steps except transportation, the data were those appropriate to the proposed Clinch River Breeder Reactor Site. For transportation accidents, national averages were used.

C.2 INPUT DATA PREPARATION FOR ROUTINE ANALYSIS

The previously described work was for extending CRAC for doing fuel cycle radiological accident calculations by adding isotopes to the input file. This section describes modifications to CRAC itself. These changes were:

1. Concentration factors were taken from Reference 4, Table 2-8 with units of $\text{Ci-m}^2/\text{Ci-sec}$
2. The inventory units are Ci/sec
3. The units are Rem-day/Ci inhaled. A breathing rate of $5.72 \times 10^5 \text{ m}^3/50 \text{ years}$ was used for the first two time periods and the remaining periods were zeroed
4. Ground contamination was given the units of $\text{Rem-m}^2\text{-sec/Ci}$ and data from Reference 4, Table 3.6, was used for the first year time period with the other periods set to zero
5. All shielding factors were set to one (a conservative assumption)

6. No evacuation is allowed
7. Resuspension inhalation is set to zero
8. Miscellaneous changes were: In time = 1, MRCON - only first column value, large number for DT2, small number for DAYS2 in CHRON subgroup, large number for DL(1,I)...DL(4,I) in acute subgroup and 0 for FATFAC and CENT

References to Appendix C

1. Reactor Safety Study, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, Appendix VII, October 1975.
2. I. B. Well, et al., "Overview of the Reactor Safety Study Consequence Model," NUREG-0340, October 1977.
3. J. D. Teresi and F. G. Boudreau, "Inhalation Dose Conversion Factors for Various Radionuclides Associated with Liquid Metal Fast Breeder Reactors," GEAP-14144, September 1976.
4. G. G. Killough and L. R. McKay, "A Methodology for Calculating Radiation Doses from Radioactivity Released to the Environment," ORNL-4992, March 1976.
5. Recommendations of the International Commission on Radiological Protection, Permissible Dose for Internal Radiation, ICRP publication 2, 1959.

APPENDIX D

OCCUPATIONAL RISK CALCULATIONS

D.1 INTRODUCTION

Personnel exposure to ionizing radiation results from three facets of operation: Routine task performance associated with conduct of production, planned exposure associated with facility and equipment maintenance, and exposures resulting from abnormal occurrences and accidents associated both with the event and with subsequent recovery and restoration activities. The previously described Reference and PT reprocessing plants are assumed to utilize remote and contact maintenance strategies comparable to that proposed for the Barnwell reprocessing plant⁽¹⁾. The mixed oxide fuel fabrication facility is assumed to be remotely maintained in all powder areas with contact maintained wet scrap recovery system and fuel pin handling and assembly lines.

The waste treatment facilities also require a mixture of contact and remote maintenance strategies. Conceptually, the facilities are a modified canyon type structure with an outer maintenance alcove/corridor for failure-prone equipment. The vessels are located in cells provided with provisions for entry (liquid systems) for contact maintenance, and provisions for remote maintenance wherever solids or slurries are handled. Plant availabilities are assumed to be about 75% (time on-line efficiency). Facilities are assumed to be close-coupled such that the respective waste treatment facility is shut-down when the parent facility goes down.

In estimating occupational exposures, it is necessary to assume that past experience is applicable, while at the same time, continued tightening of criteria will reduce exposure.

D.2 CALCULATION OF ROUTINE EXPOSURE

D.2.1 Background Information

Normal or routine occupational exposure limits are set by 10CFR part 20. Superimposed in these limits is the added requirement that every reasonable

effort be made to maintain radiation exposures and releases of radioactive materials into the environment below the part 20 limits "as low as is reasonably achievable". Reports for commercial and government facilities indicate that normal exposures are substantially less than allowed limits⁽²⁾.

For the reference and actinide partitioning/transmutation fuel cycles, the assumption is made that shielding provisions in all facilities are so designed that continuous occupancy in a 40-hour work week will result in no more than 500 millirem exposure on an annual basis. Current operating experience and current estimates of exposure on a worker population basis are presented in Tables D.2.1 and D.2.2, for reprocessing and for fuel fabrication, respectively.

The following should be noted:

1. The operating experience is for low burnup fuel, first cycle plutonium.
2. There is very little experience for mixed oxide multi-cycle plutonium reprocessing and fuel fabrication.
3. The bulk of the DOE experience is with low burnup, metallic fuels (uranium) and metal products.

D.2.2 Estimated Routine Occupational Exposure - FRP

The Reference case and PT case FRPs utilize the same equipment and flowsheets to achieve identical production rates. The design basis shielding is the same in both cases - 500 millirem per year. It is, therefore likely that routine exposure would be the same for both plants.

The two facilities would require the same level of staffing to perform routine operations involving radiation exposure. From Table D.2.1 the required staffing level is not readily apparent. Both plants operate at a standard rate of 6.67 MTHM/day. Comparing NFS and proposed 5 MTHM/day plants the effect of capacity on staffing requirements appears relatively insignificant though some difference is marked by the necessity to staff up on a transient basis for exposure at NFS⁽²⁾. If a 1/3 power effect is assumed, (from reference 3), the required staff would be:

$$(250) \left(\frac{6.67}{5} \right)^{1/3} = 275$$

based on the GESMO estimate. The increment would correspond to additional transportation and materials handling requirements (waste packaging, product handling, and storage, etc.).

Calculation of personnel exposure must also be arbitrary in that the data for NFS were biased by

- A preponderance of maintenance activities
- Higher than design criteria background radiation levels in continuously occupied zones.

The AGNS estimate of 1.6 rem/yr and the GESMO estimate of 2.0 rem/yr are more in line with shielding provisions for personnel exposure ≤ 1 rem/yr. The DOE-II and NRC numbers are probably optimistic in that maintenance is included. Taking the AGNS estimate then, the routine dose should be on the order of 440 person-rem/yr for a plant designed to 1 rem/yr. Taking half of that for stricter design criteria gives 220 person-rem/yr for an estimated routine occupational exposure.

D.2.3 Estimated Routine Occupational Exposure - FFP

An argument similar to the one above may also be advanced for the Reference FFP and PT FFP; i.e., the occupational exposures will be equivalent. Estimating the work force for a remote and highly automated fabrication plant from the data on Table D.2.2 is subject to greater uncertainty than was the case for glove box, manual type operations. Further, the plants are about half the capacity of the study case. Again, using the 1/3 power relationship

$$(250) \left(\frac{660}{350} \right)^{1/3} = 310 \text{ personnel.}$$

This is a work force equivalent to that required for the FRP. Comparing Tables D.2.1 and D.2.2 the results are not unreasonable. It may be argued that a reduction in the number of glove box workers due to automated processes would be compensated for by increased maintenance staffing.

By the same token, the range of 0.8 rem/yr to 2 rem/yr are also comparable to the FRP data in Table D.2.1. From this, it is estimated that the worker dose would be 230 person-rem/yr for the Reference and PT cases. This would be a somewhat lower individual exposure for the FFP workers.

Table D.2.1
COMPARISON OF VARIOUS ESTIMATES FOR OCCUPATIONAL EXPOSURE IN AN FRP

Facility		Staff/Capacity	Exposure (person-rem/year)	Reference
NFS*	(Liquid Products)	209/1 MTD	550	GESMO Testimony - 12/76
AGNS	(Liquid Products)	179/5 MTD	290**	AGNS FSAR
DOE-I***	(Liquid Products)	2251/-----	992	EV-0011/9
DOE-II	(UF ₆ , PuO ₂ , Waste Oxide)	900/(5 MTD)	360	ERDA 77-75
GESMO	(UF ₆ , PuO ₂ , Waste Oxide)	250/5	500	WASH 1327
NRC	(UF ₆ , PuO ₂ , Waste Oxide)	300-540/6.67	1,250	NUREG 0116-0117

*1967 data

**excludes maintenance

***low burnup metal fuels

Table D.2.2

COMPARISON OF VARIOUS ESTIMATES FOR OCCUPATIONAL EXPOSURE IN MOX FUEL FABRICATION

Facility	Staff/Capacity	Exposure	
EXXON	-----	(0.8 rem/year)	Docket 70-1257
Westinghouse	220/200 MT/yr	275 (1.25 rem/year)	License Application (7/73)
GESMO	250/350 MT/yr	500 person-rem/yr	WASH 1327
DOE-I	300/(300 MT/yr)	120 person-rem/yr	DOE 77-75
DOE-II	15,823/-----	814 person-rem/yr	EV-0011/9

D.2.4 Estimated Routine Occupational Exposure - WTFs

The FRP-WTF corresponds in size and complexity to its parent facility. The work force has been set at 300⁽⁴⁾. Based on Section D.2.2, routine exposure of 220 person rem/yr is assumed.

The FFP-WTF work force is 120 people⁽⁴⁾. This, based on Section D.2.2, corresponds to 90 person rem/yr.

D.3 OCCUPATIONAL EXPOSURE ASSOCIATED WITH PLANT MAINTENANCE

D.3.1 Background Information

Some estimates have been made for various tasks involving cell entries to remove equipment⁽⁵⁾. Aggregate doses were estimated to vary between 0.40 and 28 person-rem depending on the specific task and the cell environment. The lower value corresponds to routinely contact maintained cells and the higher value to a remote maintained cell.

D.3.2 Occupational Exposure for Maintenance - FRP

Maintenance activities fall into three categories: Those entailing entry into a process area such as to repair/replace contact maintained equipment, those involving repair/replacement of failure prone equipment in maintenance notches, and activities performed in regulated and contaminated shops. No additional maintenance exposure would be expected as a result of actinide recycle. The additional actinides do not present an insurmountable decontamination problem nor a qualitative change in the types of contamination to be coped with. The maintenance activities would be performed under special work permit (SWP). It is standard practice to set an upper limit dose rate above which direct maintenance is not allowed. This limit would be the same for the Reference and PT cases.

Cell entries are required for either repair in place or removal/replacement of failed equipment that is contact maintained. Typically, entries require extensive cell and equipment flushing, entry crew mobilization, or initial entry to complete cell/equipment decontamination and shielding of permanent "hot spots", maintenance, and finally, cleanup and demobilization. The annual

contribution to the facility occupational dose would be determined by the frequency of entries, duration, number of personnel exposed, and extent of pre-entry decontamination.

Expected dose commitments based on entry frequencies for various contact maintained cells in a fuel reprocessing plant were developed in a report to ORNL⁽⁶⁾. Table D.3.1 taken in part from that report summarizes the data extrapolated to a coprocessing plant with high-level waste calcination and thermal codenitration of uranium and plutonium. Extrapolation of values in reference 6 to those in Table D.3.1 requires that the following assumptions be made

- No work is performed in radiation fields in excess of 250 millirem/hr
- All solids/powder operations are conducted remotely or in hot cells with manipulators
- Nitric acid recovery and recycle involves double distillation of nitric acid
- Solvent recovery and recycle equipment are contact maintained except for any filtration equipment.

