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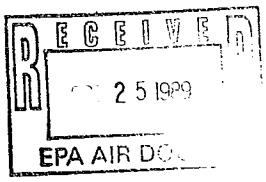

Risk Assessments

Environmental Impact Statement
for NESHAPS Radionuclides

VOLUME 2

BACKGROUND INFORMATION DOCUMENT

September 1989
U.S. Environmental Protection Agency
Office of Radiation Programs
Washington, D.C. 20460

Preface

The Environmental Protection Agency is promulgating National Emission Standards for Hazardous Air Pollutants (NESHAPs) for Radionuclides. An Environmental Impact Statement (EIS) has been prepared in support of the rulemaking. The EIS consists of the following three volumes:

VOLUME I - Risk Assessment Methodology

This document contains chapters on hazard identification, movement of radionuclides through environmental pathways, radiation dosimetry, estimating the risk of health effects resulting from expose to low levels of ionizing radiation, and a summary of the uncertainties in calculations of dose and risks.

VOLUME II - Risk Assessments

This document contains a chapter on each radionuclide source category studied. The chapters include an introduction, category description, process description, control technology, health impact assessment, supplemental control technology, and cost. It has an appendix which contains the inputs to all the computer runs used to generate the risk assessment.

VOLUME III - Economic Assessment

This document has chapters on each radionuclide source category studied. Each chapter includes an introduction, industry profile, summary of emissions, risk levels, the benefits and costs of emission controls, and economic impact evaluations.

Copies of the EIS in whole or in part are available to all interested persons; an announcement of the availability appears in the Federal Register.

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TABLE OF CONTENTS

VOLUME II: RISK ASSESSMENT

LIST OF TABLES ix

LIST OF FIGURES. xxxiii

1. INTRODUCTION 1-1

2. DEPARTMENT OF ENERGY (DOE) FACILITIES. 2-1
   2.1 OVERVIEW AND SUMMARY OF RESULTS. 2-1
   2.2 RMI COMPANY. 2-21
   2.3 LOS ALAMOS NATIONAL LABORATORY 2-24
   2.4 HANFORD RESERVATION. 2-33
   2.5 OAK RIDGE RESERVATION. 2-42
   2.6 SAVANNAH RIVER PLANT 2-51
   2.7 FEED MATERIALS PRODUCTION CENTER 2-58
   2.8 BROOKHAVEN NATIONAL LABORATORY 2-66
   2.9 MOUND FACILITY 2-71
   2.10 IDAHO NATIONAL ENGINEERING LABORATORY 2-72
   2.11 LAWRENCE BERKELEY LABORATORY 2-79
   2.12 PADUCAH GASEOUS DIFFUSION PLANT. 2-82
   2.13 LAWRENCE LIVERMORE LABORATORY 2-84
   2.14 PORTSMOUTH GASEOUS DIFFUSION PLANT. 2-86
   2.15 ARGONNE NATIONAL LABORATORY. 2-89
   2.16 PINELLAS PLANT 2-92
   2.17 NEVADA TEST SITE 2-94
   2.18 KNOLLS LABORATORY - KESSELRING 2-96
   2.19 BATTELLE COLUMBUS LABORATORY 2-98
   2.20 FERMI NATIONAL LABORATORY. 2-100
   2.21 SANDIA NATIONAL LABORATORY 2-103
   2.22 BETTIS ATOMIC POWER LABORATORY 2-106
   2.23 KNOLLS LAB - WINDSOR 2-108
   2.24 ROCKY FLATS PLANT 2-109
   2.25 PANTEX PLANT 2-112
   2.26 KNOLLS LAB - KNOLLS. 2-114
   2.27 AMES LABORATORY. 2-117
   2.28 ROCKETDYNE ROCKWELL. 2-118
   2.29 REFERENCES 2-122

3. NRC-LICENSED AND NON-DOE FEDERAL FACILITIES. 3-1
   3.1 INTRODUCTION AND BACKGROUND 3-1
   3.2 HOSPITALS 3-2
   3.3 RADIOPHARMACEUTICAL MANUFACTURERS 3-6
   3.4 LABORATORIES. 3-10
   3.5 RESEARCH AND TEST REACTORS. 3-13
   3.6 SEALED SOURCE MANUFACTURERS 3-16
   3.7 NON-LWR FUEL FABRICATORS. 3-20
   3.8 SOURCE MATERIAL LICENSEES 3-23
   3.9 LOW-LEVEL WASTE INCINERATORS. 3-25
   3.10 NON-DOE FEDERAL FACILITIES. 3-28

3.11 SUMMARY OF THE COLLECTIVE RISKS FROM ALL FACILITIES 3-30
3.12 REFERENCES. 3-32

4. URANIUM FUEL CYCLE FACILITIES. 4-1
4.1 INTRODUCTION 4-1
4.2 URANIUM MILLS 4-3
4.3 URANIUM CONVERSION FACILITIES 4-25
4.4 FUEL FABRICATION FACILITIES 4-31
4.5 NUCLEAR POWER FACILITIES. 4-40
4.6 SUMMARY 4-67
4.7 REFERENCES. 4-69

5. HIGH-LEVEL WASTE DISPOSAL FACILITIES 5-1
5.1 DESCRIPTION OF THE HIGH-LEVEL WASTE DISPOSAL FACILITIES. 5-1
5.2 BASIS OF THE EXPOSURE AND RISK EVALUATION 5-4
5.3 RESULTS OF THE DOSE AND RISK ASSESSMENT 5-7
5.4 SUPPLEMENTARY CONTROL OPTIONS AND COSTS 5-9
5.5 REFERENCES. 5-11

6. ELEMENTAL PHOSPHORUS PLANTS. 6-1
6.1 DESCRIPTION OF THE SOURCE CATEGORY. 6-1
6.2 BASIS OF THE EXPOSURE AND RISK ASSESSMENT 6-3
6.3 RESULTS OF THE EXPOSURE AND RISK ASSESSMENT 6-13
6.4 SUPPLEMENTARY CONTROL OPTIONS AND COSTS 6-18
6.5 REFERENCES. 6-23

7. COAL-FIRED UTILITY AND INDUSTRIAL BOILERS. 7-1
7.1 INTRODUCTION. 7-1
7.2 UTILITY BOILERS 7-5
7.3 INDUSTRIAL BOILERS. 7-19
7.4 REFFRENCES. 7-25

8. INACTIVE URANIUM MILL TAILINGS 8-1
8.1 DESCRIPTION OF INACTIVE URANIUM MILL TAILINGS SITES 8-1
8.2 BASIS OF THE EXPOSURE AND RISK ASSESSMENT 8-3
8.3 RESULTS OF THE RISK ASSESSMENT FOR INACTIVE MILLS . .8-5
8.4 SUPPLEMENTARY CONTROL OPTIONS AND COSTS 8-17
8.5 REFERENCES. 8-30

9. LICENSED URANIUM MILL TAILINGS FACILITIES. 9-1
9.1 DESCRIPTION OF LICENSED URANIUM MILL TAILINGS 9-1
9.2 BASIS OF THE EXPOSURE AND RISK ASSESSMENT 9-5
9.3 RESULTS OF THE RISK ASSESSMENTS FOR LICENSED MILLS. 9-12
9.4 SUPPEMENTARY CONTROL OPTIONS AND COSTS. 9-26
9.5 REFERENCES. 9-52

10. DEPARTMENT OF ENERGY RADON SITES 10-1
10.1 SITE DESCRIPTIONS. 10-1
10.2 BASIS OF THE RISK ASSESSMENT 10-9
10.3 RESULTS OF THE RISK ASSESSMENT 10-11
10.4 SUPPLEMENTARY CONTROL OPTIONS AND COST 10-20
10.5 REFERENCES 10-23

11. UNDERGROUND URANIUM MINES. 11-1
11.1 GENERAL DESCRIPTION. 11-1
11.2 BASIS OF THE EXPOSURE AND RISK ASSESSMENT. 11-5
11.3 RESULTS OF THE EXPOSURE AND RISK ASSESSMENT. . . .11-10
11.4 SUPPLEMENTARY CONTROL OPTIONS AND COSTS. 11-14
11.5 REFERENCES 11-30

12. SURFACE URANIUM MINES. 12-1
12.1 GENERAL DESCRIPTION. 12-1
12.2 BASIS OF THE DOSE AND RISK ASSESSMENT. 12-12
12.3 RESULTS OF THE DOSE AND RISK ASSESSMENT. 12-12
12.4 SUPPLEMENTARY CONTROL OPTIONS AND COSTS. 12-19
12.5 REFERENCES 12-21

13. PHOSPHOGYPSUM STACKS 13-1
13.1 SOURCE CATEGORY DESCRIPTION. 13-1
13.2 RADIONUCLIDE EMISSIONS 13-8
13.3 RESULTS OF THE HEALTH IMPACT ASSESSMENT. 13-21
13.4 SUPPLEMENTARY CONTROL OPTIONS AND COSTS. 13-27
13.5 REFERENCES 13-39

LIST OF TABLES

VOLUME II: RISK ASSESSMENT

Table 2.1-1. Department of energy facilities. 2-2

Table 2.1-2. Summary of doses and risks to nearby individuals from DOE facilities due to 1986 emissions. 2-3

Table 2.1-3. Distribution of fatal cancer risk in the population. 2-7

Table 2.1-4. Summary of doses and risks to the regional population (0-80 km) around DOE facilities . . .2-8

Table 2.1-5. Baseline risk assessment for DOE facilities. . 2-14

Table 2.1-6. Risks when emissions are limited to 3 mrem/y EDE. 2-14

Table 2.1-7 Risks when emissions are limited to 1 mrem/y EDE. 2-15

Table 2.1-8 Maximum individual risk, with Alternative 4 supplemental control strategies. 2-16

Table 2.1-9 Fatal cancers/year to nearby individuals with Alternative 4 supplemental control techniques 2-19

Table 2.1-10 Distribution of fatal cancer risk in the populations within 80 km with Alternative 4 supplemental control techniques. 2-21

Table 2.2-1. Radionuclides released to air during 1986 from RMI. 2-22

Table 2.2-2. Estimated radiation dose rates from RMI. . . . 2-23

Table 2.2-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from RMI. 2-23

Table 2.3-1. Radionuclides released to air during 1986 from Los Alamos Scientific Laboratory . . 2-28

Table 2.3-2. Estimated radiation doses from the Los Alamos Laboratory. 2-29

Table 2.3-3. Estimated fatal cancer risks from the Los Alamos Laboratory. 2-29

Table 2.3-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) gopulation from the Los Alamos Scientific Laboratory 2-29

Table 2.3-5. Effects of holdup time on the release of air activation products from the proposed stack serving the LAMPF beam stop 2-30

Table 2.4-1. Radionuclides released to air during 1986 from the Hanford Reservation 2-40

Table 2.4-2. Estimated radiation dose rates from the Hanford Reservation. 2-41

Table 2.4-3. Estimated fatal cancer risks from the Hanford Reservation. 2-41

Table 2.4-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Hanford Reservation. . . . 2-41

Table 2.5-1. Radionuclides released to air from Oak Ridge Reservation during 1986. 2-44

Table 2.5-2. Estimated radiation dose rates from the Oak Ridge National Laboratory. 2-45

Table 2.5-3. Estimated fatal cancer risks from the Oak Ridge National Laboratory. 2-45

Table 2.5-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Oak Ridge National Laboratory 2-45

Table 2.5-5. Anticipated new emission rate for tritium at CRGDF 2-46

Table 2.5-6. Summary of capital and operating costs for supplementary controls at the Oak Ridge Reservation. 2-50

Table 2.6-1. Radionuclides released to air during 1986 from Savannah River Plant. 2-54

Table 2.6-2. Estimated radiation dose rates from the Savannah River Plant 2-55

Table 2.6-3. Estimated fatal cancer risks from the Savannah River Plant 2-55

Table 2.6-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Savannah River Plant . . . 2-56

Table 2.7-1. Radionuclides released to air during 1986 from FMPC 2-59

Table 2.7-2. Estimated radiation dose rates from FMPC . . . 2-60

Table 2.7-3. Estimated fatal cancer risks from FMPC 2-60

Table 2.7-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from FMPC 2-61

Table 2.7-6. Cost estimates for acquisition and installation of HEPA filter systems. 2-64

Table 2.8-1. Radionuclide emission points stacks at Brookhaven National Laboratory 2-67

Table 2.8-2. Radionuclides released to air during 1986 from Brookhaven National Laboratory. 2-69

Table 2.8-3. Estimated radiation dose rates from the Brookhaven National Laboratory 2-70

Table 2.8-4. Estimated fatal cancer risks from the Brookhaven National Laboratory 2-70

Table 2.8-5. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Brookhaven National Laboratories 2-71

Table 2.9-1. Radionuclides released to air during 1986 from Mound Facility. 2-72

Table 2.9-2. Estimated radiation dose rates from the Mound Facility 2-73

Table 2.9-3. Estimated fatal cancer risks from the Mount Facility 2-73

Table 2.9-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Mound Facility 2-73

Table 2.10-1. Radionuclides released to air during 1986 from all Idaho Facilities. 2-78

Table 2.10-2. Estimated radiation dose rates from the Idaho National Engineering Laboratory. 2-79

Table 2.10-3. Estimated ratal cancer risks from the Idaho National Engineering Laboratory. 2-79

Table 2.10-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from INEL facilities. 2-79

Table 2.11-1. Radionuclides released to air during 1986 from Lawrence Berkeley Laboratory. 2-81

Table 2.11-2. Estimated radiation dose rates from the Lawrence Berkeley Laboratory 2-82

Table 2.11-3. Estimated fatal cancer risks from the Lawrence Berkeley Laboratory 2-82

Table 2.11-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Lawrence Berkeley Laboratory 2-82

Table 2.12-1. Radionuclides released to air during 1986 from Paducah Gaseous Diffusion Plant 2-83

Table 2.12-2. Estimated radiation dose rates from the Paducah Gaseous Diffusion Plant. 2-84

Table 2.12-3. Estimated fatal cancer risks from the Pacucah Gaseous Diffusion Plant. 2-84

Table 2.12-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Paducah Gaseous Diffusion Plant. 2-84

Table 2.13-1. Source terms and release point characterization 2-85

Table 2.13-2. Estimated radiation dose rates from Lawrence Livermore Laboratory/Sandia Livermore. 2-86

Table 2.13-3. Estimated fatal cancer risks from Lawrence Livermore Laboratory/Sandia Livermore. 2-86

Table 2.13-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Lawrence Livermore Laboratory/Sandia Livermore. 2-86

Table 2.14-1. Radionuclides released to air during 1986 from the Portsmouth Gaseous Diffusion Plant. 2-88

Table 2.14-2. Estimated radiation dose rates from the Portsmouth Gaseous Diffusion Plant 2-89

Table 2.14-3 Estimated fatal cancer risks from the Portsmouth Diffusion Plant 2-89

Table 2.14-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Portsmouth Gaseous Diffusion Plant. 2-89

Table 2.15-1. Radionuclides released to air during 1986 from Argonne National Laboratory 2-90

Table 2.15-2. Estimated radiation dose rates from the Argonne National Laboratory. 2-91