Table D.3.1

EXPECTED ANNUAL AGGREGATE DOSE FOR CONTACT MAINTENANCE
(reference 6)

Subsystem	Dose (person-rem) for Contact Maintenance Tasks
Codecontamination	80
HLLW Treatment	50
HLLW Calcination and Packaging	50*
U-Pu Partitioning	33
Solvent Recycle	40
Co-denitration and Powder Handling	33*

*Associated primarily with maintenance of auxiliary equipment such as manipulators and repair/recovery of tools, welding fasteners, etc.

Based on a mean time to cell entry of 32 months⁽⁷⁾, a mean aggregate dose of 41 person-rem per entry and the assumption that a major failure is equally likely in any of the cells, the annual maintenance dose would be the sum or 246 person-rem/yr. This is an upper limit as can be shown by the following argument. If the plant is down for 65 days per year and the maintenance crew (about 25% of the work force estimated for AGNS) is completely utilized, a maximum exposure of 225 person rem/yr could be expected based on an allowable whole body exposure of 3 rem for 1 quarter.

D.3.3 Occupational Exposure for Maintenance - FFP

It is also probable that no difference exists between Reference and PT FFP's maintenance exposure. Maintenance activities, as in the FRP, would be conducted under controlled exposure conditions established by thorough decontamination of the job site with remote techniques prior to attempting work.

For this study, it is assumed that the maintenance exposure is equivalent to normal operation. This results in an estimate of 230 person rem/yr.

D.3.4 Occupational Exposure for Maintenance - WTFs

The maintenance in the WTFs would be equivalent to similar activities in the FRP. It is again assumed that maintenance exposure is approximately equal to routine occupational exposures. The corresponding doses would then be 220 and 90 person rem/yr for the FRP-WTF and FFP-WTF, respectively.

D.4 ABNORMAL OCCURRENCES

Abnormal occurrences are defined as being individual exposures in excess of 10CFR20 limits. At NFS this corresponded to about 0.06% of the aggregate exposure⁽⁷⁾. In the seven years covered, this corresponded to a total dose of 74 rem with 1.2% of the work force exposed. Assuming that this experience directly applied, Table D.4.1 projects the exposures expected to result from abnormal occurrences. Again, these values should be taken as a conservative estimate since the NFS data is erratic (over half of the abnormal exposures occurred in one year)⁽⁸⁾.

Table D.4.1
EXPOSURES FROM ABNORMAL OCCURRENCES

Facility	No. of Individuals/yr	Dose (person rem/yr)
FRP	3	10
FFP	3	10
FRP-WTF	3	10
FFP-WTF	1	3

D.5 OCCUPATIONAL EXPOSURE ASSOCIATED WITH PLANT ACCIDENTS

D.5.1 Summary Results

Estimated exposures to personnel as the result of the previously analyzed plant accidents are given in Table D.5.1 for the Fuels Reprocessing Plant, Table D.5.2 for the FRP-Waste Treatment Facility, Table D.5.3 for the Fuels Fabrication Plant, and Table D.5.4 for the FFP-WTF. The waste treatment facility associated with the fuels fabrication plant is of lesser significance than the FRP-WTF. No estimate of the differential exposure between reference and actinide partitioning FRPs and FFPs is made for reasons previously discussed.

The afore mentioned tables were generated as follows. Inadvertent exposures coincident with the accidents and planned exposures for subsequent repair and recovery are estimated. The sum of the two are then combined with the probability of accident occurrence per plant-year to arrive at expected doses. These expected doses are then summed to arrive at the accident contribution to occupational exposure.

D.5.2 Inadvertent Exposure of Operating Personnel During Accidents

The operating personnel exposure resulting from an accident is comprised of two parts; any exposure during the course of the accident, and planned exposure during the course of recovery and repair efforts.

Table D.5.1
OCCUPATIONAL EXPOSURE FROM FRP ACCIDENTS

Description	Probability (/Plant Year)	CONSEQUENCES		RISK (PXC)
		Accident (Rem)	Repair & Recovery (Rem)	Expected Dose (Rem/Plant-Year)
3.1.1.1 Dropped Fuel Assembly	1.2×10^{-3}	---	≤ 3	4×10^{-3}
3.1.1.2 Hydrogen Fire and Explosion in the HAF Tank	3×10^{-4}	≤ 0.1	≤ 11	3×10^{-3}
3.1.1.3 Solvent Fire in the HA Contactor	2×10^{-4}	---	≤ 11	2×10^{-3}
3.1.1.4 "Red Oil" Explosion in HLW Concentrator	4×10^{-6}	≤ 0.1	≤ 11	4×10^{-5}
3.1.1.5 Explosion in the HLW Calciner	2×10^{-5}	≤ 0.1	≤ 3	6×10^{-5}
3.1.1.6 "Red Oil" Explosion in the Fuel Product Concentrator	4×10^{-6}	≤ 0.1	≤ 3	1.2×10^{-6}
3.1.1.7 Explosion in the Fuel Product Denitrator	4×10^{-7}	≤ 0.1	≤ 3	1.2×10^{-6}
3.1.1.8 Criticality in a Process Cell	2×10^{-5}	≤ 0.05	---	1×10^{-6}
3.1.1.9 Failure of a Krypton Storage Cylinder	1.3×10^{-4}	≤ 0.05	---	7×10^{-6}
3.1.1.10 Hydrogen Explosion in Uranium Reductor	9×10^{-4}	≤ 0.1	≤ 3	3×10^{-3}
3.1.1.11 Hydrogen Explosion in Fuel Product Denitrator Feed Tank	3×10^{-4}	≤ 0.1	≤ 3	9×10^{-4}
				1.3×10^{-2}

Table D.5.2
OCCUPATIONAL EXPOSURE FROM FRP-WTF ACCIDENTS

Accident Section	Description	Probability (Plant-Year)	CONSEQUENCES		RISK (PXC)
			Accident Exposure (Rem)	Repair and Recovery (Rem)	Expected Dose (Rem/Plant-Year)
3.2.1.3	Criticality in Process Cell	3×10^{-5}	≤ 0.05	---	2×10^{-7}
3.2.1.4	Hydrogen Explosion in Actinide Bearing Vessel	a. 3×10^{-4}	≤ 0.1	≤ 3	1×10^{-3}
		b. 3×10^{-4}	≤ 0.1	≤ 3	1×10^{-3}
	Hydrogen Explosion in Vessel Vent System	8×10^{-4}	≤ 0.1	≤ 3	2×10^{-3}
3.2.1.5	Nitrated Organic Explosion in Process Concentrator	2×10^{-3}	≤ 0.1	≤ 3	6×10^{-3}
3.2.1.6	"Red Oil" Explosion in Process Concentrator	2×10^{-4}	≤ 0.1	≤ 3	6×10^{-4}
3.2.1.7	Resin Explosion in CEC Unit	4×10^{-4}	≤ 0.1	---	4×10^{-5}
3.2.1.8	Facility Fire	2×10^{-4}	≤ 0.1	---	2×10^{-5}
	Solvent Mist Explosion	7×10^{-4}	≤ 0.1	≤ 3	2×10^{-3}
	TOTAL				1.2×10^{-2}

Table D.5.3
OCCUPATIONAL EXPOSURE FROM FFP ACCIDENTS

Accident Section	Description	Probability (Plant-Year)	CONSEQUENCES		RISK (PXC)
			Accident Exposure (Rem)	Repair and Recovery (Rem)	Expected Dose (Rem/Plant-Year)
3.3.1.3	Air Crash	1.5×10^{-9}	≤ 0.1	≤ 12	2×10^{-8}
3.3.1.4	H ₂ Explosion in ROR	5×10^{-3}	≤ 0.1	≤ 3	2×10^{-2}
3.3.1.5	H ₂ Explosion in Sintering	5×10^{-3}	≤ 0.1	≤ 3	2×10^{-2}
3.3.1.7	H ₂ Explosion in Wet Scrap	3×10^{-4}	≤ 0.1	≤ 3	1×10^{-3}
3.3.1.8	Criticality in Wet Scrap	6×10^{-5}	≤ 0.05	---	3×10^{-6}
3.3.1.9	Powder Shipping Container Spill	3×10^{-5}	≤ 0.1	≤ 3	1×10^{-4}
3.3.1.10	Exothermic Reaction in Powder Storage	1.5×10^{-6}	≤ 0.1	≤ 3	5×10^{-6}
	TOTAL				4×10^{-2}

Table D.5.4
OCCUPATIONAL EXPOSURE FROM FFT-WTF ACCIDENTS

Accident Section	Description	Probability (Plant-Year)	CONSEQUENCES		RISK (PXC)
			Accident Exposure (Rem)	Repair and Recovery (Rem)	Expected Dose (Rem/Plant-Year)
3.4.1.3	Criticality in Process Cell	3×10^{-5}	≤ 0.05	---	2×10^{-7}
3.4.1.4	Hydrogen Explosion in Actinide Bearing Vessel	3×10^{-4}	≤ 0.1	≤ 3	9×10^{-4}
	Hydrogen Explosion in Vessel Vent System	8×10^{-4}	≤ 0.1	≤ 3	2×10^{-4}
3.4.1.5	Nitrated Organic Explosion in Process Concentrator	2×10^{-3}	≤ 0.1	≤ 3	6×10^{-3}
3.4.1.6	"Red Oil" Explosion in Process Concentrator	2×10^{-4}	≤ 0.1	---	2×10^{-5}
3.4.1.7	Resin Explosion in CEC Unit	4×10^{-5}	≤ 0.1	---	4×10^{-5}
3.2.1.8	Facility Fire	2×10^{-4}	≤ 0.1	---	2×10^{-5}
	Solvent Mist Explosion	7×10^{-4}	≤ 0.1	≤ 3	2×10^{-4}
	TOTAL				7×10^{-3}

Inadvertent exposure of personnel during an accident would occur if a radiation field in excess of shielding capability occurred; if exhaust air cleanup failed, was overcome, or was circumvented; or if a spread of contamination occurred in a normally occupied zone (e.g., pipe gallery).

D.5.2.1 Exposure Due to Shielding Failure

Facility shielding is designed for the maximum facility inventory. Consequently, for an accident to result in excessive dose rates in operating zones, either shielding must be impaired or dose rates far in excess of maximum inventory must occur. The former is unlikely in that no combination of chemical explosives arising within the process could effect the integrity of the reinforced concrete walls.

The only accident involving a significant increase in cell radiation would be a nuclear excursion, either in a vessel, in a cell sump, or in an alcove. While a criticality in the canyon might not result in a detectable increase in operating gallery radiation levels, a criticality in a maintenance alcove probably would. The exposure is estimated to be less than 50 millirem, however, based on the January 25, 1961, criticality incident at Idaho Falls⁽⁹⁾. In the event of criticality, zone monitors, and Criticality Incident Alarms (CIAs) would provide immediate notification in affected areas. Personnel would be required to be trained in zone evacuation and appropriate response to nuclear incidents.

D.5.2.2 Exposure Due to Airborne Activity

A second possible path for exposure of operating personnel would be recirculation of radionuclides discharged in the ventilation exhaust. Under some meteorological conditions, stack exhaust gases may be taken up in the fresh air supply system and recirculated to the personnel occupied zones of the facility. This involves those accidents releasing volatile fission product compounds or fission product noble gases (such as fuel assembly rupture or criticality). It would be expected that this exposure would be low due to atmospheric dilution and dilution by the ventilation air intake and distribution system. Warning would be provided by room air radiation monitors equipped with alarms, hence, permitting prompt response which would include assuring proper personnel protection such as donning fresh-air-supplied respirators and partial shutdown of facility air supply systems.