Table 2.15-3. Estimated fatal cancer risks from the Argonne National Laboratory. 2-91

Table 2.15-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Argonne National Laboratory 2-92

Table 2.16-1. Radionuclides released to air during 1986 from Pinellas Plant. 2-92

Table 2.16-2. Estimated radiation dose rates from the Pinellas Plant 2-93

Table 2.16-3. Estimated fatal cancer risks from the Pinellas Plant 2-93

Table 2.16-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Pinellas Plant 2-94

Table 2.17-1. Radionuclides released to air during 1986 from the Nevada Test Site. 2-95

Table 2.17-2. Estimateed radiation dose rates from the Nevada Test Site 2-95

Table 2.17-3. Estimated fatal cancer risks from the Nevada Test Site 2-96

Table 2.17-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Nevada Test Site 2-96

Table 2.18-1. Radionuclides released to air during 1986 from Knolls Atomic Power Lab-Kesselring. . . . 2-97

Table 2.18-2. Estimated radiation dose rates from the Knolls Lab-Kesselring. 2-98

Table 2.18-3. Estimated fatal cancer risks from the Knolls Lab-Kesselring. 2-98

Table 2.18-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Knolls Atomic Power Lab-Kesselring 2-98

Table 2.19-1. Radionuclides released to air during 1986 from Battelle Columbus 2-100

Table 2.19-2. Estimated radiation dose rates from the Battelle Columbus Laboratory 2-101

Table 2.19-3. Estimated fatal cancer risks from the Battelle Columbus Laboratory 2-101

Table 2.19-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Battelle Columbus. 2-101

Table 2.20-1. Radionuclides released to air during 1986 from Fermi National Accelerator Laboratory 2-102

Table 2.20-2. Estimated radiation dose rates from the Fermi National Laboratory. 2-103

Table 2.20-3. Estimated fatal cancer risks from the Fermi National Laboratory. 2-103

Table 2.20-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Fermi National Laboratory 2-103

Table 2.21-1. Radionuclides released to air during 1986 from Sandia National Laboratory/Lovelace Research Institute 2-104

Table 2.21-2. Estimated radiation dose rates from the Sandia National Laboratory/Lovelace Research Institute. 2-105

Table 2.21-3. Estimated fatal cancer risks from the Sandia National Laboratory/Lovelace Research Institute. 2-105

Table 2.21-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Sandia National Laboratory/Lovelace Research Institute 2-105

Table 2.22-1. Radionuclides released to air during 1986 from Bettis Atomic Power Laboratory. 2-106

Table 2.22-2. Estimated radiation dose rates from the Bettis Atomic Power Laboratory 2-107

Table 2.22-3. Estimated fatal cancer risks from the Bettis Atomic Power Laboratory 2-107

Table 2.22-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Bettis Atomic Power Laboratory 2-107

Table 2.23-1. Radionuclides released to air during 1986 from Knolls Atomic Power Lab-Windsor 2-108

Table 2.23-2. Estimated radiation dose rates from the Knolls Lab-Windsor 2-109

Table 2.23-3. Estimated fatal cancer risks from the Knolls Lab-Windsor 2-109

Table 2.23-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Knolls Atomic Power Lab-Windsor. 2-110

Table 2.24-1. Radionuclides released to air during 1986 from Rocky Flats Plant 2-111

Table 2.24-2. Estimated radiation dose rates from the Rocky Flats Plant. 2-112

Table 2.24-3. Estimated fatal cancer risks from the Rocky Flats Plant. 2-112

Table 2.24-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Rocky Flats Plant. 2-112

Table 2.25-1. Radionuclides released to air during 1986 from the Pantex Plant. 2-113

Table 2.25-2. Estimated radiation dose rates from the Pantex Plant 2-114

Table 2.25-3. Estimated fatal cancer risks from the Pantex Plant 2-114

Table 2.25-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Pantex Plant 2-114

Table 2.26-1. Radionuclides released to air during 1986 from Knolls Atomic Power Lab-Knolls. 2-115

Table 2.26-2. Estimated radiation dose rates from the Knolls Lab-Knolls. 2-116

Table 2.26-3. Estimated fatal cancer risks from the Knolls Lab-Knolls. 2-116

Table 2.26-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Knolls Atomic Power Lab-Knolls 2-116

Table 2.27-1. Radionuclides released to air during 1986 from Ames Laboratory 2-117

Table 2.27-2. Estimated radiation dose rates from the Ames Laboratory. 2-118

Table 2.27-3. Estimated fatal cancer risks from the Ames Laboratory 2-118

Table 2.27-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Ames Laboratory. 2-118

Table 2.28-1. Radionuclides released to air during 1986 from Rocketdyne Division, Rockwell International. 2-119

Table 2.28-2. Estimated radiation dose rates from Rocketdyne Division, Rockwell International. 2-120

Table 2.28-3. Estimated fatal cancer risks from Rocketdyne Division, Rockwell International. 2-120

Table 2.28-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Rocketdyne Division, Rockwell International 2-121

Table 3-1. Estimated emissions from model hospitals. 3-3

Table 3-2. Estimated radiation dose rates from model hospitals. 3-5

Table 3-3. Estimated fatal cancer risks from model hospitals. 3-5

Table 3-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all hospitals. 3-5

Table 3-5. Effluent release rates (Ci/y) for radiopharmaceutical manufacturers. 3-7

Table 3-6. Estimated radiation dose rates from radiopharmaceutical manufacturers. 3-8

Table 3-7. Estimated fatal cancer risks from reference radiopharmaceutical manufacturers. 3-8

Table 3-8. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all radiopharmaceutical manufacturers. 3-9

Table 3-9. Effluent release rates (Ci/y) for laboratories . 3-11

Table 3-10. Estimated radiation dose rates from laboratories 3-12

Table 3-11. Estimated fatal cancer risks from laboratories . 3-12

Table 3-12. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all laboratories 3-13

Table 3-13. Effluent release rates (Ci/y) for research reactors 3-14

Table 3-14. Estimated radiation dose rates from research reactors 3-15

Table 3-15. Estimated fatal cancer risks from research reactors 3-15

Table 3-16. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from research and test reactors. 3-16

Table 3-17. Effluent release rates (Ci/y) for sealed source manufacturers 3-17

Table 3-18. Estimated radiation dose rates from sealed source manufacturers 3-18

Table 3-19. Estimated fatal cancer risks from sealed source manufacturers 3-19

Table 3-20. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from sealed source manufacturers 3-19

Table 3-21. Effluent release rates (Ci/y) for non-LWR fuel fabricators. 3-21

Table 3-22. Estimated radiation dose rates from non-LWR fuel fabricators 3-22

Table 3-23. Estimated fatal cancer risks from non-LWR fuel fabricators. 3-22

Table 3-24. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all non-LWR fuel fabricators. 3-22

Table 3-25. Effluent release rates for source material licensees. 3-23

Table 3-26. Estimated radiation dose rates from source material licensees 3-24

Table 3-27. Estimated fatal cancer risks from source material licensees 3-24

Table 3-28. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all source material licensees 3-25

Table 3-29. Effluent release rates (Ci/y) for low-level waste disposal facilities. 3-26

Table 3-30. Estimated radiation dose rates from low-level waste disposal facilities. 3-27

Table 3-31. Estimated fatal cancer risks from low-level waste disposal facilities. 3-27

Table 3-32. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all low-level waste disposal facilities . . 3-27

Table 3-33. Effluent release rates (Ci/y) for DOD facilities 3-29

Table 3-34. Estimated radiation dose rates from DOD facilities 3-29

Table 3-35. Estimated fatal cancer risks from DOD facilities 3-30

Table 3-36. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all DOD facilities. 3-30

Table 3-37. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all NRC-licensed facilities 3-31

Table 4-1. Uranium mills licensed by the U.S. Nuclear Regulatory Commission as of December 1988. 4-4

Table 4-2. Source terms for uranium milling 4-10

Table 4-3. Areas of the tailings impoundments at uranium mills and average radium-226 concentrations. . . 4-14

Table 4-4. Sources of meteorological data used in the assessment of uranium milling. 4-15

Table 4-5. Estimated populations living within 0 to 5 km of active uranium milling facilities 4-16

Table 4-6. Estimated radiation dose rates from uranium mills. 4-17

Table 4-7. Estimated fatal cancer risks from uranium mills. 4-19

Table 4-8. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from uranium mills 4-19

Table 4-9. Effluent controls for process emissions. 4-20

Table 4-11. Reported atmospheric radioactive emissions for uranium conversion facilities (Ci/y) 4-27

Table 4-12. Atmospheric radioactive emissions assumed for reference dry and wet process uranium conversion facilities. 4-29

Table 4-13. Radiation dose equivalent rates from atmospheric radioactive emissions from reference uranium conversion facilities. 4-29

Table 4-14. Fatal cancer risks due to atmospheric radioactive emissions from reference uranium conversion facilities. 4-30

Table 4-15. Estimated distribution of lifetime fatal cancer risks projected for uranium conversion facilities 4-31

Table 4-16. Light water reactor commercial fuel fabrication facilities licensed by the Nuclear Regulatory Commission as of June 1987. 4-33

Table 4-17. Light water reactor commercial fuel fabrication facilities reported annual uranium effluent releases for 1983 through 1987 in uCi/y. 4-35

Table 4-18. Atmospheric radioactive emissions assumptions for reference fuel fabrication facility. 4-37

Table 4-19. Radiation dose equivalent rates from atmospheric radioactive emissions from model fuel fabrication facility. 4-39

Table 4-20. Fatal cancer risks due to atmospheric radioactive emissions from reference fuel fabrication facility 4-39

Table 4-21. Estimated distribution of lifetime fatal cancer risks projected for all fuel fabrication facilities 4-40

Table 4-22. U.S. nuclear power generating units operable as of December 31, 1986 (DOE87). 4-41

Table 4-23. Geometric mean and standard deviation by year for selected radionuclides for boiling water reactors in the United States for 1981 through 1985 in uCi/y. 4-49

Table 4-24. Geometric mean and standard deviation by year for selected radionuclides for pressurized water reactors in the United States for 1981 through 1985 in uCi/y. 4-51

Table 4-25. Atmospheric radioactive emissions assumed for model boiling water reactor. 4-53

Table 4-26. Atmospheric radioactive emissions assumed for model pressurized water reactor. 4-54

Table 4-27. Minimum, maximum, median, and 90th percentile population densities for nuclear power reactor sites in the United States 4-55

Table 4-28. Dose rates from model light water reactors . . . 4-56

Table 4-29. Fatal cancer risks for model light water reactors 4-56

Table 4-30. Estimated distribution of lifetime fatal cancer risks projected for all power reactors. . 4-58

Table 4-31. Doses to maximally exposed individuals in mrem/y. 4-59

Table 4-32. Summary of fatal cancer risks from atmospheric radioactive emissions from uranium fuel cycle facilities 4-67

Table 4-33. Estimated distribution of lifetime fatal cancer risks for uranium fuel cycle facilities. 4-68

Table 5-1. Projected generation of spent fuel 5-2

Table 5-2. Emissions from normal operations at HLW disposal facilities 5-5

Table 5-3. WIPP discharge stacks. 5-7

Table 5-4. Estimated radiation dose rates from high-level waste disposal facilities. 5-8

Table 5-5. Estimated fatal cancer risks from high-level waste disposal facilities. 5-8

Table 5-6. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from high-level waste disposal facilities. . . . 5-10

Table 6-1. Elemental phosphorus plants. 6-2

Table 6-2. Radionuclide stack emissions measured at elemental phosphorus plants (1975-1980). 6-4

Table 6-3. Measured radionuclide concentrations in process samples at elemental phosphorus plants - 1983-1984 results 6-6

Table 6-4. Radionuclide emissions from calciners at elemental phosphorus plants - 1983-1984 results. 6-6

Table 6-5. Measured distribution of lead-210 and polonium-210 by particle size in calciner stack outlet streams at elemental phosphorus plants - 1983 results. 6-7

Table 6-6. Dissolution of lead-210 and polonium-210 from particulate samples collected from off-gas streams at FMC and Stauffer elemental phosphorus plants. 6-7

Table 6-7. Lead-210 and polonium-210 emissions measured in calciner off-gas streams at two elemental phosphorus plants - 1988 6-9

Table 6-8. Measured distribution of lead-210 and polonium-210 by particle size in calciner stack inlet and outlet streams at elemental phosphorus plants - 1988 results 6-9

Table 6-9. Estimated annual radionuclide emissions from elemental phosphorus plants. 6-10

Table 6-10. Lung clearance classification and particle sizes used in the assessment 6-12

Table 6-11. Calciner stack emission characteristics. 6-12

Table 6-12. Populations within 80 km and distances to the maximum exposed individuals of elemental phosphorus plants with the source of meteorological data used in dose equivalent and risk calculations 6-13

Table 6-13. Estimated radiation dose equivalent rates to the maximum exposed individual and to the 80-km regional population from elemental phosphorus plants. 6-15

Table 6-14. Estimated fatal cancer risks to the maximum exposed individual and to the 80-km regional population from elemental phosphorus plants. . . 6-17

Table 6-15. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from operating elemental phosphorus plants . . . 6-18

Table 6-16. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from idle elemental phosphorus plants. 6-18

Table 6-17. Estimated Po-210 emission levels achieved by control alternatives. 6-19

Table 6-18. Estimated Pb-210 emission levels achieved by control alternatives. 6-20

Table 6-19. Capital cost of control alternatives (1,000 1988 $) 6-21

Table 6-20. Annualized cost of control alternatives (1,000 1988 $) 6-22

Table 7-1. Major decay products of uranium-238. 7-3

Table 7-2. Major decay products of thorium-232. 7-3

Table 7-3. Typical uranium and thorium concentrations in coal. 7-4

Table 7-4. Uranium concentrations and distributions in coal. 7-5

Table 7-5. Coal ash distribution by boiler type 7-7

Table 7-6. Distribution of particulate control equipment for bituminous coal-fired utility boilers. 7-9

Table 7-7. U-238 emission factors for coal-fired utility boilers. 7-11

Table 7-8. Th-232 emission factors for coal-fired utility boilers. 7-12

Table 7-9. Enrichment factors for radionuclides 7-12

Table 7-10. Emissions for typical coal-fired utility boilers. 7-13

Table 7-11. Emissions for large coal-fired utility boilers . 7-14

Table 7-12. Estimated radiation dose rates from typical coal-fired utility boilers. 7-15

Table 7-13. Estimated radiation dose rates from large coal-fired utility boilers. 7-16

Table 7-14. Estimated fatal cancer risk from typical coal-fired utility boilers. 7-17

Table 7-15. Estimated fatal cancer risk from large coal-fired utility boilers. 7-17

Table 7-16. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all coal-fired utility boilers 7-18

Table 7-17. Numbers and capacities of industrial boilers . . 7-20

Table 7-18. Estimated radiation dose rates from the reference coal-fired industrial boiler 7-23

Table 7-19. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all coal-fired industrial boilers. 7-23

Table 8-1. Quantity of tailings and planned remedial actions at inactive uranium mill tailings sites. 8-4

Table 8-2. Summary of radon-222 emissions from inactive uranium mill tailings disposal sites 8-6