Definition of the ventilation system layout, air supply allocations and distance between stack discharge and air intake is lacking. It is therefore assumed that a potential 50 millirems total dose is expected for operating personnel present during an accident involving release of volatile or noble gas fission products via the stack.

D.5.2.3 Exposure Due to Spread of Contamination

Major accidents involving substantial release of chemical energy (such as a "red oil" explosion) potentially entail flow reversal of process solutions including driving process solutions into personnel occupied areas via pneumatic instrument lines; utility supply lines such as steam, air, or process water; and via cold-chemical addition lines. The release could be via pipe or tube failure, flange or fitting leak, drains or vents, etc.

The likelihood and magnitude of personnel exposure is reduced by the following factors:

- The zones most likely to incur a spread of contamination are not continuously occupied. Presence of personnel would be on a transient basis for a specific task or short duration (i.e., valve operation).
- Zones identified as potentially contaminated are equipped with continuous air sampling/monitoring (CAMS) devices and zone radiation field detection with alarms. Upon uncontrolled release of contamination redundant warnings would allow prompt evacuation of the zone.

Estimated personnel exposure for spread of contamination is assumed to be <0.1 rem. This value is felt to be conservative in that the operators will be trained to evacuate the area when the zone radiation field monitors and CAMS alarm.

D.5.3.4 Planned Occupational Exposure During Repair and Recovery Activities After an Accident

Post accident repair and recovery activities are assumed to be identical to comparable maintenance, as previously discussed in D.3.2. That is, equipment damaged by explosions would be removed and replaced by contact or remote techniques depending on the specific cell maintenance strategy.

D.6 OCCUPATIONAL DOSE DUE TO FACILITY DECOMMISSIONING

The facilities analyzed in this study are assumed to be operated for 40 years and then shutdown and decommissioned over a 5 year period. Occupational exposure for decommissioning a reprocessing plant by dismantlement has been estimated⁽⁵⁾. For the model plant cited, the expected dose was 512 person-rem. Using this number, doses are estimated in Table D.6.1 for the facilities of interest. It is assumed that the dose is proportional to the total volume of contaminated concrete.

Table D.6.1
EXPECTED DECOMMISSIONING DOSES

Facility	M ³ (Concrete)	Person-rem
FRP	2.8x10 ⁵	1575
FRP-WTF	5.9x10 ⁵	3319
FFP	1x10 ⁵	562
FFP-WTF	1.7x10 ⁵	538
Reference ⁽⁵⁾	9x10 ⁴	512

Reference to Appendix D

1. Docket 50332-40, 41, 42, 43, and 44, "Barnwell Nuclear Fuel Plant, Separations Facility License Application, FSAR," Vol. 1, 2, 3, 4, and 5.
2. GESMO Testimony, "Hearing Transcript," December 14, 1976.
3. SAI, "Analysis of Facility Requirements for the Processing of Nuclear Weapon Material," SAI-77-768-LJ, p. 72, June 24, 1977.
4. Private communication, W. D. Tedder, ORNL to R. R. Jackson.
5. Schneider, K. J. and C. E. Jenkins, "Technology, Safety, and Costs of Decommissioning a Reference Nuclear Fuel Cycle Reprocessing Plant," NUREG-0278, prepared by Battelle Pacific Northwest Laboratory for USNRC, October 1977.
6. Jackson, R. R., "Study of the Cost Impact of Remote Maintenance on Fuel Reprocessing - Task 4," SAI-007-78-RI, November 1978.

7. Bresee, J. C., and J. T. Long, "Design Philosophy for Direct Maintenance Radiochemical Plants," AICHE Progress Symposium Series, Nuclear Engineering, Part XI, Vol. 51, No. 60, 1964.
8. GESMO Testimony, "Hearing Transcript," December 9, 1976 and December 15, 1976.
9. USAFC, "Operational Accidents and Radiation Exposure Experience," WASH-1192, Division of Operational Safety, USAEC, Fall 1975.

APPENDIX E

NONRADIOLOGICAL RISKS

E.1 INTRODUCTION

In addition to radiological effects of radiochemical operations, there are conventional safety considerations common to all chemical plants such as effects of chemical effluents, industrial type accidents and transportation accidents. In this appendix an attempt is made to assess these in terms of health effects to the general populace and to the work force.

E.2 CHEMICAL EFFLUENTS

E.2.1 Basis for Assessment

Barnwell Nuclear Fuels Plant (BNFP) is used to establish the chemical releases from the Reference and PT FRPs. The chemical releases from the WTFs and from the FFP are then extrapolated from the FRP results. Additionally, NO_x removal, comparable to the WTF flowsheet, is assumed.

A list of possible FRP effluents and their maximum possible air concentrations for workers working 8 hours daily in the atmosphere is presented in Table E.2.1. Allowable concentrations in this table are referenced from 27 CFR 202, October 18, 1972. The first 13 contaminants in this table are chemicals used in Purex type plants. The last three contaminants (CO_2 , CO, SO_2) are from burning of oil or coal in the electric power stations or in boilers which supply the electricity and heat to the FRP. Of the contaminants listed, the gaseous (NO_x , CO_2 , CO, and SO_2) are potentially more serious, because of the bulk quantities released. The other chemicals will be filtered out by ventilation exhaust treatment.

Chemicals routinely used in the waste treatment facilities are listed in Table E.2.2 with the associated maximum permissible concentrations (if available). There are, in addition to the ones previously listed for the FRP case, the WTFs will also be discharging HNO_3 , NO_x , etc., as well as requiring additional site power plant capacity.

E.2.2 Reference and PT FRP Chemical Releases

The chemical releases from the two facilities are approximately the same by virtue of the similarity of processes. Both facilities process six metric

Table E. 2.1
ALLOWABLE AIRBORNE CONCENTRATIONS OF CHEMICALS
ASSOCIATED WITH OPERATION OF A TYPICAL FRP

Contaminant	Maximal Concentration in Air Allowed by 27 CFR 202	
	<u>ppm</u>	<u>mg/m³</u>
HNO ₃	2	5
NO _x (as NO ₂)	5	9
TBP		5
NH ₃	50	35
Ammonium sulfamate		15
Iron salts, soluble (Fe content)		1
Iron fumes, oxides		10
Hydroxylamine NH ₂ OH - HNO ₃		causes cancer
Hydrazine (skin) NH ₂ NH ₂	1	1.3
Octane CH ₃ (CH ₂) ₆ CH ₃	500	2,350
H ₃ PO ₄		1
U-salts, oxide		0.05, 0.25
zirconium compounds (Zr content)		5
CO ₂	5,000	9,000
CO	50	55
SO ₂	5	13

Table E.2.2

CHEMICALS ROUTINELY USED IN THE WASTE TREATMENT FACILITIES

Contaminant	Maximum Concentration in Air (27 CFR 202)		Toxicity Data (Ref 14)
	ppm	mg/m ³	
HF	3	11	110 ppm/1 min ^(a)
2-ethyl-hexanol	not listed		3200 mg/kg ^(b)
DIPB (Diisopropylbenzene)	not listed		5000 mg/kg ^(b)
CMP	not listed (Probably like TBP, is known to be nonmutagenic)		
DTPA (Diethylene triamin penta-acetic acid)	not listed		665 mg/kg ^(c)
Al(NO ₃) ₃ •9 H ₂ O	not listed		264 mg/kg ^(b)
Zn(NO ₃) ₂ •6 H ₂ O	not listed		1190 mg/kg ^(b)
ZnCl ₂ fumes		1	
Oxalic acid		1	
NaF (in weight of F)		2.5	
NaOH		2	
H ₂ O ₂ (90%)	1	1.4	

(a) threshold concentration sometimes causing death in humans

(b) 50% lethal dose, oral-rat

(c) 50% lethal dose, interperitoneal injection rat

tons of spent fuel per day using a coprocessing flowsheet. The volumes of cold chemicals are the same for the partial uranium separation. Subsequent mixed nitrate solution flows are 20% greater in the PT case (as was previously discussed in Appendix A). This results in slightly more water vapor and NO_x in the stack effluent.

E.2.2.1 FRP Gaseous Releases

The gaseous releases for the Reference and PT FRPs are extrapolated from those given in the BNFP FSAR (reference 2). Included are the emissions from the associated on site power plant which uses oil-fired boilers. Table E.2.3 lists the major constituents and gives release rates and end-of-stack concentrations. The effluent dispersion at the BNFP site boundary for a 100 meter stack is 6.64×10^{-7} under average meteorological conditions (reference 2 - Table II 1-2). This dispersion suffices to dilute the NO_2 , SO_2 , and CO to well below the maximum permitted concentrations listed in Table E.2.1.

Table E.2.3
EXTRAPOLATED FRP GASEOUS EFFLUENTS

Source	Discharge Rate				Discharge Concentration		
	$\frac{\text{H}_2\text{O}}{\text{lb/hr}}$	$\frac{\text{NO}_2}{\text{g/sec}}$	$\frac{\text{SO}_2}{\text{g/sec}}$	$\frac{\text{CO}}{\text{g/sec}}$	$\frac{\text{NO}_2}{\text{mg/m}^3}$	$\frac{\text{SO}_2}{\text{mg/m}^3}$	$\frac{\text{CO}}{\text{mg/m}^3}$
Main Stack	1.8×10^4	3	--	--	330	--	--
Cold Off Gas (via Main Stack)	0	0.77	--	--	340	--	--
Process Boiler 1	7.9×10^3	5.0	6.4	0.36	79	101	5.6
Process Boiler 2	5.2×10^3	3.3	4.4	0.36	79	101	8.4
Service Concentrator	6.1×10^3	--	--	--	--	--	--
Administration Boiler and 2 Warehouse Furnaces	1.80	0.17	0.29	1.3×10^{-3}	--	--	--
Total Major Gaseous Effluents	3.8×10^4	12	11	0.72			

In addition to these effluents, the conversion of UO_2 to UF_6 is a significant effluent contributor for the FRP. The main gaseous effluents will be 2000 g/day NH_3 . These chemicals will be diluted by dispersion many orders of magnitude below the maximum allowable concentration ($35 \text{ mg NH}_3/\text{m}^3$ or $11 \text{ mg HF}/\text{m}^3$).

E.2.2.2 FRP Liquid Effluents

The liquid effluents from the Reference and PT FRPs will be approximately the same as are the effluents from the BNFP provided in Table E.2.4. This table presents concentrations of effluents discharged in 3.5×10^6 gallons/day in cooling and clean-up water. The 3.5×10^6 gallons of effluents are assumed to be discharged to a settling pond for a retention time of 17 days. None of the concentrations in this table are above the regulation levels for discharged water.

The conversion of UO_2 to UF_6 also is a significant liquid chemical effluent contributor. Table E.2.5 presents data from Reference 2. The NH_3 liquid effluent (in the form of NH_4OH) is quite large (7.8 MT/day). It is diluted to a 1% solution prior to discharge.

E.2.2.3 FRP Solid Nonradioactive Waste

It is estimated that 5,000 - 10,000 ft^3/year of solid nonradioactive waste will be produced and disposed of at the BNFP by the standard land-fill operation⁽²⁾. These wastes are from offices, lunchrooms, cold shop areas, etc. Similar amounts of solid nonradioactive wastes ($\sim 10^4 \text{ ft}^3/\text{year}$) are assumed to be produced by the Reference or PT FRPs.

E.2.3 WTF Chemical Releases

E.2.3.1 WTF Gaseous Effluents

Gaseous chemical releases from the respective WTFs may be approximated for those expected for the FRPs given in Table E.2.3. The additional chemicals used presently in the WTF processes are assumed to be quantitatively removed by the vessel ventilation treatment and canyon exhaust gas treatment. For scaling factors, the ratio of process volumes can be used as a rough approximation of

Table E.2.4
CHEMICAL DISCHARGES INTO SETTLING POND
(Reference 2)

Chemical Discharges	<u>ppm</u>
Total Dissolved Solids	80
CO ₂	40
Calcium	1
Chloride	6
Dissolved Oxygen	5
Hardness (CaCO ₃)	6
Iron	0.5
Magnesium	0.5
Nitrogen-Nitrate	0.7
Phosphorus-Ortho	0.4
pH	6-8
Potassium	15
Sulphate	3
Sulfite	0.05
Chemical Oxygen Demand	0.02
Total Organic Carbon	60 max.

Table E.2.5
LIQUID EFFLUENTS FROM URANIUM CONVERSION

Chemical	Release Rate (gm/day)
NH ₄ OH	7.8x10 ⁶
Ca(OH) ₂	4.6x10 ⁵
CaF ₂	202
PO ₄ ³⁻	4x10 ³
PO ₄ ³⁻ (degraded)	4x10 ²
Total Suspended Solids	1.5x10 ⁴
PO ₄ ³⁻ (in cooling tower drift)	2x10 ²

utilities consumed and, hence, associated power plant emissions. For the FRP-WTF the ratio is about 1.5 (vis-a-vis the FRP). For the FFP-WTF, the ratio is about 0.67. The values in Table E.2.3 are multiplied times these ratios to arrive at the values for the respective WTFs.

E.2.3.2 WTF LIQUID EFFLUENTS

The processing of HLW in the FRP-WTF is approximately as complex as the processing of spent fuel in the FRP. The dissolution of the fuel is missing in the WTF, but the extraction and ion exchange steps with HLW are about twice as complex as in FRP. The volume of cooling water used in the FRP-WTF is approximately the same as the FRP, which essentially defines the dilution. It is estimated that 4.7x10⁶ gallons/day of liquid effluents will be produced by the FRP-WTF. In addition, there will be ~15,000 gallons/day of sanitary wastes. The FFP-WTF is assumed to be 67% of these values.

The concentration of contaminants in such effluents is presented in Table E.2.4. It is assumed that the amount of effluent H₃PO₄ from solvents will be the same as from the FRP. The organic materials (DIPB, 2-EHOH, resins, DTPA) will be burned in the incinerators and will not affect the liquid chemical effluents. Some of these could infiltrate the cooling water, but because of the low toxicity (Table E.2.2) the effects will be insignificant.

E.2.4 FFP Chemical Effluents

With the exception of the wet scrap recovery portion, this is a dry process, and the effluents are negligible. The wet scrap recovery is about 20% of the FRP throughput. For the Reference cycle, the fresh fuel from this scrap is dissolved in HNO_3 (using a small addition of HF) and all actinides are precipitated. The precipitated oxalates are filtered, dried, calcined, and reduced in the stream of hydrogen to UO_2 , PuO_2 , Am_2O_3 , or Cm_2O_3 , which are sent back to the blender. These operations are similar to the processes in the head end of the FRP, except for being about 1/15th the amount. Because of the absence of HLW, it is estimated that effluents are 50% less than a head end processing spent fuel, so the effluents may be estimated by reducing Table E.2.3 by a factor of 1/30. The liquid effluent is estimated as a reduced composition of Table #.2.4 diluted in 1.2×10^5 gallons/day of water. Sewage and cleaning water is estimated as 5000 gallons/day.

E.2.5 Summary of Chemical-Caused Health Effects

The principal health effects from the FRP will result from NO_2 , SO_2 , and CO in the amounts shown in Table E.2.3 which, at the site boundary, will be diluted well below air quality standards. The problem for purposes of this study is to establish a connection between chemical effluents and health effects to provide a common basis for comparison with radiological and nonradiological occupational effects of the two fuel cycles.

Reference 4 provides a relative health effects comparison among several pollutants as shown in Table E.2.6. This table provides a relative ranking of health effects from the pollutants. Reference 5 relates health effects to the emission of 1×10^7 lbs. of SO_2 . These data are combined to give the effects estimated in Table E.2.7.

The person-days of lost time may be interpreted as economic deaths by assuming 6000 person-days lost or equivalent to an economic death then the annual health effects of this should include 18 economic deaths for a total of 24 premature and economic deaths/year.

In summary, the anticipated effects of gaseous nonradiological effects on the public for the Reference and PT cycles is shown in Table E.2.8.

Table E.2.6

RELATIVE HEALTH EFFECTS OF POLLUTANTS PER UNIT OF MASS

Pollutant	Mass % in Pollutant	Health Effect %	Health Effect Factor per Unit Mass
CO	52	2	1
HC	12	32	69.3
NO ₂	11	30	70.9
SO ₂	17	29	36.7
Aerosol (particles)	8	12	39.0

Table E.2.7

ESTIMATED ANNUAL HEALTH EFFECTS FROM FRP POLLUTION
(Assumed 75% Plant Capacity)

Pollutant	Mass Discharged (lbs)	Person-Days of Aggravated Heart-Lung Disease Symptoms	Premature Deaths
CO	3.7×10^4	3.7×10^4	2.0
HC (incomplete combustion*)	3.8×10^5	5.5×10^3	0.3
NO ₂	6.2×10^5	2.4×10^4	1.3
SO ₂	5.7×10^5	1.5×10^4	0.8
TOTAL		8.1×10^4	4.4

* Estimated from Reference 5 using scaling for plant size. It includes burning, transport and storage of heating oil.

Table E.2.8

SUMMARY EFFECTS (PER YEAR) GASEOUS NONRADIOLOGICAL EFFLUENTS

Plant/Cycle	Premature Deaths	Economic Deaths	Total
FRP	4	14	18
FRP-WTF	7	21	28
FFP	0.2	0.6	0.8
FFP-WTF	3	9	12
Reference Total	4.2	14.6	18.6
PT Total	14.2	44.6	58.8

E.3 INDUSTRIAL HAZARDS AND ACCIDENTS

E.3.1 Basis for Assessment

Previous analyses have assumed applicability of 28 years of DOE (previously AEC, then ERDA) experience. Table E.3.1 below is taken from Reference 6 and was in turn developed from the actuarial data base in WASH-1192⁽⁷⁾. DOE accident frequency experience has been historically low compared with conventional chemical plant industrial accident frequencies⁽⁷⁾.

Table E.3.1

CONSTRUCTION/INDUSTRIAL ACCIDENT FREQUENCIES

Accident Category	Job Classification	Frequency (Accident/10 ⁶ Man-Hours) 1943-70 (28 year average)
Lost Time Injuries:	Heavy Construction	10
	All Construction	5.36
	AEC Operations	2.12
Fatalities:	Construction	0.042
	AEC Operations	0.023
	Government Functions	0.004

One such comparison has been made for Hanford Operations vis-a-vis conventional experience. This comparison is given in the table below.

Table E.3.2
INDUSTRIAL ACCIDENT STATISTICS*

	Accident Frequency (# of accidents per 10^6 man hours worked)	Accident Severity (Lost time per 10^6 man hours worked)
Hanford Chemical Processing and Waste Management	1.04	97 days
Chemical Industry	4.17	389 days
All Industry	11.20	670 days

*Prepared testimony, Alex G. Fremling, presented at June 18, 1977
hearings before a subcommittee of the Committee on Government Operations,
House of Representatives, held in Richland, Washington

Part of the difference may be attributable to the remote operations required in radiochemical plants which serve to separate the personnel from the operations. In part, it may also be attributable to the physical protection afforded by the massive shielding generally required to protect the personnel from radiation exposure.

For purposes of estimating conventional industrial-type accident frequencies, the assumption is made that future radiochemical plant experience will be no worse than DOE experience.

E.3.2 Construction of Fuel Cycle Facilities

For purposes of estimating accident frequencies for construction of the fuel cycle facilities, the values previously given in Table E.3.1 for all construction are used. To arrive at an estimate of man-hour requirements for construction, two approaches were used. First, a crude estimate of construction

time and level of staffing was made based on facility volume. Second, a "value added" figure based on fractional facility cost for one man-day of effort was obtained from a construction cost estimate⁽⁸⁾.

Table E.3.3 lists the facilities and expected volumes. It is assumed that construction requires five years for any of the listed facilities. This may be conservative for the FFP and the FFP-WTF by virtue of smaller sizes compared to the FRP and FRP-WTF. The second column in the table is the sum of unit outputs of various crafts taken from Reference 9 for concrete structures. Also taken from Reference 9 are percentages of total effort for conventional research facilities and laboratories that the physical structure represents (as compared to mechanical, electrical and other crafts). The final column is the total manpower estimate for a five-year construction project to produce the given facility.

Table E.3.3
ESTIMATED MANPOWER REQUIREMENTS TO
CONSTRUCT THE FUEL CYCLE FACILITIES

Facility	Volume (m ³)	Unit Product (m ³ /Manday)*	% of Total Job**	Total Manpower (Manhours)
FRP	2.8×10^5	0.2	10	4.4×10^6
FFP	1×10^5	0.2	10	1.6×10^6
FRP-WTF	5.9×10^5	0.2	10	9×10^6
FFP-WTF	1.7×10^5	0.2	10	2.7×10^6

*Page 60, Reference 9, general concrete work for a crew of 31 men. Concrete represents about 20% of total facility volume in the Hanford Purex plant (Reference 10).

**By difference, page 251, Reference 9.

The cost estimator quoted \$500/man-day⁽⁸⁾. This compares with \$516/man-day for operating cost versus operating staff for the AGNS separations plant⁽¹⁰⁾, given the facility costs and resultant man-power requirement.

Comparing the two estimates, the values in Table E.3.3 are about a factor of two lower than those in Table E.3.4. For conservative estimate, the higher values are used in Table E.3.5 for accident frequencies.

Table E.3.4
MANPOWER REQUIREMENT BASED ON \$500/MAN-DAY

Facility	Assumed Structure Cost (\$)	Manpower (Man-hours)
FRP	5×10^8	8×10^6
FFP	2×10^8	3.2×10^6
FRP-WTF	1×10^9	1.6×10^7
FFP-WTF	2×10^8	3.2×10^6

Table E.3.5
ESTIMATED ACCIDENT FREQUENCIES BASED ON
MANHOURS OF CONSTRUCTION IN TABLE E.3.4

Facility	Lost Time Injuries*	Fatalities
FRP	39	0.3
FFP	14	0.1
FRP-WTF	84	0.7
FFP-WTF	16	0.2

*Assumed equivalent to 97 days or 776 hours lost time per injury (Table E.3.2)

E.3.3 Plant Operations

Table E.3.6 gives estimates of accident frequencies resulting in lost time injuries and fatalities for the fuel cycle facilities of concern in this study. The estimates are based on 28 year averages for DOE.