Table 8-3. Estimated number of persons living within 5 km of the centroid of tailings disposal sites for inactive mills 8-7

Table 8-4. Estimated exposures and risks to individuals living near inactive tailings sites after UMTRCA disposal is completed. 8-8

Table 8-5. Estimated fatal cancers per year in the regional (0-80 km) populations around inactive tailings disposal sites 8-10

Table 8-6. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from inactive uranium mill tailings disposal sites. . 8-11

Table 8-7. Estimated exposures and risks to individuals living near inactive tailings sites assuming a 6 $pCi/m^2/s$ radon flux limit. 8-12

Table 8-8. Estimated fatal cancers per year in the regional (0-80 km) populations around inactive tailings disposal sites assuming a 6 $pCi/m^2/s$ radon flux limit 8-13

Table 8-9. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from inactive uranium mill tailings disposal sites assuming a 6 $pCi/m^2/s$ radon flux limit 8-14

Table 8-10. Estimated exposures and risks to individuals living near inactive tailings sites assuming a 2 $pCi/m^2/s$ radon flux limit. 8-15

Table 8-11. Estimated fatal cancers per year in the regional (0-80 km) populations around inactive tailings disposal sites assuming a 2 $pCi/m^2/s$ radon flux limit. 8-16

Table 8-12. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from inactive uranium mill tailings disposal sites assuming a 2 $pCi/m^2/s$ radon flux limit 8-17

Table 8-13. Estimated depths of earth cover needed to achieve given radon flux rates 8-23

Table 8-14. Major volumes and surface areas used to calculate the costs to achieve given radon-222 flux rates 8-25

Table 8-15. Estimated costs of reducing average radon-222 flux rate to 20 $pCi/m^2/s$ 8-26

Table 8-16. Estimated costs of reducing average radon-222 flux rate to 6 pCi/m^2/s. 8-27

Table 8-17. Estimated costs of reducing average radon-222 flux rate to 2 pCi/m^2/s. 8-28

Table 9-1. Operating status of licensed conventional uranium mills as of June 1989. 9-3

Table 9-2. Summary of operable tailings impoundment areas and radium-226 content at operating and standby mills. 9-7

Table 9-3. Summary of radon source terms calculated for operable mill tailings impoundments. 9-9

Table 9-4. Summary of uranium mill tailings impoundment areas, flux rates, and post-UMTRCA radon-222 release rates. 9-10

Table 9-5. Estimated number of persons living within 5 km of the centroid of tailings impoundments of licensed mills. 9-11

Table 9-6. Estimated exposures and risks to individuals living near operable tailings impoundments . . . 9-13

Table 9-7. Estimated fatal cancers per year in the regional (0-80 km) populations around operable tailings impoundments. 9-15

Table 9-8. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from operable uranium mill tailings piles. . . . 9-15

Table 9-9. Estimated exposures and risks to individuals living near licensed tailings impoundments post-UMTRCA disposal 9-17

Table 9-10. Estimated fatal cancers per year in the regional (0-80 km) populations around licensed tailings impoundments post-UMTRCA disposal 9-18

Table 9-11. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from licensed uranium mill tailings piles post-UMTRCA disposal 9-19

Table 9-12. Estimated exposures and risks to individuals living near licensed tailings impoundments post-disposal to 6 pCi/m^2/s. 9-20

Table 9-13. Estimated fatal cancers per year in the regional (0-80 km) populations around licensed tailings impoundments post-disposal to 6 $pCi/m^2/s$ 9-21

Table 9-14. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from licensed uranium mill tailings piles post-disposal to 6 $pCi/m^2/s$. 9-22

Table 9-15. Estimated exposures and risks to individuals living near licensed tailings impoundments post-disposal to 2 $pCi/m^2/s$. 9-23

Table 9-16. Estimated fatal cancers per year in the regional (0-80 km) populations around licensed tailings impoundments post-disposal to 2 $pCi/m^2/s$. 9-24

Table 9-17. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from licensed uranium mill tailings piles post-disposal to 2 $pCi/m^2/s$. 9-25

Table 9-18. Estimated depths of earth cover needed to achieve given radon flux rates 9-27

Table 9-19. Estimated costs of reducing average radon-222 flux rate to 20 $pCi/m^2/s$ 9-30

Table 9-20. Estimated costs of reducing average radon-222 flux rate to 6 $pCi/m^2/s$. 9-31

Table 9-21. Estimated costs of reducing average radon-222 flux rate to 2 $pCi/m^2/s$. 9-32

Table 9-22. Estimated total costs for new tailings control technologies 9-36

Table 9-23. Summary of estimated radon-222 emissions for new tailings control technologies. 9-36

Table 9-24. Unit cost categories for partially below-grade impoundments 9-43

Table 9-25. Costs for a single cell partially below-grade new model tailings impoundment 9-45

Table 9-26. Costs for a phased design, partially below-grade, new model tailings impoundment. . . 9-46

Table 9-27. Costs for a continuous design, partially below-new model tailings impoundment 9-47

Table 9-28. Additional areas of operable impoundments to be controlled to achieve average radon-222 flux of 20 pCi/m^2/s. 9-51

Table 10-1. Characteristics of the four raffinate pits and activity levels of major radionuclides in the currently stored materials . . . 10-4

Table 10-2. Estimated volumes of radioactive wastes stored in Weldon Spring Quarry. 10-6

Table 10-3. Volumes of contaminated soil on the MSP storage pads. 10-7

Table 10-4. Radon source strength, areas, and radon flux rates at the MUMT. 10-11

Table 10-5. Estimated exposures and risks to individuals living near DOE radon sites assuming current radon emission rates. 10-12

Table 10-6. Estimated exposures and risks to individuals living near DOE radon sites assuming post-remediation radon emission rates 10-13

Table 10-7. Estimated fatal cancers/year to the regional (0-80 km) populations around DOE radon sites for current radon emission rates. 10-14

Table 10-8. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around the FMPC for current radon emission rates 10-15

Table 10-9. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around the NFSS for current radon emission rates 10-15

Table 10-10. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around the WSCP for current radon emission rates 10-16

Table 10-11. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around the WSQ for current radon emission rates 10-16

Table 10-12. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around the MSP for current radon emission rates 10-17

Table 10-13. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around the MUMT for current radon emission rates 10-17

Table 10-14. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around the FMPC for post-remediation radon emission rates. 10-18

Table 10-15. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around the MSP for post-remediation radon emission rates. 10-18

Table 10-16. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around the MUMT for post-remediation radon emission rates. 10-19

Table 10-17. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around all DOE radon sites for current radon emission rates. 10-19

Table 10-18. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population around all DOE radon sites for post-remediation radon emission rates. 10-20

Table 10-19. Summary of capital costs to reduce radon emissions from DOE radon sites. 10-22

Table 11-1. Currently operating underground uranium mines in the United States. 11-2

Table 11-2. Estimated annual radon-222 emissions from underground uranium mining sources (EPA83b) . . 11-6

Table 11-3. Radon-222 concentrations and annual release rates in mine ventilation exhaust air 11-7

Table 11-4. Estimated exposures and risks to individuals living near underground uranium mines 11-11

Table 11-5. Estimate committed fatal cancers per year due to radon-222 emissions from underground uranium mines 11-13

Table 11-6. Estimated distribution of the fatal cancer risk caused by radon-222 emissions from all underground uranium mines 11-14

Table 11-7. Current mine ventilation exhaust vents. 11-19

Table 11-8. Estimated lifetime fatal cancer risk to the maximum exposed individual and the committed fatal cancers per year due to radon-222 emissions from underground uranium mines as a function of vent stack height 11-21

Table 11-9. Effectiveness of various stack heights. 11-24

Table 11-10. Estimated costs (dollars) to extend the heights of the ventilation exhaust stacks at each underground uranium mine. 11-26

Table 11-A-1. Weights of stack liner per vertical foot. . 11-A-3

Table 11-A-2. Weights of structural steel used. 11-A-3

Table 11-A-3. Exhaust stack costs (dollars) for individual stacks 11-A-4

Table 11-A-4. Number and size of exhaust shafts assumed for cost estimate 11-A-5

Table 12-1. Uranium ore production from surface mines, 1948-1986 12-2

Table 12-2. Breakdown by state of surface uranium mines with > 1,000 tons production. 12-3

Table 12-3. Federal laws, regulations, and guidelines for uranium mining. 12-5

Table 12-4. Estimated additional uranium resources by land status 12-6

Table 12-5. Estimated status[a] of surface uranium mine reclamation. 12-11

Table 12-6. Mines characterized in the field studies. . . .12-13

Table 12-7. Estimated radon-222 emissions from surface uranium mines 12-14

Table 12-8. Estimated particulate emissions from surface uranium mines 12-15

Table 12-9. Estimated exposures and risks to individuals living near surface uranium mines 12-17

Table 12-10. Estimated fatal cancers per year in the regional (0-80 km) populations due to radon-222 emissions from surface uranium mines 12-18

Table 12-11. Estimated distribution of the fatal cancer risk caused by radon-222 emissions from all surface uranium mines 12-19

Table 12-12. Estimated lifetime fatal cancer risks from particulate emissions 12-19

Table 12-13. Estimated depths of cover to reduce radon-222 emissions at surface uranium mines. 12-20

Table 12-14. Estimated costs to reduce radon emissions at surface uranium mines 12-20

Table 13-1. The location and characteristics of phosphogypsum stacks in the United States . . . 13-3

Table 13-2. Summary of the phosphogypsum stacks in each state. 13-4

Table 13-3. Average radionuclide concentrations in phosphogypsum, pCi/g dry weight 13-5

Table 13-4. Results of radon-222 flux measurements on phosphogypsum stacks in Florida. 13-10

Table 13-5. Radon-222 flux values applied to various regions of phosphogypsum stacks 13-11

Table 13-6. Results of radon-222 flux measurements on phosphogypsum stacks in Idaho 13-13

Table 13-7. Estimates of annual radon-222 emissions from phosphogypsum stacks 13-16

Table 13-8. Annual radionuclide emissions in fugitive dust from a model 31-ha phosphogypsum stack . .13-17

Table 13-9. Average net airborne radionuclide concentrations measured at the W.R. Grace stack 13-18

Table 13-10. The ten highest individual lifetime risks estimated to result from radon-222 emissions from phosphogypsum stacks 13-22

Table 13-11. Estimated increased risk of fatal cancer and the dose equivalent rates from maximum exposure to fugitive dusts for an individual living near phosphogypsum stacks. 13-24

Table 13-12. The 10 regional populations estimated to receive the highest collective risks from radon-222 emissions from phosphogypsum stacks. 13-24

Table 13-13. Estimated distribution of the fatal cancer risk caused by radon-222 emissions from phosphogypsum stacks. 13-25

Table 13-14. A summary of the committed fatal cancers due to radon-222 emissions from phosphogypsum stacks located in five regions in the United States 13-26

Table 13-15. Estimated number of fatal cancers from fugitive dust emissions for the population living within 80 km of the model phosphogypsum stacks. 13-27

Table 13-16. Characteristics of gypsum stacks. 13-30

Table 13-17. Mean characteristics of the stacks in each group. 13-35

Table 13-18. Radon emissions from grouped gypsum stacks. . .13-36

Table 13-19. Cost of mitigation. 13-37

Table 13-20. Risk of cancer death. 13-38

Table 13-B-1. Estimated dimensions and areas of phosphogypsum stacks. 13-B-2

Table 13-C-1. Estimated lifetime fatal cancer risks to nearby individuals caused by radon-222 emissions from phosphogypsum stacks 13-C-3

Table 13-C-2. Summary of committed fatal cancers per year within 80 km of phosphogypsum stacks. . . . 13-C-6

Table 13-D-1. Estimated distribution of lifetime fatal cancer risk caused by radon-222 emissions from seven phosphogypsum stacks in Texas. . 13-D-2

Table 13-D-2. Estimated distribution of lifetime fatal cancer risk caused by radon-222 emissions from 10 phosphogypsum stacks in the Bartow, FL, region. 13-D-2

Table 13-D-3. Estimated distribution of lifetime fatal cancer risk caused by radon-222 emissions from six phosphogypsum stacks in Illinois . 13-D-3

Table 13-D-4. Estimated distribution of lifetime fatal cancer risk caused by radon-222 emissions from seven phosphogypsum stacks in Louisiana 13-D-3

Table 13-D-5. Estimated distribution of lifetime fatal cancer risk caused by radon-222 emissions from three phosphogypsum stacks in Idaho. . 13-D-4

Table 13-E-1. Values used to scale risk 13-E-4

Table 13-E-2. Cost breakdown. 13-E-6

LIST OF FIGURES

VOLUME II: RISK ASSESSMENT

Figure 9-1. Shape and layout of the model single-cell impoundment 9-38

Figure 9-2. Size of partially above-grade model single cell impoundment. 9-39

Figure 9-3. Size of below-grade and partially above-grade cell of model phased impoundment. 9-41

Figure 9-4. Shape and layout of model phased disposal impoundment 9-42

Figure 11-1. General framing plan of a mine ventilation exhaust stack 11-25

Figure 13-1. Effect of release height on individual risk for a model stack 13-20

LIST OF FIGURES

VOLUME II: RISK ASSESSMENT

Figure 9-1. Shape and layout of the model single-cell impoundment 9-38

Figure 9-2. Size of partially above-grade model single cell impoundment. 9-39

Figure 9-3. Size of below-grade and partially above-grade cell of model phased impoundment. 9-41

Figure 9-4. Shape and layout of model phased disposal impoundment 9-42

Figure 11-1. General framing plan of a mine ventilation exhaust stack 11-25

Figure 13-1. Effect of release height on individual risk for a model stack 13-20

1. INTRODUCTION

The purpose of this report is to serve as a background information document in support of the Environmental Protection Agency's (EPA's) final rules for sources of airborne emissions of radionuclides pursuant to Section 112 of the Clean Air Act.

This report presents an analysis of the exposures and risks caused by radionuclides emitted into the air from 12 source categories. The analysis draws upon and updates previous evaluations and incorporates revisions to the estimates based on new information developed during the public comment period for the proposed rules. Specific changes from the analyses presented in the draft report are noted in the appropriate sections of the text and on the AIRDOS/DARTAB/ RADRISK input sheets in Appendix A. The report presents the Agency's most current assessment of the risks and impacts caused by these facilities. The evaluation covers the following source categories:

1. Department of Energy (DOE) Facilities;
2. Nuclear Regulatory Commission (NRC) Licensed and non-DOE Federal Facilities;
3. Uranium Fuel Cycle Facilities;
4. High-Level Waste Disposal Facilities;
5. Elemental Phosphorus Plants;
6. Coal-Fired Boilers;
7. Inactive Uranium Mill Tailings;
8. Licensed Uranium Mill Tailings;
9. DOE Radon Sites;
10. Underground Uranium Mines;
11. Surface Uranium Mines; and
12. Phosphogypsum Stacks.