Table E.3.6
 EXPECTED LOST TIME INJURIES AND FATALITIES
 FOR ONE-YEAR OPERATION OF FUEL CYCLE FACILITIES

Facility	Lost Time Injury/yr	Fatalities/yr
FRP (ref & PT)	1.3	0.014
FFP (ref & PT)	1.3	0.014
FRP-WTF	1.3	0.015
FFP-WTF	0.3	0.003

activities. Previously assumed work force levels are used, these being 300 (FRP), 310 (FF), 300 (FRP-WTF), and 120 (FFP-WTF). These are multiplied by the values previously cited (Table E.3.1).

E.3.4 Decommissioning of Fuel Cycle Facilities

An estimate of nonradiological risk for decommissioning an FRP is given in NUREG 0278, of 1.7 lost-time injuries and 9.1×10^{-3} fatalities for a five year decommissioning effort requiring about 142 man years of labor⁽⁶⁾. The study examined an AGNS-type plant with liquid products. The cost estimate was about \$635 per cubic meter of contaminated building volume.

Between the Reference and PT cases, the only significant difference in decommissioning results from the addition of the WTFs. For both, it is assumed that:

- the cost per m^3 of contaminated building volume is approximately \$635
- the ratio of labor to total cost is constant
- The relationship between costs of plants of different capacities is governed by a 0.36 exponential of the ratio of capacities⁽¹¹⁾

Based on these assumptions, the values given below are derived.

Table E.3.7
NONRADIOLOGICAL RISKS ASSOCIATED WITH DECOMMISSIONING

Facility	Lost Time Injuries	Fatalities
FRP	5.1	2.7×10^{-2}
FRP-WTF	16	6×10^{-2}
FFP	1.7	9.1×10^{-3}
FFP-WTF	3.2	1.7×10^{-2}
TOTAL	26	1.1×10^{-1}

E.4 CONVENTIONAL TRANSPORTATION ACCIDENTS

Transportation accidents involving collisions represent a fatality risk to both workers and the general public. It is assumed that onsite traffic control by speed limitations and temporary barriers at crossing for onsite movement precludes serious accident risk. Offsite accident frequencies are assumed to be comparable to conventional rail freight (for casks) and conventional trucking (for other wastes).

The accident statistics for 1974 were taken from Reference 12 and are summarized in Table E.4.1.

Table E.4.1
ASSESSED FATALITY FREQUENCY FOR TRAIN SHIPMENTS

Activity	Fatalities (1974)	Fatalities* per Freight Mile
Train	139	2.6×10^{-7}
Train Services	1692	3.2×10^{-6}
Non Train	77	1.4×10^{-7}
TOTAL	1908	3.6×10^{-6}

*Based on combined mileage of 6.4×10^7 passenger-miles and 4.69×10^8 freight-miles due to combined freight and passenger shipments.

For freight shipments by truck, Reference 13 gives 3×10^{-6} accidents/truck-mile (p.64). The frequency of fatalities per accident is 0.03 fatalities/truck-accident. This results in 9×10^{-8} fatalities per truck mile.

REFERENCES

1. Handbook of Laboratory Safety, 2nd Ed., N.V. Steere, Ed., The Chemical Rubber Company (1976).
2. Allied Gulf Nuclear Services, Inc., "Safety Analysis Report - Barnwell Nuclear Fuel Plant," USNRC Docket 50-332, Tables 8.8-1, 8.9-2, and Figure 5.24-1, October (1973).
3. Proceedings of the Fourth International Clean Air Congress, Editors: S. Kasuga, N. Suzuki, T. Yamada, G. Kimura, K. Onue, The Japanese Union of Air Prevention Association, (1977), Japan.
4. E.G. Walther, D. Comarow, "How to Calculate the Importance of Air Pollutants and Their Sources," Ref. (1), p. 973.
5. Comparative High-Cost Benefit Study of Affirmative Sources of Electrical Energy, WASH-1224, USAEC, December (1974).
6. Schneider, K.J., and C.E. Jenkins, "Technology, Safety, and Costs of Decommissioning a Reference Nuclear Fuel Reprocessing Plant," NUREG-0278, prepared by Battelle Pacific Northwest Laboratory for USNRC, October 1977.
7. United States Atomic Energy Commission, "Operational Accidents and Radiation Exposure Experience," WASH-1192, Division of Operational Safety, USAEC, Fall 1975.
8. Private communication, Marvin Schneller to R.R. Jackson, June 6, 1979.
9. Godfrey, Rober S., Ed., "Building Systems Cost Guide 1978," 3rd Edition, Robert Snow Means Company, Inc., 1978.
10. Yoder, R.A. et al., "Estimated Facility Cost for Remote Maintenance -- Task 3," SAI-006-78-RI, SAI-RI, November 1978.
11. SAI, "Analysis of Facility Requirements for the Processing of Nuclear Weapon Material," SAI-77-768-LJ, June 24, 1977, p. 72.
12. Bureau of Census, "Statistical Abstract of the United States - 1976", 97th edition, U.S. Department of Commerce, U.S. Government Printing Office, 1976.
13. National Safety Council, "Safety Facts," 1976.
14. Christiansen, E.H. et al., "Registry of Toxic Effect of Chemical Substances" 1975 edition, U.S. HEW Department, N105H, June 1975.

APPENDIX F

PROBABILITY OF CASK PENETRATION*

With the Sandia demonstrations of cask integrity against locomotive and abutment impact (Yoshimura⁽²⁾), the possibility of cask penetration remains as a possible failure mode. There are two classes of penetrators - missiles having sufficient kinetic energy to penetrate a cask before coming to rest, or stationary probes that penetrate a cask due to the kinetic energy of the cask. These are investigated in two approximations using ballistic penetration formulae.

While there are several penetration formulae available, the simple and apparently conservative formula of Thomson⁽³⁾ that appears in Clarke, et al.⁽⁴⁾ is used here. It is:

$$T = \frac{24(mV^2/2)}{Y \pi R^2} \quad (1)$$

where T is the cask thickness (in) to be penetrated, $mV^2/2$ is the energy available (ft-lb), πR^2 is the area of the penetrator (in²) and Y is the yield strength (psi) of the material being penetrated.

Case I Missile Penetration

For this case, the length of the missile is not critical and it is convenient to use the parameter $\lambda = L/R$, i.e., the ratio of the missile length to radius. Using this parameter and the specific gravity of the missile (ρ),

$$m = 3.545 \times 10^{-3} \rho \lambda R^3 \quad (2)$$

With this, Eq. 1 may be rewritten in terms of velocity as:

*This Appendix uses the methods of Fullwood, et al.⁽¹⁾

$$V = \left(\sqrt{\frac{\pi TY}{0.0425 \rho R}} \right) \lambda^{-\frac{1}{2}} = \frac{a}{\sqrt{\lambda}} \quad (3)$$

The train velocity probability distribution data presented in Clarke, et al.⁽⁴⁾ may be approximated as

$$P(\text{velocity} > V) = (1 - V/120)^4.$$

This is applicable to speeds less than 120 ft/sec. Furthermore, the probability of a missile not bending⁽⁴⁾ is $\sin^{-1}(0.25/\lambda)/25^0$ which is approximated as $1/\lambda$; hence, the probability of penetration is

$$P = 1/\lambda (1 - V/120)^4; \quad (4)$$

using Eq. 3, this becomes

$$P = 1/\lambda \left(1 - \frac{a}{120\sqrt{\lambda}} \right)^4 \quad (5)$$

Data from train wrecks⁽⁴⁾ indicates that missiles ranging from 1-3 inches diameter occur in 5.5% of the accidents. Thus the missile radius is somewhat known but not the missile length, which is also needed to specify λ . A further constraint is applied by maximizing the probability. This occurs when $a/40 = \lambda$; hence, Eq. 5 becomes

$$P = \frac{1422}{a^2} = \frac{60\rho^R}{\pi TY} \quad (6)$$

Equation 6 is the probability of penetration if a train-wreck associated missile impacts the cask. The probability of a train wreck is $1.5 \times 10^{-6}/\text{car-mile}$ ⁽⁴⁾ and missiles occur in 5.5% of the cases, the probability of penetration is

$$P(\text{missile penetration/car-mile}) = \frac{1.6 \times 10^{-6} \rho^R}{TY} \quad (7)$$

Since steel is the most likely missile capable of penetration $\rho = 7.75$, the average missile is 0.75 inches in radius and the yield strength of stainless steel is 9×10^4 psi; hence,

$$P(\text{missile penetration/car-mile}) = \frac{1.0 \times 10^{-10}}{T(\text{in})} \quad (8)$$

Case II Stub Penetration

If the energy for penetration comes from the cask motion, the formulation is somewhat simpler. Equation 1 using 200,000 lbs as the cask weight becomes

$$V = \frac{R\sqrt{TY}}{155} \quad (9)$$

Using Eq. 4,

$$P(\text{stub}) = \frac{R}{L} \left(1 - \frac{R\sqrt{TY}}{18600} \right)^4$$

using the previous parameters

$$P(\text{stub penetration/car-mile}) = \frac{6 \times 10^{-8}}{L} \left(1 - \frac{\sqrt{T}}{83} \right)^4 \quad (10)$$

where it is conservatively assumed that all missiles are stubs.

Application to the Reference and PT Cask

Casks are not homogenous and it is necessary to estimate the stainless steel equivalent thickness of the other cask materials. This is done using Eq. 1 by noting that the product YT contains the cask characterization. For example, penetration of the Reference Cask requires going through 3.5 inches of stainless steel and 6.0 inches of lead. Using this yield strength scaling, the equivalent thickness is $3.5 + 6.0 \times 3,000/90,000 = 3.7$ inches of stainless steel equivalent. In this manner, the following table is prepared. It should be noted that the probe length is set by the cask geometry. It is also believed that this analysis is conservative in the mathematical approximations, in the use of flat plate formulae on curved surfaces in estimating the probability of missiles and probes and by not considering the probability of probes of considerable length.

Table F.1

SUMMARY OF CASK PENETRATION PROBABILITIES IN RAIL ACCIDENTS

<u>Cask</u>	Stainless Steel Equivalent Thickness (in)	Probe Length (in)	<u>Penetration Probability (/c-mi)</u>		
			<u>Missile</u>	<u>Probe</u>	<u>Combined</u>
Reference Category 1	0.75	3.75	1.3×10^{-10}	1.5×10^{-8}	1.5×10^{-8}
Reference Category 2-4	3.6	21.75	3×10^{-11}	3×10^{-9}	3×10^{-9}
PT Category 2-4	5.0	50.0	2×10^{-11}	1.1×10^{-9}	1.1×10^{-9}
HLW Category 1	2.0	7.0	5×10^{-11}	8×10^{-9}	8×10^{-9}
HLW Category 2-4	3.3	22.3	3×10^{-11}	3×10^{-9}	3×10^{-9}
Cladding Hulls 3-4	1.25	5.4	8×10^{-11}	1×10^{-8}	1×10^{-8}

References to Appendix F

1. R. R. Fullwood, E. A. Straker, D. Moffat, W. Aron and P. Lobner, "Preliminary Risk Analysis of the Transportation of Nuclear Waste," SAI-032-76-PA, December 1976.
2. H. R. Yoshimura, "Crash Testing of Spent Nuclear Fuel Shipping Systems," SAND 78-0451C, April 1978.
3. W. T. Thomson, "An Approximate Theory of Armor Penetration," J. Appl. Physics 26, 1, p. 80, January 1955.
4. R. K. Clarke, J. T. Foley, W. F. Hartman and D. W. Larson, "Severities of Transportation Accidents," SLA-74-0001, July 1976.

APPENDIX G

ROUTINE RADIOLOGICAL PUBLIC RISK

Tables G-1 and G-2 present the calculated isotopic releases. These were used as input to a modified form of CRAC (Appendix C) to determine the annual population doses and health effects as presented in Table G-3. The fraction of material leaving the plant is considered to be 1 for ^3H , 0.01 for ^{85}Kr , and 0.001 for ^{129}I . The fractions for the particulates are those for HEPA filters given in Appendix A.6 with 0.1 credit being given to the deep bed filter used in the FRP, FRP-WTF, and the FFP-WTF.

Table G-1

CALCULATED ANNUAL ROUTINE RELEASES FROM THE FRP AND FRP-WTF (Ci/Year)
 (Filter Penetration Fractions in Accordance With
 Appendix A.6 - 6×10^{-10})

Isotopes	FRP (Reference)	FRPF (PT)	FRP-WTF
^3H	1.2×10^6	1.2×10^6	
^{85}Kr	1.5×10^5	1.6×10^5	
^{129}I	7.3×10^{-2}	7.5×10^{-2}	
^{238}Pu	4.6×10^{-2}	4.4×10^{-2}	8.9×10^{-4}
^{239}Pu	6.6×10^{-4}	7.7×10^{-4}	1.5×10^{-5}
^{240}Pu	1.9×10^{-3}	2.0×10^{-3}	5.5×10^{-5}
^{241}Pu	0.47	0.48	1.2×10^{-2}
^{242}Pu	1.5×10^{-5}	2.2×10^{-5}	4.3×10^{-7}
$^{242\text{m}}\text{Am}$	5.9×10^{-5}	6.5×10^{-5}	6.5×10^{-5}
^{242}Cm	2.7×10^{-2}	4.4×10^{-2}	4.4×10^{-2}
^{243}Cm	1.5×10^{-4}	4.4×10^{-4}	4.4×10^{-4}
^{244}Cm	3.0×10^{-2}	0.18	0.18
^{90}Sr	7.4×10^{-2}	7.5×10^{-2}	7.5×10^{-2}
^{106}Ru	0.35	0.37	0.37
^{134}Cs	0.11	0.13	0.13
^{137}Cs	0.13	0.13	0.13
^{144}Ce	0.40	0.41	0.41
^{147}Pm	0.12	0.12	0.12
^{154}Eu	1.2×10^{-2}	1.4×10^{-2}	1.4×10^{-2}
^{155}Eu	8.1×10^{-3}	9.2×10^{-3}	9.3×10^{-3}

(^3H DF = 1, ^{85}Kr DF = 100, ^{129}I DF = 100)

Table G-2

CALCULATED ANNUAL ROUTINE RELEASES FROM THE FFP AND FFP-WTF (Ci/Year)
 (Filter Penetration Fractions in Accordance With Appendix A.6 -
 2×10^{-12} FFP and 6×10^{-10} FFP-WTF)

Actinides	FFP (Reference)	FFP (PT)	FFP-WTF
^{234}U	1.4×10^{-10}	3.5×10^{-11}	1.6×10^{-9}
^{235}U	1.2×10^{-12}	1.2×10^{-12}	5.2×10^{-11}
^{236}U	3.5×10^{-11}	3.5×10^{-11}	1.6×10^{-9}
^{237}U	1.9×10^{-11}	1.8×10^{-11}	8.3×10^{-10}
^{237}Np	2.5×10^{-11}	2.4×10^{-11}	1.1×10^{-9}
^{238}Pu	2.4×10^{-10}	1.4×10^{-10}	3.3×10^{-8}
^{239}Pu	7.1×10^{-6}	6.0×10^{-6}	2.7×10^{-4}
^{240}Pu	1.2×10^{-7}	1.3×10^{-7}	5.7×10^{-6}
^{241}Pu	3.0×10^{-7}	3.2×10^{-7}	1.5×10^{-5}
^{242}Pu	7.8×10^{-5}	7.5×10^{-5}	3.4×10^{-3}
^{241}Am	2.2×10^{-9}	3.1×10^{-9}	1.4×10^{-7}
$^{242\text{m}}\text{Am}$	1.3×10^{-7}	4.2×10^{-7}	4.3×10^{-4}
^{242}Cm		1.7×10^{-8}	1.7×10^{-5}
^{243}Cm		7.2×10^{-6}	7.4×10^{-3}
^{244}Cm		7.1×10^{-8}	7.3×10^{-5}
		2.2×10^{-5}	2.3×10^{-2}

Table G-3
ROUTINE ANNUAL RADIOLOGICAL POPULATION DOSE
AND HEALTH EFFECTS

Plant or Cycle	Population Dose (Person-Rem/Yr)	Health Effects (Latent Cancers/Yr)
FRP (Reference)	680	0.12
FRP (PT)	730	0.29
FRP-WTF	5.3	0.24
FFP (Reference)	7×10^{-3}	1.9×10^{-4}
FFP (PT)	1.7×10^{-2}	6.8×10^{-4}
FFP-WTF	0.55	0.028
Reference Cycle	690	0.12
PT Cycle	736	0.56

APPENDIX H

SENSITIVITY AND ERROR ANALYSIS

H.1 BACKGROUND

The fuel cycle risk is of the form of a summation over the subrisk categories, r_i , as shown in Equation H1.

$$R = \sum_{i=1}^K r_i \quad (H1)$$

where k is the number of subrisk categories.

If r_i takes on a modified value r_j' then

$$R' = \sum_{i=1}^K r_i (1 - \delta_{ij}) + r_j' \quad (H2)$$

where δ_{ij} is the Kronecker delta; and the difference is

$$\Delta R = r_j - r_j' \quad (H3)$$

and the fractional difference is

$$\frac{\Delta R}{R} = \frac{r_j \frac{\Delta r_j}{r_j}}{R} \quad (H4)$$

This result would also be obtained by partial differentiation, but was done as above to demonstrate the system linearity and hence applicability to large values of Δr_j .

In general, the risk in any mode j is composed of the product of accident probability and consequences

$$r_j = p_j C_j \quad (H4)$$

and general practice results in a single consequence for a failure mode, but the probabilities may be the result of fault tree analysis. Fault tree or other logic descriptor may be factored into minimum cutsets of the form

$$p_j = \sum_{i=1}^{I_j} \prod_{k=1}^{K_j} p_{ik} \quad (H5)$$

However, even this complex form showing redundancies is linear in the individual probabilities. To see this, consider a special case:

$$P = p_i + p_2 p_3 + p_4 p_5 p_6 \quad (H6)$$

Here

$$\frac{\Delta P}{P} = \frac{p_4 p_5 p_6 \frac{\Delta p_6}{p_6}}{P} \quad (H7)$$

hence, it is linear in Δp and large uncertainties in Δp will result in the correct values of ΔR .

H.2 OVERALL SENSITIVITY

Using the methods described in the previous sections, sensitivity analyses were performed on the risk estimates of the two cycles. This is done in a top-down approach whereby the sensitivity is related to the risk level being addressed and the connection to the overall risk is accomplished by multiplication by the ratio of subrisk to total risk. Table H-1 shows the sensitivity of the major categories of risk contributors to the overall risk. As an example of the preceding description for selecting a subtotal risk sensitivity, the table indicates that "Radiological Operations Affecting the Public" has a sensitivity of 0.4 in the Reference cycle. From Table H-1 the radiological subtotal is 2.2×10^{-3} and the risk total is 0.36; hence,

$$\frac{0.4 \times 2.2 \times 10^{-3}}{0.36} = 2 \times 10^{-3} .$$

except for roundoff errors, this is in agreement with the table value of 1.9×10^{-3} for this category's sensitivity to the total risk.

H.3 SENSITIVITY OF RISK CONTRIBUTORS

Using these methods, the sensitivity of the risk contributors is calculated and presented in Table H-1. Table H-2 presents the nonradiological risk sensitivity. In this and in subsequent tables, the item "Relation to Total Risk" is included to provide the relationship for relating the sensitivity of these items to the total risk as was indicated in the previous paragraph. Tables H-3 through H-6 provide the sensitivities of the radiological accidents. H-7 presents the sensitivity of the fault trees in Appendix B to component changes identified either by component labels or in order of appearance on the trees.

H.4 ERROR ANALYSIS

Very few of the sources of data used in this report quoted error bounds thus necessitating the estimation of errors based on the sources of data and anticipated regulatory, plant and process changes that could affect the present estimates. These error estimates are presented in Table H-8. The overall error estimates at the bottom of the table are calculated by the usual rule for error propagation in a linear system, namely, the square root of the sum of the squared products of the sensitivity and the error estimates. This method assumes no form for the distribution fraction but does assume that the errors are independent. In doing this calculation, it was assumed that the error estimates are equal to the square root of the variance. From this, it is estimated that the Reference risk is between 0.2 to 0.8 fatalities/GWe-year and the PT risk is between 0.5 to 2.8 fatalities/year.

While these error estimates are appropriate for estimating the errors in the risk assessment, they cannot be used directly in estimating the ratio of PT to Reference risk because many of the errors are correlated. This problem of correlation in the error estimate was treated by assuming a complete correlation and computing the ratio of risk for each cycle at its extreme value. Thus the ratio of PT to Reference risk is estimated to be in the range 2.5 to 3.6 about the expected value of 3.