For each source category, the EPA is presenting the following information:

1. A general description of the source category, including a brief description of the processes that lead to the emission of radionuclides to air and a characterization of the emission controls that are currently in use to limit such emissions;

2. The basis for the exposure and risk assessment, including radionuclide emissions data, characteristics of the release point(s), and the sources for the demographic and meteorological data that were used;

3. The results of the risk assessment, including estimates of the exposure and lifetime fatal cancer risk to nearby individuals, the exposure and number of committed deaths/year in the regional (0-80 km) populations, and the distribution of the fatal cancer risk in the regional populations; and

4. An evaluation of supplementary control options and costs for source categories or segments of source categories with the highest estimated risks and impacts.

In making the risk assessments every effort has been made to assess facilities on a site-specific basis, using measured data for emissions and actual data on the configuration of the release point(s) and the locations of nearby individuals. For source categories where measured emissions data are not available, emissions have been estimated using the bases and the assumptions given for that source category. Where locations of nearby individuals are not known, the assessment is made to the point of maximum offsite concentrations. The intent of each assessment is to provide a realistic estimate of the exposures and risks that could be received by individuals.

For certain source categories, the number of facilities makes such site-specific evaluations impractical. In these instances, for example nuclear power reactors, reference (actual) facilities are used or model facilities are defined and evaluated. When a reference or model facility is used, the exposure and risk estimates presented are for hypothetical individuals and populations selected as representative of the demography around actual facilities.

The exposures presented represent 50-year committed dose equivalents. Estimated doses are presented for organs where the dose represents 10 percent or more of the fatal cancer risk. For radon exposures, both the radon concentration (pCi/l) and the working levels (WL) are reported. The working levels include the contribution from radon decay products, calculated as a function of distance (see Volume I).

The fatal cancer risks for nearby or maximum individuals are lifetime risks. They represents the probability of a typical individual dying from a lifetime (70 year) exposure to the concentration of radionuclides estimated at that environmental location. Chapter 7 of Volume I discusses the uncertainties that are associated with this assumption. The number of committed fatal cancers per year (deaths/year) of operation is the

estimated number of cancers that will occur in the exposed population from one year's release of radionuclides. Due to the latency period for cancers, these deaths will occur in the future, not in the year that the release takes place.

As discussed in Chapter 7 of Volume I, modeling uncertainties, completeness uncertainties, and parameter uncertainties are associated with each of the exposure and risk estimate. However, throughout this volume, exposure and risk estimates are presented as discrete values. The reader is referred to Chapter 7 of Volume I and the "Analysis of the Uncertainties in the Risk Assessment Performed in Support of the Proposed NESHAPS for Radionuclides" (EPA89) for information on the range and distribution of the parameter uncertainties associated with the estimates.

REFERENCES

EPA89 U.S. Environmental Protection Agency, "Analysis of the Uncertainties in the Risk Assessment Performed in Support of the Proposed NESHAPS for Radionuclides," Washington, DC, September 1989.

2. DEPARTMENT OF ENERGY (DOE) FACILITIES

2.1 OVERVIEW AND SUMMARY OF RESULTS

2.1.1 General Description of DOE Facilities

The DOE facilities source category comprises sites that are owned by the Federal government and operated by contractors under the supervision of the DOE. The sites addressed in this chapter are the active DOE sites that release significant quantities of radionuclides to the air. These facilities and their locations are listed in Table 2.1-1. These facilities are engaged in numerous aspects of nuclear energy. They support the nation's nuclear weapons capability by designing and producing nuclear weapons for the Department of Defense (DOD). They support the commercial nuclear power sector through enrichment of uranium and nuclear reactor development and safety programs. They are also involved in biomedical research, environmental safety, and nuclear waste disposal programs.

The diversity of operations at these sites makes it difficult to assess DOE facilities on a generic basis. The major emissions from the facilities, however, are similar and consist largely of inert gases such as argon-41, krypton-85, krypton-88, and xenon-133. These gases are heavier than air and only slightly soluble in water. Tritium, oxygen-15, uranium-234, and uranium-238 are also commonly emitted.

A site-by-site discussion of each facility is presented in the following sections along with an estimate of the doses and risks associated with the current (1986) releases of radionuclides to the atmosphere. Details of the inputs supplied to the AIRDOS-EPA/DARTAB/RADRISK risk assessment computer codes are presented for each site in Appendix A.

Historically, the Department of Energy has been self-regulating with respect to environmental controls. Since the 1970's, limits on releases of radioactive materials have roughly paralleled those established by the Nuclear Regulatory Commission (NRC). In 1985, the EPA promulgated a NESHAP for DOE facilities (40 CFR 61, Subpart H) which limits radionuclide releases to air from any DOE facility to quantities that do not cause nearby individuals a dose greater than 25 mrem/y to the whole body or 75 mrem/y to any organ.

The summary tables in Section 2.1 and the individual facility discussions incorporate source terms, stack heights, meteorology, and other model parameters that reflect comments received from DOE and the specific facilities. Model input parameters are described in the AIRDOS input sheets presented in the appendix. Draft version input sheets may be compared to these sheets to determine changes in AIRDOS input parameters.

Table 2.1-1. Department of Energy facilities.

Facility	Location
Los Alamos National Laboratory	Los Alamos, New Mexico
Oak Ridge Reservation	Oak Ridge, Tennessee
Savannah River Plant	Aiken, South Carolina
Reactive Metals, Inc.	Ashtabula, Ohio
Feed Materials Production Center	Fernald, Ohio
Hanford Reservation	Richland, Washington
Brookhaven National Laboratory	Long Island, New York
Mound Facility	Miamisburg, Ohio
Idaho National Engineering Laboratory	Upper Snake River,Idaho
Lawrence-Berkeley Laboratory	Berkeley, California
Paducah Gaseous Diffusion Plant	Paducah, Kentucky
Lawrence Livermore/Sandia Laboratory	Livermore, California
Portsmouth Gaseous Diffusion Plant	Piketon, Ohio
Argonne National Laboratory	Argonne, Illinois
Pinellas Plant	Pinellas County, Florida
Nevada Test Site	Nye County, Nevada
Knolls Atomic Power Laboratory	Kesselring, New York
Battelle Memorial Institute	Columbus, Ohio
Fermi National Accelerator Laboratory	Batavia, Illinois
Sandia National Laboratories/Lovelace	Albuquerque, New Mexico
Bettis Atomic Power Laboratory	West Mifflin, Pennsylvania
Knolls Atomic Power Laboratory	Windsor, Connecticut
Rocky Flats Plant	Jefferson Co., Colorado
Pantex Plant	Amarillo, Texas
Knolls Atomic Power Laboratory	Schenectady, New York
Ames Laboratory	Ames, Iowa
Rockwell International	Santa Susana, California

2.1.2 Summary of the Dose and Risk Assessment

The following tables present the tabulated results of the risk assessment for 27 facilities in this source category. Table 2.1-2 shows the risk figures representing the highest cancer risk to a selected individual. Table 2.1-3 presents the aggregate risk distribution table for all DOE facilities. Table 2.1-4 presents the population exposures and total deaths per year for all DOE facilities.

Results for each site are also tabulated and presented in the following sections.

Table 2.1-2 Summary of doses and risks to nearby individuals from DOE facilities due to 1986 emmissions.

Site	Primary Radio-nuclide	1986 Emissions (Ci/y)	Organ Doses (mrem/y)		Maximum Individual Risk
Los Alamos Laboratory, NM	O-15	8.6E+4	Gonads	9.5E+0	2E-4
	C-11	1.8E+4	Remainder	7.4E+0	
	N-13	4.8E+3	Breast	8.9E+0	
			Lungs	8.8E+0	
			Red marrow	7.0E+0	
Oak Ridge National Lab., TN	U-234	1.5E-1	Lungs	2.2E+1	8E-5
	H-3	3.1E+4	Remainder	2.0E+0	
	U-238	2.8E-2			
Savannah River Plant, GA	H-3	4.2E+5	Remainder	3.2E+0	7E-5
	Ar-41	8.3E+4	Gonads	2.6E+0	
			Breast	2.6E+0	
			Lungs	2.7E+0	
			Red marrow	2.6E+0	
Reactive Metals, Inc., OH	U-234	5.6E-4	Lungs	2.5E+1	4E-5
	U-238	5.3E-3			
Feed Materials Prod. Ctr., OH	U-234	2.0E-2	Lungs	1.9E+1	3E-5
	U-238	2.0E-2			
Hanford Reservation, WA	Ar-41	1.3E+5	Lungs	2.8E+0	3E-5
	Pu-238	8.9E-2	Remainder	1.0E+0	
	Pu-239	3.1E-3	Gonads	1.1E+0	
			Endosteum	6.3E+0	
Brookhaven National Lab., NY	Ar-41	1.2E+3	Gonads	8.0E-1	2E-5
			Remainder	6.2E-1	
			Breast	7.2E-1	
			Red marrow	6.2E-1	
			Lungs	6.1E-1	

Table 2.1-2 Summary of doses and risks to nearby individuals from DOE facilities (continued).

Site	Primary Radio-nuclide	1986 Emissions (Ci/y)	Organ	Doses (mrem/y)	Maximum Individual Risk
Mound Facility, OH	H-3	3.6E+3	Remainder	3.5E-2	1E-6
			Gonads	3.3E-2	
			Breast	3.3E-2	
			Lungs	3.3E-2	
			Red marrow	3.3E-2	
Idaho National Eng. Lab., ID	Ar-41	1.8E+3	Gonads	2.9E-2	6E-7
	Sb-125	9.3E-1	Remainder	2.3E-2	
	Kr-88	1.4E+2	Breast	2.7E-2	
			Lungs	2.4E-2	
			Red marrow	2.3E-2	
Lawrence Berkeley Lab., CA	H-3	7.6E+1	Remainder	1.9E-2	5E-7
			Gonads	1.8E-2	
			Red marrow	2.5E-2	
			Breast	1.8E-2	
			Lungs	1.8E-2	
Paducah Gaseous Diff. Plant, KY	U-234	1.8E-4	Lungs	2.5E-1	4E-7
	U-238	1.8E-4			
Lawrence Livermore Lab., CA	H-3	2.0E+3	Remainder	1.1E-2	3E-7
			Gonads	1.1E-2	
			Breast	1.1E-2	
			Lungs	1.1E-2	
			Red marrow	1.1E-2	
Portsmouth Gaseous Diff. Plant, OH	U-234	2.8E-2	Endosteum	3.4E-1	2E-7
	U-238	1.0E-2	Remainder	3.0E-2	
			Red marrow	2.3E-2	
Argonne National Lab., IL	C-11	9.0E+1	Lungs	3.1E-2	1E-7
	H-3	5.0E+1	Remainder	2.7E-3	

Table 2.1-2 Summary of doses and risks to nearby individuals from DOE facilities (continued).

Site	Primary Radio-nuclide	1986 Emissions (Ci/y)	Organ Doses (mrem/y)		Maximum Individual Risk
Pinellas Plant, FL	H-3	1.9E+2	Remainder	4.7E-3	1E-7
			Gonads	4.4E-3	
			Breast	4.4E-3	
			Lungs	4.4E-3	
			Red marrow	4.3E-3	
Nevada Test Site, NV	Xe-133	3.6E+4	Gonads	5.3E-3	1E-7
	H-3	1.2E+2	Remainder	3.5E-3	
			Breast	6.5E-3	
			Thyroid	1.9E-2	
Knolls Lab-Kesselring, NY	Ar-41	1.6E-1	Remainder	3.8E-3	1E-7
	CO-60	3.4E-6	Red marrow	6.9E-3	
	C-14	3.4E-1	Breast	4.4E-3	
			Gonads	2.5E-3	
Battelle Memorial Inst., OH	K-40	3.0E-4	Lungs	3.1E-3	2E-8
	U-235	2.6E-6	Gonads	8.7E-4	
	Pu-239	4.0E-7	Remainder	7.2E-4	
			Breast	7.8E-4	
Fermi National Lab., IL	C-11	3.4E+0	Gonads	9.2E-4	2E-8
			Remainder	7.1E-4	
			Breast	8.6E-4	
			Lungs	9.1E-4	
			Red marrow	7.0E-4	
Sandia National Lab.-Lovelace, NM	Ar-41	5.5E+0	Remainder	5.3E-4	1E-8
	Pb-212	8.5E-3	Gonads	5.9E-4	
			Lungs	1.2E-3	
			Breast	5.4E-4	
			Red marrow	5.6E-4	
Rocky Flats Plant, CO	U-238	1.7E-5	Lungs	6.3E-3	1E-8
	Am-241	4.8E-6	Endosteum	1.6E-2	
			Remainder	7.5E-4	

Table 2.1-2 Summary of doses and risks to nearby individuals from DOE facilities (continued).

Site	Primary Radio-nuclide	1986 Emissions (Ci/y)	Organ	Doses (mrem/y)	Maximum Individual Risk
Bettis Atomic Power Lab., PA	U-234	6.0E-7	Lungs	4.3E-3	1E-8
	U-238	6.0E-7			
	Sb-125	3.2E-5			
Knolls Lab-Windsor, CT	Ar-41	7.8E-2	Gonads	3.8E-4	8E-9
			Remainder	3.0E-4	
			Breast	3.5E-4	
			Red marrow	3.0E-4	
			Lungs	2.9E-4	
Pantex Plant, TX	U-238	1.0E-5	Lungs	2.2E-3	4E-9
Knolls Lab-Knolls, CT	U-234	3.3E-6	Lungs	1.7E-3	3E-9
Ames Laboratory, IA	H-3	7.6E-2	Remainder	1.6E-5	4E-10
			Gonads	1.3E-5	
			Breast	1.3E-5	
			Red marrow	1.3E-5	
			Lungs	1.3E-5	
Rocketdyne Rockwell, CA	Sr-90	1.3E-5	Red marrow	7.0E-6	2E-11

Table 2.1-3. Distribution of fatal cancer risk in the population.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	2*	5E-6
1E-5 to 1E-4	590,000	2E-1
1E-6 to 1E-5	1,000,000	3E-2
< 1E-6	65,000,000	1E-2
TOTALS	67,000,000	3E-1

* EPA believes there are people at this risk at two facilities (RMI, LASL). However, we cannot quantify the number because a site visit has not been made.

Table 2.1-4 Summary of doses and risks to the regional population (0-80 km) around DOE facilities.

Site	0-80 km Population	Population Organ Exposure (person-rem/y)		Deaths/y
Los Alamos Laboratory, NM	160,000	Gonads	1.0E+1	4E-3
		Remainder	1.1E+1	
		Breast	9.7E+0	
		Lungs	1.1E+1	
		Red marrow	9.2E+0	
Oak Ridge National Lab., TN	160,000	Lungs	4.3E+2	3E-2
		Remainder	2.0E+0	
Savannah River Plant, GA	550,000	Remainder	6.7E+2	2E-1
		Gonads	5.5E+2	
		Breast	5.5E+2	
		Lungs	5.6E+2	
		Red marrow	5.5E+2	
Reactive Metals, Inc., OH	1,400,000	Lungs	3.4E+1	8E-4
Feed Materials Prod. Ctr., OH	3,300,000	Lungs	1.1E+2	3E-3
Hanford Reservation, WA	350,000	Lungs	5.6E+1	6E-3
		Remainder	1.7E+2	
		Gonads	1.5E+1	
		Endosteum	1.7E+2	
Brookhaven National Lab., NY	5,200,000	Gonads	3.8E+0	1E-3
		Remainder	3.0E+0	
		Breast	3.4E+0	
		Red marrow	2.9E+0	
		Lungs	2.9E+0	

Table 2.1-4 Summary of doses and risks to the regional population (0-80 km) around DOE facilities (continued).