Table H-1

FRACTIONAL CHANGE IN TOTAL OR SUBTOTAL RISK FOR
A FRACTIONAL CHANGE IN RISK CONTRIBUTOR

Risk Contributor	Sensitivity to the Subtotal Risk		Sensitivity to the Total Risk	
	Reference	PT	Reference	PT
Nonradiological Operations Affecting the Public	0.8	0.9	0.8	0.9
Nonradiological Accident Affecting the Public	0.2	0.10	0.2	0.10
Nonradiological Accident Affecting the Workers	1.2×10^{-3}	9×10^{-4}	1.2×10^{-3}	9×10^{-4}
Radiological Operations Affecting the Workers	0.6	0.3	3×10^{-3}	1.7×10^{-3}
Radiological Operations Affecting the Public	0.4	0.7	1.9×10^{-3}	3×10^{-3}
Radiological Accidents Affecting the Public	5×10^{-4}	3×10^{-4}	2×10^{-6}	1.3×10^{-6}
Radiological Accidents Affecting the Workers	4×10^{-5}	1.3×10^{-5}	1.7×10^{-7}	7×10^{-8}

Table H-2

LIFETIME AVERAGED NONRADIOLOGICAL RISK SENSITIVITIES

Reference Cycle Facility	Occupational			Public Routine
	Construction	Operation	Decommissioning	
FRP	0.75	0.48	0.75	0.29
FFP	0.25	0.52	0.25	0.71
Relation to Total Risk	0.03	0.0015	0.003	0.74
PT Cycle Facility				
FRP	0.23	0.3	0.25	0.52
FRP-WTF	0.53	0.33	0.55	0.43
FFP	0.08	0.3	0.08	0.001
FFP-WTF	0.15	0.06	0.15	0.05
Relation to Total Risk	0.04	0.0011	0.003	0.62

Table H-3

SENSITIVITY OF FRP RADIOLOGICAL ACCIDENTS
(In Order of None, One and Two HEPA Filter Failures)

Description	Accident Affecting Public		Accident Affecting Workers
	Reference	PT	Reference or PT
H ₂ Fire and Explosion in HAF Tank	-- 4×10^{-6} 1.7×10^{-6}	-- 6×10^{-6} 2×10^{-6}	0.2
Solvent Fire in HA Contractor	-- 3×10^{-5} 1×10^{-5}	-- 3×10^{-5} 4×10^{-6}	0.15
Red Oil Explosion in HLW Concentrator	-- 1.3×10^{-7} 4×10^{-8}	-- 1.7×10^{-5} 9×10^{-8}	3×10^{-3}
Explosion in HLW Calcliner	-- 2×10^{-5} 1×10^{-5}	-- 1.7×10^{-5} 7×10^{-6}	5×10^{-3}
Red Oil Explosion in Fuel Product Concentrator	-- 1.4×10^{-7} 2×10^{-7}	-- 4×10^{-7} 1.7×10^{-7}	9×10^{-5}
Explosion in Fuel Product Denitrator	-- 1×10^{-6} 4×10^{-7}	-- 2×10^{-7} 3×10^{-6}	9×10^{-5}
Criticality in Process Cell	6×10^{-3} -- --	6×10^{-2} -- --	8×10^{-5}
Failure of Krypton Cylinder	1 -- --	1 -- --	5×10^{-4}
Hydrogen Explosion in Reductor	-- 2×10^{-3} 9×10^{-4}	-- 1.9×10^{-3} 9×10^{-4}	0.2
Fuel Assembly Drop	1.3×10^{-2} -- --	1.3×10^{-2} -- --	0.3
Hydrogen Explosion in Fuel Product Denitrator Feed	-- 7×10^{-4} 3×10^{-4}	-- 6×10^{-4} 3×10^{-4}	0.07
Relation to Total Risk	2×10^{-6}	7×10^{-7}	4×10^{-6}

Table H-4

SENSITIVITY OF FFP RADIOLOGICAL ACCIDENTS
(In Order of None, One, and Two HEPA Filter Failures)

Description	Accident Affecting Public		Accident Affecting Workers
	Reference	PT	Reference or PT
Air Crash	0.7 -- --	0.6 -- --	5×10^{-12}
H ₂ Explosion in ROR	5×10^{-8} 2×10^{-6} 8×10^{-6}	6×10^{-8} 4×10^{-6} 2×10^{-6}	0.5
H ₂ Explosion in Sintering	2×10^{-6} 1×10^{-4} 3×10^{-4}	1.8×10^{-6} 1.6×10^{-4} 0.34	0.5
H ₂ Explosion in Wet Scrap	3×10^{-9} 1×10^{-7} 3×10^{-8}	3×10^{-9} 8×10^{-8} 2×10^{-8}	3×10^{-2}
Criticality in Wet Scrap	0.3 -- --	8×10^{-2} 2×10^{-3} --	8×10^{-5}
Powder Shipping Container Spill	3×10^{-10} 1.3×10^{-8} 5×10^{-8}	3×10^{-10} 1×10^{-8} 8×10^{-8}	3×10^{-3}
Exothermic Reaction in Powder Storage	8×10^{-11} 4×10^{-9} 1.3×10^{-3}	1×10^{-10} 4×10^{-9} 1.8×10^{-8}	1.3×10^{-4}
Relation to Total Risk	1.3×10^{-8}	6×10^{-8}	4×10^{-6}

Table H-5

SENSITIVITY OF THE WTF RADIOLOGICAL ACCIDENT
(In Order of None, One and Two HEPA Filters Failing)

Description	Accident Affecting Public		Accident Affecting Workers	
	FRP-WTF	FFP-WTF	FRP-WTF	FFP-WTF
Criticality in Process Cell	1 -- --	1 -- --	1.7×10^{-8}	3×10^{-5}
Hydrogen Explosion in Actinide Bearing Vessel	(a) 1.1×10^{-5} 1.1×10^{-3} 6×10^{-4} (b) 8×10^{-6} 8×10^{-4} 3×10^{-4}	3×10^{-5} 3×10^{-4} 1.5×10^{-2}	8×10^{-2} 8×10^{-2}	0.13 3×10^{-2}
Hydrogen Explosion in Vessel Vent System	4×10^{-5} 4×10^{-3} 1.3×10^{-3}	5×10^{-5} 5×10^{-4} 2×10^{-4}	0.17	3×10^{-2}
Nitrated Organic Explosion in Am Product Concentrator	3×10^{-5} 3×10^{-3} 1.3×10^{-3}	2×10^{-4} 2×10^{-3} 1×10^{-2}	0.5	0.9
Red Oil Explosion in Process Concentrator	4×10^{-6} 4×10^{-4} 1.3×10^{-3}	7×10^{-6} 7×10^{-5} 1×10^{-2}	5×10^{-2}	3×10^{-3}
Resin Explosion In CEC Unit	1×10^{-3} 1×10^{-2} 4×10^{-2}	2×10^{-4} 1×10^{-3} 8×10^{-3}	3×10^{-3}	6×10^{-3}
Facility Solvent Fire	8×10^{-4} 6×10^{-2} 6×10^{-2}	1.7×10^{-4} 1.7×10^{-2} 8×10^{-2}	1.7×10^{-3}	3×10^{-3}
Solvent Mist Explosion	1.3×10^{-4} 4×10^{-3} 4×10^{-3}	3×10^{-5} 3×10^{-3} 8×10^{-3}	0.17	3×10^{-2}
Relation to Total Risk	3×10^{-8}	2×10^{-8}	4×10^{-6}	2×10^{-6}

Table H-6
SENSITIVITY OF RADIOLOGICAL TRANSPORTATION RISKS

Transportation Mode	Fuel Cycle	
	Reference	PT
Spent Fuel	0.2	0.3
Fresh Fuel (MOX)	0.08	0.09
Fresh Fuel (U)	0.16	0.2
Cladding Hulls	0.5	0.4
HLW	0.02	0.016
Non-HLW	0.03	0.02
Relation to Total Risk	0.008	0.007

Table H-7
FAULT TREE SENSITIVITY ANALYSIS

Fault Tree Figure Number	Components																							
	C ₁	C ₂	C ₃	C ₄	C ₅	C ₆	C ₇	C ₈	C ₉	C ₁₀	C ₁₁	C ₁₃	C ₁₄	C ₁₅	C ₁₆	C ₁₇	C ₁₉	C ₂₀	C ₂₁	C ₂₆	C ₂₇	C ₂₉	C ₃₀	
B-1.1			1	1	1	3x10 ⁻³																		
B-1.2		1						1	0.3		0.7	1	1				1							
B-1.3	0.9	0.9	0.13	0.13	0	0.5	0.5	1	1	6x10 ⁻³	6x10 ⁻³													
B-1.4				1	1	1	0.9	0.9	0.12	0.12														
B-1.5	1	1	1	1	0.09	0.9																		
B-1.6	1	1	0.14	0.14	0.14	0.2		0.14				0.6	0.6	0.12	0.12	0.08				0.06	0.12	1.4x10 ⁻³	0.08	
B-1.7	4x10 ⁻⁶	4x10 ⁻⁶	4x10 ⁻⁶	4x10 ⁻⁶	0.9	0.9	0.11																	
B-2.1	1	0.14	0.07	0.4	0.4	0.4	1																	
B-2.2	1	1		1	1																			
B-2.3	1	0	0.011	1		0.18	0.18		0.18						0.4	0.012	0.04	0.018	1.8x10 ⁻³	0.18				
B-3.1	2x10 ⁻³	0.5	0.5	8x10 ⁻³	0.5	0.5	1	1	5.2x10 ⁻⁴															
B-3.2	1	1	1.7x10 ⁻⁴	1.7x10 ⁻⁴	1.7x10 ⁻⁴	2x10 ⁻⁴	2x10 ⁻⁴	2x10 ⁻⁴																
B-3.3	1	8x10 ⁻³		0.018		1																		
B-4.1	0.17	0.6	0.2																					
B-4.2	5x10 ⁻⁵	8x10 ⁻⁴		1	6.7x10 ⁻⁵																			
B-4.3	0.01	0.11	4x10 ⁻³																					
B-4.4	2x10 ⁻⁶	6x10 ⁻⁵	1.2x10 ⁻⁶	2x10 ⁻⁶	1		1																	

Table H-8

ESTIMATED ERRORS IN THE RISK CONTRIBUTORS
AND THE ESTIMATED OVERALL ERROR

Risk Contributor or Overall Risk	Estimated Error Factor*
Nonradiological Operations Affecting the Public	2
Nonradiological Accidents Affecting the Public	0.2
Nonradiological Accidents Affecting the Workers	0.2
Radiological Operations Affecting the Workers	0.5
Radiological Operations Affecting the Public	2
Radiological Accidents Affecting the Public	10
Radiological Accidents Affecting the Workers	10
Reference Cycle	2
PT Cycle	2
Reference Cycle Radiological	0.9
PT Cycle Radiological	1.4

*Multiplier or divisor of the quoted value

APPENDIX I
GLOSSARY OF KEY TERMS AND ACRONYMS

Actinides: Radioactive elements with atomic number above 88 (abbreviated as AN).

Activity: A measure of the rate at which radioactive material is emitting radiation; usually given in terms of the number of nuclear disintegrations occurring in a given quantity of material over a unit of time.

ALARA: As low as reasonably achievable. In the nuclear industry all radiation doses are maintained as far below prescribed standards as reasonably achievable.

Alpha particle: A positively charged particle emitted by certain radioactive material. It is made up of two neutrons and two protons; hence it is identical with the nucleus of a helium atom.

Back end of the fuel cycle: Includes spent fuel storage, fuel reprocessing, mixed-oxide fabrication, transportation, and waste management.

Background radiation: The varying radiation of man's natural environment. It results from cosmic rays and from the naturally radioactive elements of the earth, including those from within man's body.

Burial grounds: Areas designated for disposal of containers of radioactive wastes and obsolete or worn-out equipment by near-surface burial.

C: Accident consequences

Calcine: Material heated to a temperature below its melting point to bring about loss of moisture and cause oxidation to a chemically stable form.

Canister: A metal container for radioactive solid waste.

Cask: A container that provides shielding and containment during transportation of radioactive materials.

Cation exchange chromatography: A unit operation utilizing ion exchange to separate chemically similar elements by taking advantage of differences in absorption.

CENT: Centrifuge or centrifugation

CMP: The bidentate extractant (dihexyl-N, N-diethyl carbamyl methylene phosphate); used with a diisopropyl benzene diluent.

Colocated: Refers to location of facilities at a common site, thereby minimizing transportation needs.

Coprocessing: A term referring to a flowsheet where uranium and plutonium are jointly decontaminated and denitrated in contrast to the conventional aqueous separation of plutonium and uranium for subsequent separate purification and oxide conversion.

CRAC: A computer code, calculation of reactor accident consequences, for evaluating population radiological exposures from releases of radionuclides to the environs.

Critical mass: The minimum mass of fissionable material that, with a specified geometrical arrangement, moderation, and material composition, will self-sustain a fission chain reaction.

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 loss of neutrons.

Curie (Ci): A unit of measure of radioactivity; 1 curie = that quantity of any nuclide which undergoes 3.7×10^{10} disintegrations per second.

df: Depletion factor - factor by which an incident aerosol is diminished on passing through a filter.

Decontamination: The selective removal of radioactive material from a surface or from within another material.

Decontamination factor (DF): The ratio of the original contamination level to the contamination level after decontamination.

Design basis accident: A postulated event or sequence of events for which the confinement system must meet its functional goals (ANSI-N-287-1976).

Design basis earthquake: Postulated earthquake of 0.25 g horizontal and 0.16 g vertical acceleration.

Design basis fire: A postulated fire that consumes all flammable and combustible materials enclosed within a fire resistant barrier of at least a 2 hour rating (ANSI-N-287-1976).

Disposal: The planned release of radioactive waste in a manner that includes recovery, or its placement in a manner which is considered permanent so that recovery is not provided for.

Dose: Herein generally means the more rigorous term "dose-equivalent." The latter, expressed in units of rem, implies a consistent basis for estimates of consequential health risk, regardless of rate, quantity, source, or quality of the radiation exposure.

DOT: U.S. Department of Transportation

DTPA: diethylenetriaminepentaacetic acid

2 EH₂O: 2-ethyl hexanol, used as an extractant for detergents and other organics.

Epsilon (ϵ): Used in fault tree quantification to indicate that the event of concern has much lower expected frequency of occurrence than the other events of concern.

FFP: Fuel fabrication plant to produce mixed-oxide fuel pellets (remote operation, mixed remote and contact maintenance).

FFS: Fuel fabrication site, the fuel fabrication plant plus all other on-site facilities that existed before construction of the waste treatment plant.

Fissile material: One of several actinides which under proper conditions result in a chain reaction.

Fission (nuclear): The splitting of a heavy nucleus into two or, rarely, more fragments.

Fission product: A nuclide produced by the fission of a heavier element or the subsequent daughters resulting from radioactive decay of the nuclide thus formed.

Fissionable material: Any material fissionable by neutrons of low energy such as certain isotopes of uranium and plutonium.

Front-end of the fuel cycle: Mining, milling, enrichment, and fabrication of UO₂ fuel.

FRP: Fuel reprocessing plant, either a conventional Purex separations plant or a modified coprocessing Purex plant with safeguards (remote operation, mixed remote and contact maintenance).

FRS: Fuel reprocessing site; the fuel reprocessing plant, a mixed actinide nitrate-to-oxide conversion facility, and all other on-site facilities that existed before construction of the waste treatment plant.

Fuel (nuclear reactor): Fissionable material used as the source of power when placed in a critical arrangement in a nuclear reactor.

Fuel cycle: The complete series of steps involved in supplying fuel for nuclear reactors. It includes mining, refining, enrichment, fabrication of fuel elements, use in a reactor, chemical processing to recover the fissionable material remaining in the spent fuel, reenrichment of the fuel material, refabrication of new fuel elements, and management of radioactive waste and transportation between these facilities.

Fuel element: A tube, rod, or other form into which fissionable material is fabricated for use in a reactor.

Gamma ray: Electromagnetic radiation, similar in nature to x-rays, emitted by the nuclei of some radioactive substances during radioactive decay.

GESMO: Final Generic Environmental Impact Statement (EIS) on the use of recycle plutonium to form mixed oxide fuel for LWR's.

GWe: Gigawatts (billions of watts) of electricity generated. (About 1/3 of the thermal power)

Half-life: The time required for a radioactive substance to decay to half its original value. After a period equal to 10 half-lives, the radioactivity has decreased to about 0.1% of its original value. Biological half-life: time required for half of an ingested (or inhaled) substance to be eliminated by natural processes.

HAW: High activity waste, refers to the raffinate from the Purex HA column (also called HLLW).

Hazard analysis: A systematic identification energy sources and potentially unsafe paths for energy release.

HEPA: High-efficiency particulate air (filter).

HL: High-level in radioactivity ($> 10^4$ rem).

HLLW (High-level liquid waste): The aqueous waste resulting from operation of the first cycle solvent extraction system (or its equivalent) in a facility for reprocessing irradiated reactor fuels as well as concentrated wastes from subsequent cycles.

HLS: High-level solids (dissolver solids and/or cladding hulls).

HLW (High-level waste): DOE management directives define high-level waste to include high-level wastes, products from solidification of high-level waste, and irradiated fuel elements if discarded without reprocessing. A proposed NRC regulation (10 CFR 60.3) defines high-level waste to include irradiated fuel, high-level liquid waste, and products from its solidification. In the GEIS there are instances, however, where discarded spent fuel and high-level waste (as wastes from the reprocessing of spent fuel) are cited separately.

IL: Intermediate-level in radioactivity ($10^3 < \text{rem} < 10^4$).

ILLW: Intermediate-level liquid waste.

Interim storage: Storage operations for which a) monitoring and human control are provided, and b) subsequent action involving treatment, transportation, or final disposition is expected.

Light water reactor (LWR): May be either a BWR or PWR; uses as coolant ordinary water (H_2O) instead of heavy water (D_2O).

LL: Low-level in radioactivity ($0 < \text{rem} < x$).

Long-lived nuclides: Radioactive isotopes with half-lives greater than about 30 years.

Low-level waste (LLW): Waste which is not high-level or transuranic.

Maximum individual: A person whose location and habits tend to maximize his radiation dose.

Maximum Permissible Concentration (MPC): The quantity of radioactive material in air, water, etc., per unit volume or weight, from which a human, if continuously exposed, should not (in the light of present knowledge) sustain appreciable body damage.

Megawatts (MW): Millions of watts.

Mixed-oxide fuel fabrication plant (MOX FFP): Plant where uranium oxide and plutonium oxide are mixed and fabricated into fuel elements for use in nuclear power plants.

MOX: Mixed actinide oxides (primarily U, Np, Pu, Am, and Cm).

MTHM: Metric ton of heavy metal.

MWD/T: Unit of burn-up for a irradiated fuel assembly, megawatt days per ton, which relates power level and core time per weight of fissile and fertile material.

Neutron: Stable particle in a nucleus of very slightly greater mass than a proton but without nuclear change.

NO_x : Oxides of nitrogen.

NRC: Nuclear Regulatory Commission.

Nuclear excursion: Similar to a criticality except the neutron production exceeds the neutron loss that is a power level or K_{eff} greater than 1 exists.

Nucleus: The inner core of the atom, consisting primarily of neutrons and protons, which make up almost the entire mass of the atom but only a minute part of its volume.

Nuclide: A species of atom characterized by its mass number, atomic number, and nuclear energy state, provided that the mean life in that state is long enough to be observable.

p: Probability of failure or accident.

Parent: A radionuclide that upon disintegration yields a specified nuclide, the daughter, either directly or as a later member of a radioactive decay series.

Partition: To separate one element from others; in processing operation, the separation of elements such as uranium and plutonium.

Pressurized water reactor (PWR): A reactor system that uses a pressurized water primary cooling system. Steam formed in a secondary cooling system is used to turn turbines to generate electricity.

PT: Partitioning and transmutation; refers to the separation of actinides from wastes and actinide recycle through an LWR for fissioning to shorter lived products.

Rad: Radiation absorbed dose, the basic unit of absorbed dose of ionizing radiation. A dose of 1 rad means the absorption of 100 ergs of radiation energy per gram of absorbing material.

Red oil: A term applied to a complex mixture of nitrated organic compounds, usually complexed with heavy metal nitrate such as uranium or plutonium. The material is potentially reactive, decomposing autocatalytically and exothermically.

rem (roentgen equivalent man): A quantity used in radiation protection to express the effective dose equivalent for all forms of ionizing radiation. It is the product of the absorbed dose in rads and factors related to relative biological effectiveness.

Repository (Federal): A federally owned and operated facility for storage or disposal of specific types of waste from DOE sites and/or licensees.

Risk: Product of the consequences and the probability of the event's occurrence.

Roentgen: A unit for measuring gamma or "x-ray" radiation. The roentgen is defined by measuring the effect of the radiation on air. It is that amount of gamma or x-rays required to produce ions carrying 1 electrostatic unit of charge in 1 cubic centimeter of dry air under standard conditions.

Seismicity: The tendency for the occurrence of earthquakes.

Shielding: A material interposed between a source of radiation and personnel for protection against the danger of radiation. Common shielding materials are concrete, water and lead.

Shipping cask: A specially designed container used for shipping radioactive materials.

Short-lived nuclides: Radioactive isotopes with half-lives no greater than about 30-years, e.g., ^{137}Cs , ^{90}Sr , ^{85}K , and ^3H .

Solidification: Conversion of liquid radioactive waste to a dry, stable solid.

Spent Fuel (SF): Nuclear reactor fuel that has been used to the extent that it can no longer be used in a nuclear power plant.

Storage: Retention of waste in some type of manmade device in a manner permitting retrieval.

SWT: Salt waste treatment (the 2-ethylhexanol solvent extraction cycle).

TBP: Tributyl phosphate, used as a solvent for recovery of heavy metal nitrates such as uranium, usually.

Transmutation: Conversion of a radioisotope to a shorter-lived or stable isotope by bombarding it with nuclear particles.

Transportation: Movement of materials between sites. Intra-site movement is not considered. Includes alternative methods for packaging, handling, and transport of waste materials and plutonium compounds. Concepts include all conventional methods of land and water transport required by the waste management system.

Transuranic (TRU) elements: Elements with mass number greater than 92. They include, among others, neptunium, plutonium, americium, and curium.

Transuranic waste: Waste material measured or assumed to contain more than a specified concentration of transuranic elements. TRU waste is currently identified as containing 10 nanocuries or more of transuranic activity (usually ^{239}Pu) per gram of waste.

Waste immobilization: Process of converting waste to a stable, solid form which will tie up the radionuclides, thereby preventing (or slowing) their migration to the biosphere.

Waste management: The planning, execution and surveillance of essential functions related to the control of radioactive (and nonradioactive) waste, including treatment, solidification, initial or long-term storage, surveillance, and isolation.

WTF: Waste treatment facility at either a fuel reprocessing or a mixed-oxide fuel fabrication plant (remote operation, mixed remote and contact maintenance).