Site	0-80 km Population	Population Organ Exposure (person-rem/y)		Deaths/y
Mound Facility, OH	2,900,000	Remainder	3.3E+0	3E-3
		Gonads	3.0E+0	
		Breast	3.0E+0	
		Lungs	3.0E+0	
		Red marrow	3.0E+0	
Idaho National Eng. Lab., ID	100,000	Gonads	7.3E-2	2E-5
		Remainder	6.3E-2	
		Breast	6.8E-2	
		Lungs	6.1E-2	
		Red marrow	5.7E-2	
Lawrence Berkeley Lab., CA	5,000,000	Remainder	7.8E-1	3E-4
		Gonads	7.0E-1	
		Red marrow	1.0E+0	
		Breast	7.0E-1	
		Lungs	7.0E-1	
Paducah Gaseous Diff. Plant, KY	500,000	Lungs	3.1E-1	1E-5
Lawrence Livermore Lab., CA	5,300,000	Remainder	4.2E+0	1E-3
		Gonads	3.7E+0	
		Breast	3.7E+0	
		Lungs	3.8E+0	
		Red marrow	3.7E+0	
Portsmouth Gaseous Diff. Plant, OH	620,000	Endosteum	5.7E+0	9E-5
		Remainder	7.7E-1	
		Red marrow	4.0E-1	
Argonne National Lab., IL	7,900,000	Lungs	2.5E-1	8E-5
		Remainder	2.1E-1	

Table 2.1-4 Summary of doses and risks to the regional population (0-80 km) around DOE facilities (continued).

Site	0-80 km Population	Population Organ Exposure (person-rem/y)		Deaths/y
Pinellas Plant, FL	1,900,000	Remainder	5.3E-1	2E-4
		Gonads	4.7E-1	
		Breast	4.7E-1	
		Lungs	4.7E-1	
		Red marrow	4.7E-1	
Nevada Test Site, NV	3,500	Gonads	1.2E-2	3E-6
		Remainder	8.1E-3	
		Breast	1.5E-2	
		Thyroid	5.7E-2	
Knolls Lab-Kesselring, NY	1,200,000	Remainder	3.2E-2	2E-5
		Red marrow	6.5E-2	
		Breast	3.7E-2	
		Gonads	1.5E-2	
Battelle Memorial Inst., OH	1,900,000	Lungs	1.5E-2	3E-6
		Gonads	6.2E-3	
		Remainder	5.2E-3	
		Breast	5.7E-3	
Fermi National Lab., IL	7,700,000	Gonads	4.1E-3	1E-6
		Remainder	3.2E-3	
		Breast	3.9E-3	
		Lungs	4.1E-3	
		Red marrow	3.2E-3	
Sandia National Lab.-Lovelace, NM	500,000	Remainder	1.9E-2	8E-6
		Gonads	2.1E-2	
		Lungs	4.9E-2	
		Breast	1.9E-2	
		Red marrow	2.1E-2	
Rocky Flats Plant, CO	1,900,000	Lungs	1.2E-1	9E-6
		Endosteum	2.0E-1	
		Remainder	9.3E-3	

Table 2.1-4 Summary of doses and risks to the regional population (0-80 km) around DOE facilities (continued).

Site	0-80 km Population	Population Organ Exposure (person-rem/y)		Deaths/y
Bettis Atomic Power Lab., PA	3,100,000	Lungs	3.5E-2	1E-6
Knolls Lab-Windsor, CT	3,200,000	Gonads	2.3E-3	2E-6
		Remainder	4.2E-3	
		Breast	4.9E-3	
		Red marrow	8.1E-3	
		Lungs	2.5E-3	
Pantex Plant, TX	260,000	Lungs	3.5E-3	7E-8
Knolls Lab-Knolls, CT	1,200,000	Lungs	3.1E-2	1E-6
Ames Laboratory, IA	680,000	Remainder	2.3E-4	9E-8
		Gonads	1.8E-4	
		Breast	1.8E-4	
		Red marrow	1.8E-4	
		Lungs	1.8E-4	
Rocketdyne Rockwell, CA	8,800,000	Red marrow	1.4E-3	7E-8

2.1.3 Summary of the Supplementary Control Alternatives

The facilities chosen for discussion of supplemental control alternatives are those that yielded an effective dose equivalent of 1 mrem/yr or higher. These facilities are:

1. Oak Ridge Reservation
2. Los Alamos Scientific Laboratory
3. Savannah River Plant
4. FMPC

Current emission control technologies and detailed discussions of supplemental control technologies at each of these facilities are presented in Sections 2.2 through 2.7.

Alternative 1: baseline emissions

MIR: 8.1×10^{-5}
Incidence: 0.22
Impact: None

Alternative 2: emissions limited to 10 mrem/y EDE.

MIR: 8.1E-5
Incidence: 0.22
Impact, alternative 1 to alternative 2:
Incremental Capital Cost: $0
Incremental Annual Operating Cost: $0
Incremental Incidence Reduction: None

All DOE facilities have baseline emissions corresponding to an EDE of 10 mrem/y or less. Therefore, Alternative 2 is identical to Alternative 1.

Alternative 3: emissions limited to 3 mrem/y EDE.

MIR: 7.5E-5
Incidence: 0.03
Impact, alternative 2 to alternative 3:
Incremental Capital Cost: $5.9 million
Incremental Annual Operating Cost: $182,000
Incremental Incidence Reduction: 0.04

To reach this limit, supplemental emission controls would be required at two DOE facilities: Oak Ridge National Laboratory and Los Alamos National Laboratory.

At Oak Ridge, an additional stage HEPA filter and high-energy Venturi scrubber, at an estimated capital cost of $2,650,000, would reduce emissions of uranium-234 and uranium-238 from the Y-12 plant. In addition, a tritiated water sieve/dryer system, at an estimated capital cost of $1,660,000, would reduce emissions of tritium from ORNL. These emission reductions would be sufficient to allow ORNL to reach the Alternative B limit.

At Los Alamos, beam stop modifications and a delay tunnel and new venting stack at the Meson Physics Facility would sufficiently reduce emissions of oxygen-15, carbon-11, and nitrogen-13, at a capital cost of $1,600,000.

Alternative 4: emissions limited to 1.0 mrem/y EDE.

MIR: 2.4E-5

Incidence: 0.094

Impact, alternative 3 to alternative 4:

Incremental Capital Cost: $134 million
Incremental Annual Operating Cost: $8,111,000
Incremental Incidence Reduction: 0.036

To reach Alternative 4, additional emission controls would be required at RMI, Savannah River and FMPC.

For Savannah River, additional stage HEPA filters would be required on the F and H stacks and in the P, X, and C reactor areas, at an estimated capital cost of $130 million.

For FMPC, HEPA filters for Plants 4, 5, and 8 and additional dust collector and scrubber stacks, at an estimated capital cost of $4.2 million would be required.

2.1.4 Effect of Supplementary Control Alternatives

Tables 2.1-5 through 2.1-7 present the risk distributions for the population at risk for the DOE facilities. Table 2.1-5 presents the risk distribution for the baseline case, which assumes 1986 emissions with no supplemental control strategies implemented. Table 2.1-6 presents the risk distribution for Alternative 3, which assumes that supplemental controls have been applied to ensure that an effective dose equivalent to nearby individuals would be no more than 3 mrem/y at any of the DOE facilities. Table 2.1-7 presents the risk distribution for Alternative 4, which assumes that supplemental controls have been applied to ensure that an effective dose equivalent to nearby individuals would be no more than 1 mrem/y at any of the DOE facilities.

The maximum individual risks, assuming implementation of Alternative 4 supplemental control strategies, are presented in Table 2.1-8.

The number of deaths per year, assuming implementation of Alternative 4 supplemental control strategies, are presented in Table 2.1-9.

Table 2.1-5. Baseline risk assessment for DOE facilities.

Highest Lifetime Individual Fatal Cancer Risk: 1E-04

Population Risk (those within 80 km): 0.2

Distribution of Fatal Cancer Risk in Populations Within 80 km:

Risk interval	Number of persons	Deaths/y
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	2	5E-6
1E-5 to 1E-4	590,000	2E-1
1E-6 to 1E-5	1,000,000	3E-2
<1E-6	65,000,000	1E-2
Total	67,000,000	3E-1

Table 2.1-6. Risks when emmissions are limited to 3 mrem/y EDE.

Highest Lifetime Individual Fatal Cancer Risk: 4E-05

Population Risk (those within 80 km): 0.1

Distribution of Fatal Cancer Risk in Populations Within 80 km:

Risk interval	Number of persons	Deaths/y
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	1	2E-6
1E-5 to 1E-4	560,000	2E-1
1E-6 to 1E-5	250,000	7E-3
<1E-6	66,000,000	2E-2
Total	67,000,000	2E-1

Table 2.1-7. Risks when emmissions are limited to 1 mrem/y EDE.

Highest Lifetime Individual Fatal Cancer Risk: 2E-05

Population Risk (those within 80 km): 0.09

Distribution of Fatal Cancer Risk in Populations Within 80 km:

Risk interval	Number of persons	Deaths/y
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	250,000	4E-2
1E-6 to 1E-5	540,000	4E-2
<1E-6	66,000,000	1E-2
Total	67,000,000	1E-1

Table 2.1-8. Maximum individual risk, with Alternative 4 supplemental control strategies.

Site	Primary Radio-nuclide	1986 Emissions (Ci/y)*	Maximum Individual Risk**
Hanford Reservation, WA	Ar-41	1.3E+5	3E-5
	Pu-238	8.9E-2	
	Pu-239	3.1E-3	
Savannah River Plant, GA	H-3	4.2E+5	2E-5
	Ar-41	8.3E+4	
Oak Ridge National Lab., TN	U-234	1.5E-1	2E-5
	H-3	3.1E+4	
	U-238	2.8E-2	
Brookhaven National Lab., NY	Ar-41	1.2E+3	2E-5
Los Alamos Laboratory, NM	O-15	8.6E+4	2E-5
	C-11	1.8E+4	
	N-13	4.8E+3	
Reactive Metals, Inc., OH	U-234	5.6E-4	1E-5
	U-238	5.3E-3	
Feed Materials Prod. Ctr., OH	U-234	2.0E-2	1E-5
	U-238	2.0E-2	
Mound Facility, OH	H-3	3.6E+3	1E-6
Idaho National Eng. Lab., ID	Ar-41	1.8E+3	6E-7
	Sb-125	9.3E-1	
	Kr-88	1.4E+2	
Lawrence Berkeley Lab., CA	H-3	7.6E+1	5E-7

* With supplemental emission controls.
** Nearby generic individual from population run.

Table 2.1-8. Maximum individual risk, with Alternative 4 supplemental control strategies (continued).

Site	Primary Radio-nuclide	1986 Emissions (Ci/y)*	Maximum Individual Risk**
Paducah Gaseous Diff. Plant, KY	U-234 U-238	1.8E-4 1.8E-4	4E-7
Lawrence Livermore Lab., CA	H-3	2.0E+3	3E-7
Portsmouth Gaseous Diff. Plant, OH	U-234 U-238	2.8E-2 1.0E-2	2E-7
Argonne National Lab., IL	C-11 H-3	9.0E+1 5.0E+1	1E-7
Pinellas Plant, FL	H-3	1.9E+2	1E-7
Nevada Test Site, NV	Xe-133 H-3	3.6E+4 1.2E+2	1E-7
Knolls Lab-Kesselring, NY	Ar-41 CO-60 C-14	1.6E-1 3.4E-6 3.4E-1	1E-7
Battelle Memorial Inst., OH	K-40 U-235 Pu-239	3.0E-4 2.6E-6 4.0E-7	2E-8
Fermi National Lab., IL	C-11	3.4E+0	2E-8
Sandia National Lab.-Lovelace, NM	Ar-41 Pb-212	5.5E+0 8.5E-3	1E-8
Rocky Flats Plant, CO	U-238 Am-241	1.7E-5 4.8E-6	1E-8

* With supplemental emission controls.
** Nearby generic individual from population run.

Table 2.1-8. Maximum individual risk, with Alternative 4 supplemental control strategies (continued).

Site	Primary Radio-nuclide	1986 Emissions (Ci/y)*	Maximum Individual Risk**
Bettis Atomic Power Lab., PA	U-234 U-238 Sb-125	6.0E-7 6.0E-7 3.2E-5	1E-8
Knolls Lab-Windsor, CT	Ar-41	7.8E-2	8E-9
Pantex Plant, TX	U-238	1.0E-5	4E-9
Knolls Lab-Knolls, CT	U-234	3.3E-6	3E-9
Ames Laboratory, IA	H-3	7.6E-2	4E-10
Rocketdyne Rockwell, CA	Sr-90	1.3E-5	2E-11

* With supplemental emission controls.
** Nearby generic individual from population run.

Table 2.1-9. Fatal cancers/year to nearby individuals, with Alternative 4 supplemental control technologies.

Site	0-80 km Population	Deaths/y*
Los Alamos Laboratory, NM	160,000	2E-3
Oak Ridge National Lab., TN	550,000	7E-3
Savannah River Plant, GA	550,000	8E-2
Reactive Metals, Inc., OH	1,400,000	7E-5
Feed Materials Prod. Ctr., OH	3,300,000	9E-4
Hanford Reservation, WA	350,000	6E-3
Brookhaven National Lab., NY	5,200,000	1E-3
Mound Facility, OH	2,900,000	3E-3
Idaho National Eng. Lab., ID	100,000	2E-5
Lawrence Berkeley Lab., CA	5,000,000	3E-4
Paducah Gaseous Diff. Plant, KY	500,000	1E-5
Lawrence Livermore Lab., CA	5,300,000	1E-3
Portsmouth Gaseous Diff. Plant, OH	620,000	9E-5
Argonne National Lab., IL	7,900,000	8E-5
Pinellas Plant, FL	1,900,000	2E-4

* In population within 80 km.

Table 2.1-9. Fatal cancers/year to nearby individuals, with Alternative 4 supplemental control technologies (continued).

Site	0-80 km Population	Deaths/y*
Nevada Test Site, NV	3,500	3E-6
Knolls Lab-Kesselring, NY	1,200,000	2E-5
Battelle Memorial Inst., OH	1,900,000	3E-6
Fermi National Lab., IL	7,700,000	1E-6
Sandia National Lab.-Lovelace, NM	500,000	8E-6
Rocky Flats Plant, CO	1,900,000	9E-6
Bettis Atomic Power Lab., PA	3,100,000	1E-6
Knolls Lab-Windsor, CT	3,200,000	2E-6
Pantex Plant, TX	260,000	7E-8
Knolls Lab-Knolls, CT	1,200,000	1E-6
Ames Laboratory, IA	680,000	9E-8
Rocketdyne Rockwell, CA	8,800,000	7E-8

* In population within 80 km.

Table 2.1-10. Distribution of fatal cancer risk in the populations within 80 km with Alternative 4 supplemental control technologies.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	250,000	4E-2
1E-6 to 1E-5	540,000	4E-2
< 1E-6	66,000,000	1E-2
Totals	67,000,000	1E-1

2.2 RMI COMPANY

2.2.1 Description and Existing Controls

2.2.1.1 Site Description

RMI Company (RMI), formerly Reactive Metals, Inc., is located in northeastern Ohio in the City and County of Ashtabula approximately 80 km northeast of Cleveland, 65 km north of Warren, and 80 km north of Youngstown, the closest major population centers. According to the 1980 U.S. Census, the population within 80 km of the facility is about 1.4 million.

2.2.1.2 Major Release Points and Existing Emission Control Technology

RMI operates an extrusion plant which fabricates uranium rods and tubing from ingots for use as fuel elements in nuclear reactors. The ingots are first extruded by a press into either rods or tubing, cooled, and then sectioned by abrasive sawing. Scrap material is fed to a pyrophoric incinerator to form a uranium oxide. The RMI facility also conducts activities an an NRC licensee. Releases from both DOE and NRC activities are included in this assessment.

2.2.2 Basis for the Dose and Risk Assessment

2.2.2.1 Source Terms and Release Point Characterization

The only radioactive material released to the air from RMI is insoluble natural uranium. The total airborne releases, in Ci/y, from all sources during 1986 are listed below in Table 2.2-1.

Table 2.2-1. Radionuclides released to air during 1986 from RMI.*

Nuclide	Release Rate (Ci/y)
U-234	5.6E-4
U-235	4.4E-5
U-238	5.3E-3

* Ajusted, see text.

Releases from the RMI plant consist of natural, depleted, and slightly enriched uranium. During 1986, the year for which the assessment is made, control technology upgrades consisting of HEPA filters were begun at RMI. These upgrades were completed on stack 4 during 1986 and reduced the emissions for that stack from approximately 12,000 μCi for the first half of the year to 0.06 μCi during the second half. The emissions shown in Table 2.2-1 were used to assess the risk. They reflect the emissions during 1986 adjusted to account for the addition of HEPA filters on stack 4. Continued upgrades of the effluent controls during 1987, 1988, and the discontinuation of stacks without HEPA filtration have further reduced emissions. In 1988, RMI reports a total uranium release of 7E-4 Ci/y, approximately a factor of 10 lower than the source term used in this assessment (RMI89).

To evaluate the health impact from the operation of RMI, releases from the facility were assumed to be from six stacks with heights given in the Appendix. The released uranium-234 was assumed to be in equilibrium with its daughters thorium-234 and protactinium-234m. Default particle sizes (1.00 AMAD) and solubility class Y were assumed based on information from RMI (RMI89).

2.2.2.2 Other Parameters Used in the Assessment

The nearest individual was assumed to be located 310 m from the release point (RMI86).

Meteorological data used in the assessment are from Erie, Pennsylvania. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Food consumption rates appropriate to an urban location were used.

2.2.3 Results of the Dose and Risk Assessment

The major contributors to exposure are uranium-234 (52 percent) and uranium-238 (46 percent). The predominant exposure pathway is inhalation for uranium-234 and uranium-238.

The results of the dose and risk assessment are presented in Tables 2.2-2 through 2.2-4. Table 2.2-2 presents the doses

received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.2-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.2-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.2-2. Estimated radiation dose rates from RMI.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	2.5E+1	3.2E+1

Table 2.2-3. Estimated fatal cancer risks from RMI.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
4E-5	8E-4

Table 2.2-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from RMI.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	1	6E-7
1E-6 to 1E-5	98,000	2E-4
< 1E-6	1,400,000	5E-4
Totals	1,400,000	8E-4

2.2.4 Supplementary Controls

As noted in Section 2.2.2.1, RMI has recently completed the upgrade of its effluent control system which was begun in 1986. This has consisted of addition of HEPA filters on stacks 1 and 4 and the discontinuation of unfiltered stacks.

2.2.4.1 Emission Reduction

The upgrade of the effluent control system has resulted in a reduction of uranium emissions. During 1986, when only stack 4 was retro-fittee for half the year, total uranium releases were 1.7E-2 Ci/y. During 1988, with the upgrade complete, total uranium releases were 7E-4 Ci/y, a reduction of 96 percent.

2.2.4.2 Costs of Supplementary Controls

No data were provided by RMI on the costs of the additional effluent controls. Further reductions could be achieved by placing additional HEPA filters in series. No estimates of the costs or efficiencies of such additional controls have been made.

2.3 LOS ALAMOS NATIONAL LABORATORY

2.3.1 Description and Existing Controls

2.3.1.1 Site Description

Los Alamos National Laboratory is one of the prime research and development centers for DOE's nuclear weapons program. This facility is located about 100 km north-northeast of Albuquerque, New Mexico. In addition to defense-related activities, programs include research in the physical sciences, energy resources, environ- mental studies, and biomedical applications of radiation.

Radionuclides are released from 13 technical areas at this site. These areas contain research reactors that produce materials for use in high-temperature chemistry applications, weapons systems development, nuclear safety program development, accelerator operations, biomedical research, development of isotope separation processes, and waste disposal.

2.3.1.2 Major Release Points and Existing Emission Control Technology

The following sections describe the emission control technology currently in use at the six sources being evaluated. Possible application of additional control technology, the effects of such improvements on discharge rates, and the costs of such improvements are also discussed. Generic information on the emission control technology for the nonspecific or minor sources is also provided (Mo86).

2.3.1.2.1 Omega West Reactor Stack

The Omega West research reactor, located in TA-2, is used for a wide variety of experimental programs. The reactor is a heterogeneous water-cooled tank-type reactor, with a maximum power level of 8 MWth.

Argon-41 (t 1/2 =1.8 hr) was the only radionuclide above the limits of detectability released to the atmosphere from the Omega West reactor stack in 1986. The argon-41 is produced by neutron activation of the natural argon in air. Process air streams and part of the building ventilation exhaust are discharged to the atmosphere from the reactor stack, which is located about 300 m from the reactor. The total air flow to the stack is about 28.3 m^3/min. The stack is approximately 0.2 m in diameter, and its height is approximately 46 m above ground level. The stack is continuously monitored. Charcoal cartridges are installed in the process air stream to remove any radioiodine present. There is no technology in place to remove argon-41 from the air stream flowing to the stack. Some reduction in the argon-41 level is provided by delay (approximately one hour) as the air flows from the reactor building to the stack.

2.3.1.2.2 LAMPF Main Stack

The Clinton P. Anderson Los Alamos Meson Physics Facility (LAMPF) in TA-53 consists primarily of a linear proton accelerator, approximately 800 m long, designed to produce an 800 MeV proton beam with an average intensity of one milliampere. The proton beam and secondary particles produced when the energetic protons strike a target are used in a wide variety of experimental programs. Fields of investigation include medium energy nuclear physics, biophysics, radiochemistry, and cancer therapy.

Interaction of the proton beam and secondary particles with air produces several activation products. These activation products, which include beryllium-7, carbon-11, nitrogen-13, oxygen-15, argon-41, and tritium, were the only radionuclides released to the atmosphere from the LAMPF facility in 1986. The activation products are discharged to the atmosphere from the LAMPF main stack. The main stack receives the air flow from a single fan exhaust system. Air flow to the main stack is about 480 m^3/min. The stack has a diameter which varies from about 1.5 m to 0.9 m at the top. The stack height is about 30.5 m above ground level.

Air flowing to the LAMPF stack is passed through a single stage of HEPA filtration to remove particulates. There is no technology in place to remove gaseous radionuclides from the air stream. Areas where the air activation products are produced are continuously ventilated to remove the radionuclides as they are formed. Due to the short half-lives of some of the activation products formed, some reduction in the radionuclide release is obtained by decay due to holdup as the air flows from the various source points to the stack. The extent of the reduction will depend on the radionuclides. In the case of oxygen-15 (t 1/2 = 2.0 min), the holdup could reduce the release significantly. In the case of tritium (t 1/2 = 12.3 yr) and

beryllium-7 (t 1/2 = 53.3 days), the holdup would have essentially no effect on the releases.

2.3.1.2.3 Stack FE-6-HP Site

The tritium handling facility is located at the HP site (TA-33). A wide variety of experimental programs involving the use of tritium is carried out at the facility. Large amounts of tritium are released to the atmosphere from the facility stack (FE-6). A single fan-exhaust system is used to ventilate the facility and feeds to the FE-6 stack. More than 84 percent of the tritium discharged to the atmosphere at LANL is released from Stack FE-6.

The average air flow to the stack is about 200 m^3/min. The stack is 0.61 m in diameter, and the height above ground level is about 23 m.

The tritium handling facility is scheduled to be replaced in several years. Physical containment of the tritium during experimental activities is the principal method for controlling tritium emissions from the tritium handling facility stack. Work areas are ventilated to maintain the tritium concentration, due to leaks, below the concentration guide for controlled areas. A dryer system is used to remove tritiated water from the air flowing to the stack.

2.3.1.2.4 South Stack-Wing 3 - CMR

The Chemistry Metallurgy Research Building (CMR) located in TA-3 is a large multiwinged building in which a wide variety of research programs is carried out. Each wing of the facility is equipped with one or two stacks to handle the wing's air flow. Small amounts of radionuclides are discharged to the atmosphere from most of the building stacks. Wing 3 houses a variety of analytical chemistry groups which provide services for the entire laboratory. Approximately 55 percent of the plutonium released to the atmosphere at LANL in 1986 was discharged from the south stack of Wing 3 of the facility. No other radionuclides were detected in the stack air flow in 1986.

The air flow to the stack comes from a single fan and exhaust system (FE-19) serving a number of laboratories. The air flow to the stack is about 1,400 m^3/min. The stack has a diameter of about 1 m, and the height above ground level is about 17 m. The air flowing to the south stack of Wing 3 of the Chemistry Metallurgy Research Building is passed through a two-stage prefilter and a single-stage bag filter prior to discharge from the stack. It is estimated that the filter system removes 90 to 95 percent of the particulates.

2.3.1.2.5 Main Stack - Building 3-DP Site

Building 3 at the DP site (TA-21) is used for enriched uranium recovery operations. Small amounts of uranium are discharged to the atmosphere from several stacks used to ventilate the building. Uranium-235 released from the main stack of the building accounted for about 55 percent of the total uranium released to the atmosphere at LANL in 1986. The chemical form of the uranium released from the stack is unknown. No other radionuclides were detected in the air leaving the stack.

The main building stack serves to ventilate building work areas using a single fan-exhaust system (FE-1). Air flow to the stack is 480 m^3/min. The stack is about 1 m in diameter and about 15 m above ground level. There is no equipment in place to reduce emissions from the main stack of Building 3, except for local HEPA filters in gloveboxes.

2.3.1.2.6 Core Wing Stack Radiochemistry Site

The radiochemistry site in TA-48 is used for a variety of programs involving radioactive materials. Laboratory hoods, glove boxes, and "hot cells" are used to contain the radioactive materials. Small quantities of radioactive materials are released to the atmosphere from several stacks at the facility. About 87 percent of the mixed fission products (MFP) released to the atmosphere at LANL in 1986 were released from the Core Wing Stack, which is one of the stacks used to ventilate the radiochemistry facility.

Two fan-exhaust systems (FE-45 and FE-46) discharge into the Core Wing Stack. A number of glove boxes are serviced by the two fan-exhaust systems. Total air flow to the stack is about 1,400 m^3/min, with the air flow almost equally divided between the two fan-exhaust systems. The Core Wing Stack has a diameter of about 1.5 m and a height of approximately 21.3 m above ground level. The glove boxes which discharge to the two fan-exhaust systems serving the Core Wing Stack are provided with a single stage of HEPA filters.

2.3.2 Basis for the Dose and Risk Assessment

2.3.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed below in Table 2.3-1.

In modeling the site, all releases were assumed to be made from the LAMPF, since this is the major source of dose. The releases were assumed from a 30.5-m stack. Default particle sizes (1.00 Amad) and solubility classes (Class D for carbon-11, nitrogen-13, and oxygen-15) were assumed.

Table 2.3-1. Radionuclides released to air during 1986 from Los Alamos Scientific Laboratory.

Nuclide	Release Rate (Ci/y)
Ar-41	7.3E+2
C-11	1.8E+4
H-3	1.1E+4
I-131	3.8E-5
N-13	4.8E+3
O-14	2.6E+3
O-15	8.6E+4
P-32	7.0E-5
Pu-238	9.9E-5
Pu-239	1.1E-4
Sr-90	2.6E-3
U-235	7.1E-4
U-238	1.4E-4

2.3.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Santa Fe, New Mexico. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 750 m from the assumed release point (Em87). Food consumption rates appropriate to an urban location were used.

2.3.3 Results of the Dose and Risk Assessment

The major contributors to exposure are oxygen-15 (57 percent), carbon-11 (29 percent), and nitrogen-13 (7 percent). The predominant exposure pathway is air immersion.

The results of the dose and risk assessment are presented in Tables 2.3-2 through 2.3-4. Table 2.3-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.3-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.3-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.3-2. Estimated radiation doses from the Los Alamos Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Gonads	9.5E+0	1.0E+1
Remainder	7.4E+0	1.1E+1
Breast	8.9E+0	9.7E+0
Lungs	8.8E+0	1.1E+1
Red marrow	7.0E+0	9.2E+0

Table 2.3-3. Estimated fatal cancer risks from the Los Alamos Laboratory.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
2E-4	4E-3

Table 2.3-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Los Alamos Scientific Laboratory.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	1	3E-6
1E-5 to 1E-4	2,500	9E-4
1E-6 to 1E-5	100,000	2E-3
< 1E-6	50,000	7E-4
Totals	160,000	4E-3

2.3.4 Supplementary Controls

2.3.4.1 LAMPF Main Stack

The results of the dose and risk assessment show that 98 percent of the dose is due to emissions of oxygen-15, carbon-11, and nitrogen-13, short-lived air activation products from the LAMPF Main Stack.

A permanent committee was formed at LANL several years ago to review LAMPF operations (Em87, Mo85). One objective of the committee is to evaluate potential methods for reducing releases of airborne radioactivity from LAMPF operations. One plan currently under consideration is to enclose one of the primary beam stop areas, which is a major producer of air activation products. The enclosed area would not be vented during accelerator operation. Venting would be done only after the accelerator shuts down and the short-lived radioisotopes have had a chance to decay. The overall effectiveness of the proposed modification for reducing airborne emissions from LAMPF has not been determined. If the plan is implemented, construction of the enclosure will start within two years (Mo86).

The large air flow to the LAMPF main stack (about 480 m^3/min) makes it very difficult to use any existing technology to remove the gaseous activation products from the air stream. The most realistic approach would be to provide additional holdup time to allow some decay of the short half-lived radionuclides, as indicated above. Extremely large air storage volumes would be required to reduce radionuclide releases significantly. For example, if an atmospheric pressure air storage system having a storage volume of 9,300 m^3 were applied to the air flowing to the LAMPF stack, the additional holdup time provided would be about 19.4 minutes. Table 2.3-5 presents the reductions in radionuclide emissions as a function of holdup time.

Table 2.3-5. Effect of holdup time on the release of air activation products from the proposed stack serving the LAMPF beam stop.

Radionuclide	Fraction of the Radionuclide Generated at the Beam Stop Released to the Atmosphere	
	Single Tank (20 min. additional holdup time)	Dual Tank (40 min. additional holdup time)
Oxygen-15	0.00108	1.18E-6
Carbon-11	0.505	0.255
Nitrogen-13	0.25	0.0625

As a result, total emissions from the stack would be reduced from about 109,000 Ci/y to about 5,000 Ci/y at the same level of programmatic activities.

The air storage tank would be constructed of carbon steel and located on a concrete pad adjacent to the LAMPF stack,

assuming adequate space is available. A tank with a storage volume of 9,300 m^3 would be 30 m in diameter by about 13.2 m high.

The estimated capital cost for an atmospheric pressure air storage system, with a storage volume of 9,300 m^3, would be about $1,600,000. The estimated operating costs would be about $90,000 per year. The capital cost of air storage systems of varying size would vary approximately as the eight-tenths power of the size ratio. Annual operating costs would be almost independent of the size ratio (Mo86).

2.3.4.2 Omega West Reactor Stack

The Omega West research reactor, located in TA-2, is a heterogeneous water-cooled tank-type reactor. The maximum power level is 8 MWth. The reactor is used for a wide variety of experimental programs. The reactor is under DOE jurisdiction and meets DOE standards for research reactors which are equivalent to NRC standards for research reactors.

Argon-41 is produced by neutron activation of the natural argon in air. Process air streams and part of the building ventilation exhaust are discharged to the atmosphere from the reactor stack, which is located about 300 m from the reactor. The total air flow to the stack is about 28.3 m^3/min. The stack is approximately 0.2 m in diameter, and its height is approximately 46 m above ground level. The stack is continuously monitored.

The argon-41 released from the reactor stack can be reduced by providing additional holdup time to allow the argon-41 to decay. An atmospheric pressure or pressurized air storage system could be used to provide the holdup time. The atmospheric pressure storage volumes required to obtain various reductions in the argon-41 emissions at a normal airflow of 28.3 m^3/min are given in Table 2.3-5. The use of a pressurized air storage system would reduce the storage volume required for a given decontamination factor (DF) but would probably increase the overall cost of the system.

2.3.4.3 Stack FE-6-HP Site

The tritium handling facility is located at the HP site (TA-33). A wide variety of experimental programs involving the use of tritium is carried out at the facility. Large amounts of tritium are released to the atmosphere from the facility stack (FE-6). A single fan-exhaust system is used to ventilate the facility and feeds to the FE-6 stack. More than 84 percent of the tritium discharged to the atmosphere at LANL is released from Stack FE-6.

The average air flow to the stack is about 200 m^3/min. The stack is 0.61 m in diameter, and the height above ground level is about 23 m.

The tritium handling facility is scheduled to be replaced in several years.

The chemical form of the tritium is unknown, but since any tritiated water should be removed by the dryer, the tritium is probably present as molecular hydrogen.

The large volume of air flowing to Stack FE-6 and the very low concentration of tritium in the air make effective reduction of the tritium released from the stack both difficult and costly. In addition, because the tritium handling facility is to be replaced in a few years, it is difficult to justify large expenditures for additional emission control technology.

Assuming the tritium is present in the air stream primarily as molecular hydrogen, adequate removal of the tritium from the air would require its conversion to water. A drying step would then be required to remove the tritiated water from the air prior to discharge. Subsequent recovery of the tritiated water from the dryer and its final disposal would present additional problems. A risk analysis would have to be carried out to determine if disposal of the tritiated water would present less of a risk then release of the tritium, as molecular hydrogen, to the atmosphere.

If removal of tritium from the air flowing to Stack FE-6 becomes necessary, a recovery system similar to the emergency tritium cleanup system (ETC) which is used at the Tritium Systems Test Assembly (TSTA) at LANL could probably be used. The ETC system is designed to process air at the rate of about 39 m^3/min. Therefore, a similar system for Stack FE-6 would have to be designed for air flow about five times as large (200 m^3/min). The ETC system was not intended for continuous operations, but only for emergency use. However, the system could probably be designed for continuous use.

2.3.4.4 South Stack-Wing 3 - CMR

The Chemistry Metallurgy Research (CMR) Building located in TA-3 is a large multiwinged building, housing a wide variety of research programs. Each wing of the facility is equipped with one or two stacks to handle the wing air flow. Small amounts of radionuclides are discharged to the atmosphere from most of the building stacks. Wing 3 houses a variety of analytical chemistry groups which provide services for the entire laboratory. The air flow to the stack comes from a single fan and exhaust system (FE-19) serving a number of laboratories. The air flow to the stack is about 1,400 m^3/min. The stack has a diameter of about 1 m, and the height above ground level is about 17 m.

The chemical form and isotopic composition of the plutonium discharged are unknown.

Because the amount of plutonium released from the stack in question and its effect on the environment are already very small, additional equipment to reduce the plutonium release probably would result in only slight decreases in the total risks due to this facility. If additional reductions are necessary, however, they could be attained by installing a HEPA filter system in addition to or in place of the existing bag filter system. A bank of at least 48 HEPA filters, measuring 61 cm x 61 cm x 30 cm would be needed to handle the air flow. The HEPA filter system would provide at least a 99 percent reduction in the plutonium release from the stack.

2.3.4.5 Main Stack - Building 3-DP Site

Building 3 at the DP site (TA-21) is used for enriched uranium recovery operations. Small amounts of uranium are discharged to the atmosphere from several stacks used to ventilate the building. Uranium-235 released from the main stack of the building accounted for about 34 percent of the total uranium released to the atmosphere at LANL in 1986. The chemical form of the uranium released from the stack is unknown. No other radionuclides were detected in the air leaving the stack.

The main building stack serves to ventilate building work areas using a single fan-exhaust system (FE-1). Air flow to the stack is 480 m^3/min. The stack is about 1 m in diameter, and the height of the stack is about 15 m above ground level.

The amount of uranium released from the main stack of Building 3 is already very small, and its effect on the environment is minimal. If reductions become necessary, however, a filter system could probably be installed. A HEPA filter system would be preferred. A bank of at least 18 HEPA filters measuring 61 cm x 61 cm x 30 cm would be required to handle the air flow to the stack.

Installation of a HEPA filter system would provide at least a 99.9 percent reduction in the uranium release from the stack. The system would consist of three modules, each rated at 250 m^3/min, with two modules in operation and one module in standby. Each module would consist of nine HEPA filters, two dampers, and one 300 m^3/min blower.

2.4 HANFORD RESERVATION

The Hanford Reservation was established in 1943 as a plutonium production facility for nuclear armaments. Information used to evaluate the facility was obtained from DOE and Hanford reports (Mo84, PNL87). Plutonium production has decreased, and other programs have filled the gap, such as management and storage of radioactive wastes, reactor operations, fuel

fabrication, energy research and development, and biophysical and biomedical research. The reservation, which is located 270 km south of Seattle, Washington, is separated into four areas, which are designated the 100, 200, 300, and 400 Areas. The activities of each area are described briefly.

2.4.1.1 100 Area

The 100 Area contains the nine plutonium production reactors for which the site was originally developed. Eight of these reactors are currently shut-down. Operating facilities during 1986 include the N-Reactor and the 1706 Laboratory, which provides support services for the reactor. N-Reactor has subsequently been shut-down pending the resolution of safety concerns.

2.4.1.2 200 Area

Activities conducted in the 200 Area include fuel processing, nuclear waste treatment and storage, equipment decontamination, and research. Plutonium reclamation from spent fuel is performed at the PUREX Plant in this area.

2.4.1.3 300 Area

The major facilities in the 300 Area are the Hanford Engineering Development Laboratory, the fuel fabrication facility, and the Life Sciences Laboratory. The Hanford Engineering Development Laboratory, the largest operation in this area, supports all activities of the development program for the fast breeder reactor. Life science research in this area includes plutonium inhalation studies and other programs investigating the physiological effects of radioactive materials.

2.4.1.4 400 Area

The only facility currently in operation in the 400 Area is the Fast Flux Test Facility. When the Fuel Materials Examination Facility currently under construction is completed, the 400 Area will be the center of the Hanford breeder reactor research program.

2.4.2 Major Release Points and Existing Emission Control Technology

2.4.2.1 Stack 116-N Serving the 105-N Reactor Building

Argon-41, which constitutes the primary airborne radioactive emission from N-Reactor, is produced from the leakage of air into the reactor system and subsequent activation of the stable argon in the air. Noble gases and volatile fission products, such as xenon-133 and iodine-131, come from leaks in fuel element claddings. Nonvolatile particulate fission and activation

products, such as cobalt-60, europium-154, and molybdenum-99, become airborne as a result of the primary coolant contacting exposed surfaces, then drying and becoming suspended in air currents.

The ventilation systems in 105-N are separated into five zones based on their potential for contamination with airborne radioactive material. The 116-N stack is the main discharge point for airborne radioactive material from N-Reactor. Immediately preceding the 116-N stack is the 117-N filter and diversion facility through which the exhaust air is routed prior to release to the stack. The stack exhausts to the atmosphere 61 m above ground level.

The 117-N facility contains four separate air filtration cells. The air from Zones I, II, and III of the 105-N building enters through three separate ducts. Air from Zone I passes through two filtration cells, air from Zone II passes through a third filtration cell, and air from Zone III normally bypasses the filter cells as it is routed through the facility. In the event of an emergency, however, Zone III exhaust can be combined with Zone II exhaust to provide filtration for Zone III exhaust. The fourth filtration cell is on standby for emergency backup.

The first, second, and fourth filtration cells are composed of a series of three filter bank stages. The first stage is an aluminum mesh screen used as a moisture separator to protect the remaining filters in the event of entrained moisture in the air stream. The second stage is a high-efficiency particulate air (HEPA) filter. Minimum efficiency for removal of particulate matter larger than 0.3 microns is 99.97 percent. These filters are routinely tested for efficiency. The third stage contains granular activated charcoal which removes 95 percent of the inorganic halogen gases in the air stream.

The third filtration cell contains two stages, a HEPA filter and an activated charcoal absorber.

Zones IV and V serve offices, administration areas, and the reactor control room. Ventilation air from these areas is exhausted through roof exhausters without treatment.

2.4.2.2 PUREX Main Stack No. 291-A-1

The four sources of gases that exhaust through the 61-m-high 291-A-1 main stack of the Hanford PUREX facility are: the declad and dissolver off-gas system, the process off-gas system, the plutonium oxide conversion facility off-gas system, and the canyon ventilation system.

2.4.2.2.1 Declad and Dissolver Off-Gas System

The PUREX facility has the capability to process irradiated fuel to separate and recover plutonium, uranium, and neptunium.

In the head-end of the process, the cladding is chemically removed from the fuel elements and the fuel is then dissolved in the same vessel. The decladding and dissolving are accomplished in three dissolver vessels. The dissolvers have parallel systems for treatment of the declad and dissolver off-gases.

The declad off-gases first go through a downdraft condenser tower that condenses moisture and removes part of the nitrogen oxides as nitric acid. The gases pass through an ammonia scrubber and then through a steam heater and an electric heater. The gases are heated to 196°C before passing into the silver reactor.

The decontamination factor (DF) for the silver reactor averages 100. The cell B silver reactor has a 2.44-m deep packing bed of 1.3 cm ceramic saddles, while the cells A and C silver reactors have a 0.88-m deep bed of 1.3-cm saddles on top of a 0.30-m deep bed of 2.5-cm saddles. The saddles are coated with silver nitrate. Iodine-129 and iodine-131 are removed in the silver reactor. When the efficiency falls, the silver reactor bed is regenerated with fresh silver nitrate solution that is then baked on the packing. When a reactor becomes plugged, it is replaced and sent to a low-level waste burial ground.

From the silver reactor, the declad gases pass through two deep-bed glass fiber filters in series. The gases are then exhausted through the main stack, 291-A-1.

During the dissolution step, the gases follow a similar path. The ammonia scrubber does not operate during dissolution. The gases exiting the second glass fiber filter are routed to the 293-A Building in which two acid absorbers in series remove 90 percent of the remaining iodine and 90-92 percent of the remaining nitrogen oxides. The gases are then sent to the main stack, 291-A-1.

Krypton-85 is a major radionuclide released during the declad and dissolving processes. There is no cleanup of krypton-85 at PUREX.

2.4.2.2.2 Process Off-Gas System

The PUREX process produces off-gases from condensers and other process equipment. These are combined and routed through the process off-gas cleanup system.

The gases go through a condenser to remove the condensable vapors. Then the noncondensable gases are heated in a steam heater to 160°C and pass through a silver reactor that removes radioactive iodine that remained in solution during the fuel dissolving process and that evolves during processing steps. This silver reactor has a very low efficiency. From the silver reactor, the gases pass through a deep-bed glass fiber filter and from there to the ventilation system No. 1 air tunnel.

2.4.2.2.3 Plutonium Oxide Conversion Facility Off-Gas System

While the PUREX plant has been on standby, a plutonium oxide facility has been added to the plant.

Off-gases from the plutonium nitrate storage vessels and the prereduction tank pass through a heater and then through two stages of HEPA filtration. There is a combined flow of about 1,583 l/min at 60°C. Blowers deliver these gases to the ventilation system No. 1 air tunnel.

Off-gases from the calciner pass through a porous stainless steel filter at a flow rate of about 186 l/min at 157°C to remove plutonium oxide particles. These gases, along with the off-gases from the filtrate concentrator and the vessel vent gases from the oxide rework facility, are fed to a scrubber to remove nitric acid. The off-gases from the vacuum header pass through a vacuum tank and are combined with the scrubber off-gases. The combined gases then pass through two vacuum dropout tanks in series to remove entrained liquids. The combined gas flow of about 400 l/min then goes through a heater and two stages of HEPA filtration in series. A vacuum pump delivers the gases to the blowers that exhaust to the ventilation system No. 1 air tunnel.

2.4.2.2.4 Canyon Ventilation System

Ventilation system No. 1 provides ventilation air for the process cells in the PUREX canyon. Added to this air are the gases from the process off-gas cleanup system and from the plutonium oxide conversion facility off-gas treatment system.

The combined gases are exhausted through filters at a flow rate of 3,570 m^3/min. Two glass fiber filters and one HEPA filter are installed in parallel. Each unit is designed to handle the full canyon ventilation air flow. Unit one, which was installed in 1955, now has marginal capacity because of the accumulation of solids. Unit two is run in parallel with unit one. Unit three is on standby. The filters are installed underground. When they are no longer usable, they will be sealed and left in place. Recent tests have shown the two fiberglass filters to have efficiencies greater than 99.95 percent for 0.3 micron particles. Unit three is designed to remove 99.97 percent of the 0.3 micron particles from the ventilation air. Fans deliver the filtered gases to the PUREX main stack, 291-A-1.

2.4.2.3 Combined Exhaust, Buildings 405, 4621E, 4717; Building 491-S, and Building 4717

Radioactive gases generated in the Fast Flux Test Facility (FFTF) are a result of neutron activation of the reactor cover gas or are released from the fuel through defective fuel cladding. These gases are processed through the Radioactive

Argon Processing System (RAPS) and released to the atmosphere through the combined exhaust. There are about 200-280 l/min of gases from this source. Effluent from cells and spaces subject to potential contamination is processed through the Cell Atmosphere Processing System (CAPS) before release through the combined exhaust. The CAPS contributes about 1,700-2,000 l/min to the combined exhaust. Other contributions to the combined exhaust are about 100 m^3/min from the normal heating and ventilating system and about 570 m^3/min from the containment heating and ventilating system.

Gases from the fission gas monitor and from the argon blower and valve cell exhaust go through the 491-S Building directly to the atmosphere without treatment. Should the monitors on the inlet to Building 491-S show the presence of radionuclides, the gases can be routed through the CAPS. If the Building 491-S outlet monitors show contamination, a routing through HEPA filters is available. The Building 4717 lower area heating and ventilating system exhausts directly to the atmosphere. Should the radiation monitor detect contamination, the blowers would be shut down until the situation could be evaluated.

2.4.2.4 Radioactive Argon Processing System (RAPS)

Inputs to the RAPS consist of about 170-200 l/min of argon reactor cover gas and about 28-57 l/min bleed from the argon atmosphere hot cell. The compressors, one online and one on standby, draw the gases through a vacuum tank and filters which remove moisture and oils. The gases then pass to a surge and delay tank equipped with baffles, which delays their passage for about 30 hours to allow decay of argon-41. From the surge and delay tank, the gases pass to the cold box, which operates at cryogenic temperatures. Heat exchangers using liquid nitrogen cool four charcoal-delay beds that operate in series. The adsorption of the gases by the charcoal beds provides about 3.25 days of delay for krypton and about 284 days of delay for xenon. This allows for decay of the short-lived radioisotopes. If there has been no failed fuel cladding, the gases would then be routed to the combined exhaust or to the CAPS. If there has been some failed fuel cladding, longer-lived noble gases could be present. In this case, the gases from the charcoal-delay beds would be routed to a liquid nitrogen-cooled fractional distillation column. Here, the liquid portion would contain the longer-lived noble gases. The liquid would be warmed and the noble gases sent to a noble gas storage vessel. The gas portion from the fractional distillation column would be routed to the combined exhaust or to the CAPS.

2.4.2.5 Cell Atmosphere Processing System (CAPS)

Inputs to the CAPS consist of: (1) about 1,415 l/min of discharge from nitrogen atmosphere cells, (2) about 1-2 l/hr from the gas chromatograph that samples the argon atmosphere reactor cover gas, (3) about 425-570 l of contaminated argon about once a

week from the gas tag sample trap, and (4) effluent from the RAPS, if radiation monitors detect radioactivity above 1E-3 microcuries per cubic centimeter.

As with the RAPS, the gases are drawn into a vacuum tank and through filters to remove moisture and oils by two compressors, one online and one on standby, and thence into a surge and delay tank for decay of argon-41. The CAPS input flow normally has a very low radioactivity level (<<1E-7 uCi/cc). In normal operation, the gases from the surge and delay tank are then routed to the combined exhaust. If radiation monitors detect radiation, the gases are routed to the cold box. Two drying units dry the gas to a dewpoint of -68°C or less. The liquid from the drying unit may contain some tritium and is sent to the liquid waste system. Two liquid-nitrogen-cooled charcoal-delay beds in series provide decay time for short-lived radionuclides. If the gases exiting the charcoal-delay beds have a radioactivity of less than 5E-3 uCi/cm^3, they are routed to the combined exhaust. If the radioactivity exceeds this limit, the gases are routed back to the CAPS vacuum tank for another pass through the CAPS.

2.4.3 Basis for the Dose and Risk Assessment

2.4.3.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.4-1.

In modeling the site, all releases were assumed to be made from the 200 Area, since this is the major source of dose, and the nearest individual at risk is assumed to be 15,000 m from the source (PNL87). The releases were assumed from a 10-m stack. Default particle sizes (1.00 AMAD for plutonium-238) and solubility classes (Class Y for plutonium-238) were assumed.

2.4.3.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Moses Lake/Grant, Washington. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 15,000 m from the assumed release point.Food consumption rates appropriate to a rural location were used.

2.4.3 Results of the Dose and Risk Assessment

The major contributors to exposure are argon-41 (61 percent) and plutonium-238 (33 percent). The predominant exposure pathways are inhalation for uranium-238 and air immersion for argon-41.

Table 2.4-1. Radionuclides released to air during 1986 from the Hanford Reservation.

Nuclide	Release Rate (Ci/y)
Am-241	5.3E-4
Ar-41	1.3E+5
Ce-144	2.6E-3
Co-60	1.1E-2
Cs-137	8.0E-3
Cs-138	1.9E+3
H-3	8.7E+1
I-129	5.3E-1
I-131	5.6E-1
I-132	2.6E-1
I-133	2.3E+0
I-135	3.5E-1
Kr-85	5.3E+5
Kr-85m	3.3E+2
Kr-87	8.5E+2
Kr-88	3.6E+2
La-140	3.4E-2
Mo-99	9.6E-2
Nb-95	3.5E-3
Pb-212	1.8E-1
Pm-147	1.2E-2
Pu-238	8.9E-2
Pu-239	3.2E-3
Pu-241	1.4E-2
Rb-88	3.6E+2
Ru-106	4.5E-1
Sn-113	1.8E-1
Sr-90	1.2E-3
Tc-99	2.0E-4
U-234	6.8E-5
U-235	8.4E-6
U-236	5.4E-7
U-238	4.2E-5
Xe-133	6.7E+1
Xe-135	1.3E+3
Zr-95	4.0E-3

The results of the dose and risk assessment are presented in Tables 2.4-2 through 2.4-4. Table 2.4-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.4-5 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.4-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.4-2. Estimated radiation dose rates from the Hanford Reservation.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	2.8E+0	5.6E+1
Remainder	1.0E+0	1.7E+1
Gonads	1.1E+0	1.5E+1
Endosteum	6.3E+0	1.7E+2
Red marrow	1.2E+0	2.3E+1
Breast	9.4E-1	1.2E+1

Table 2.4-3. Estimated fatal cancer risks from the Hanford Reservation.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
3E-5	6E-3

Table 2.4-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Hanford Reservation.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	5,200	1E-3
1E-6 - 1E-5	140,000	4E-3
< 1E-6	210,000	1E-3
Totals	350,000	6E-3

2.4.5 Supplementary Controls

The N-Reactor shutdown in 1987 has reduced emissions of argon-41 and plutonium-238 sufficiently to lower the estimated maximum exposure below 1 mrem/y. Therefore, additional emission controls for airborne radionuclides are not discussed.

2.5 OAK RIDGE RESERVATION

2.5.1 Description and Existing Controls

The Oak Ridge Reservation (ORR), located in eastern Tennessee, occupies approximately 15,000 ha in a valley between the Cumberland and southern Appalachian mountain ranges. The ORR lies just southwest of the city of Oak Ridge and about 24 km west of Knoxville, Tennessee. The reservation is bounded on the northeast, southeast, and southwest by the Clinch River.

2.5.1.1 Site Description

The major facilities at the ORR are the Y-12 plant, the Oak Ridge National Laboratory (ORNL), and the Oak Ridge Gaseous Diffusion Plant (ORGDP). In addition to these major facilities, the Oak Ridge Associated Universities and the Comparative Animal Research Laboratory are also located at the site.

The Y-12 plant, located adjacent to the city of Oak Ridge, is a major nuclear weapons production facility, processing enriched uranium. Its major missions include fabricating nuclear weapons components, processing source and special nuclear material, and providing support to the weapons design laboratories. While the actual processes employed at the Y-12 plant are classified, the activities associated with these missions include production of lithium compounds, recovery of enriched uranium from scrap materials, and fabrication of uranium and other materials into finished parts and assemblies. Fabrication operations include vacuum casting, arc melting, powder compaction, rolling, forming, heat treating, and machining.

The ORNL is a large multipurpose research laboratory where basic and applied research in all areas relating to energy is conducted. The ORNL facilities include nuclear reactors, chemical pilot plants, research laboratories, and radioisotope production laboratories.

The significant airborne radioactive emissions from the ORNL are from the Central Radioactive Gas Disposal Facility (CRGDF) and the Tritium Target Fabrication Building. The CRGDF is equipped with charcoal filters for radioiodines and HEPA filters for particulate emissions. There are no controls for the noble gases krypton and xenon or for tritium. The Tritium Target Fabrication Building also releases tritium without effluent control.

Until the summer of 1985, the ORGDP's primary mission was to provide enriched uranium for use in nuclear reactors. The ORGDP uses the gaseous diffusion process. The facility was placed in "ready standby" in August 1985. Since that time, the decision has been made to shut down permanently the enrichment cascade. ORGDP is also involved in developing and demonstrating more

energy-efficient and cost-effective methods of enriching uranium, such as the gas centrifuge process and the atomic vapor laser isotopic separation (AVLIS) system. However, the gas centrifuge process was shut down in 1985, and the work on AVLIS has been significantly reduced.

2.5.1.2 Major Release Points and Existing Emission Control Technology

There are approximately 350 process exhaust stacks at the Y-12 plant, of which approximately 85 serve operations with the potential to release uranium to the atmosphere. Although actual emission controls are classified, it is known that the majority of the stacks serving uranium operations are equipped with particulate control devices such as HEPA and fabric filters.

The purge cascade was the largest source of airborne radioactive emissions at the ORGDP. Effluents from the purge cascade were passed through sodium fluoride traps, alumina traps, and potassium hydroxide (KOH) scrubbers.

2.5.2 Basis for the Dose and Risk Assessment

2.5.2.1 Source Terms and Release Point Characterization

The airborne emissions from all facilities at ORR are summarized in Table 2.5-1. These emissions data were obtained from the DOE's Effluent Information System and the Annual Environmental Monitoring Report for 1986 (Or87a, Or87c).

In modeling the site, all releases were assumed to be made from the Y-12 plant, since this is the major source of uranium. Data on the actual stacks at the Y-12 Plant are classified. Therefore, the releases were assumed from a 10-m stack, with a flow of 200 cfm (Mo86).

Default particle sizes (1.00 AMAD) were assumed. The uranium-234 was assumed to be one-half solubility class W and one-half solubility class Y.

2.5.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Knoxville, Tennessee. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located in the city of Oak Ridge, 750 m from the assumed release point. Food consumption rates appropriate to a rural location were used.

Table 2.5-1. Radionuclides released to air from Oak Ridge Reservation during 1986.

Nuclide	1986 Emissions (Curies/year) Y-12	ORNL	ORGDP	Other	Total
C-14				1.0E-4	1.0E-4
Cu-64				2.0E-6	2.0E-6
Ga-67				3.0E-6	3.0E-6
H-3		3.1E+4		4.0E-3	3.1E+4
I-125				1.5E-5	1.5E-5
I-131		3.6E-2		1.3E-4	3.6E-2
Kr-85		1.1E+4			1.1E+4
Pa-234M			3.7E-4		3.7E-4
Tc-99	1.3E-2		1.2E-1		1.3E-1
Tc-99M				3.0E-6	3.0E-6
Th-234			3.7E-4		3.7E-4
Tl-201				5.0E-6	5.0E-6
U-234	7.0E-2		7.4E-3		7.7E-2
U-234	7.7E-2				7.7E-2
U-235	6.4E-3				6.4E-3
U-236			8.0E-6		8.0E-6
U-238	2.8E-2		3.6E-4		2.8E-2
Xe-133		5.2E+4			5.2E+4
Y-90				2.0E-5	2.0E-5

2.5.3 Results of the Dose and Risk Assessment

The major contributors to exposure are uranium-234 (40 percent), tritium (35 percent), and uranium-238 (13 percent). The predominant exposure pathway is inhalation for uranium-234 and uranium-238, and ingestion for tritium.

The results of the dose and risk assessment are presented in Tables 2.5-2 through 2.5-4. Table 2.5-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.5-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.5-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.5-2. Estimated radiation dose rates from the Oak Ridge National Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	2.2E+1	4.3E+2
Remainder	2.0E+0	7.8E+1

Table 2.5-3. Estimated fatal cancer risks from the Oak Ridge National Laboratory.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/year
8E-5	3E-2

Table 2.5-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Oak Ridge National Laboratory.

Risk Interval	Number of Persons	Deaths/year
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	28,000	8E-3
1E-6 - 1E-5	760,000	3E-2
< 1E-6	60,000	8E-4
Totals	850,000	3E-2

2.5.4 Supplementary Controls

The emission control technology (ECT) currently used to reduce airborne radioactive emissions at facilities in the major Oak Ridge areas was described in Section 2.5.1. Potential additional emission control technologies are described in the following sections (Mo86).

2.5.4.1 Additional Emission Control Technology for the ORNL Central Radioactive Gas Disposal Facility

The major portion of the radiological hazard from the gas disposal facility is due to the emission of tritium. Practical control technology exists for removal of these materials from low flow rate air streams only. Because of the high rate of emission (64 m^3/sec) from the stack of this facility, additional control technology must be implemented before the individual source stream is diluted with ventilation air or other gas streams.

Much of the tritium emission is in the form of tritiated water. This portion can be removed by passing the source stream through a dryer containing molecular sieve materials for water removal and then regenerating the adsorber material with heat. A pair of such dryers, operated alternately, will provide for the continuous removal of tritiated water from the source. Table 2.5-5 presents the expected emission rate for tritium at the CRGDF if this additional control technology is implemented.

Table 2.5-5. Anticipated new emission rate for tritium at CRGDF.

Present Emission Rate (Ci/y)	Postulated ECT Removal Efficiency	New Emission Rate (Ci/y)
3.12E+4	90%	3.12E+3

The cost of an emission control system for the removal of tritiated water is estimated at $1.66 million. This includes $1 million for construction, $0.2 million for engineering, and a $0.46 million contingency. These cost estimates are highly dependent upon the ease of incorporating the potential controls into the existing gas handling system. It is possible that the existing gas handling system would have to be completely replaced to accommodate more controls.

2.5.4.2 Additional Emission Control Technology for the ORNL Tritium Target Fabrication Building

Tritiated water can be removed from the gaseous exhaust by passing the exhaust air stream through a dryer containing molecular sieve materials for water removal and then regenerating the adsorber material by the application of heat. A pair of such driers, operated alternately, would provide for the continuous drying of the exhaust and the collection of tritiated water for storage or further processing.

Analytical information concerning the gases present in the stack exhaust indicates that only about 1 percent of the tritium is in the form of tritiated water. At this time, it is not

practical to remove tritium in the form of hydrogen gas from large gaseous stream flow emanating from the Tritium Target Fabrication Building.

It is postulated that over 90 percent of the tritiated w would be collected by the application of the additional technology; however, since the tritiated water represents onl small portion of the total tritium from this facility, the present emissions (1.2×10^3 Ci/y) would not be significantly reduced.

The cost of an emission control system specifically desi for the removal of tritiated water is estimated at $1.66 mill This includes $1 million for construction, $0.2 million for engineering, and a $0.46 million contingency.

2.5.4.3 Additional Emission Control Technology for the Y-12 Plant (Uranium Product Recovery)

There are three sources of emissions from uranium produc recovery. The major source is the West Head House, Building 9212. The emission controls described here apply to this facility.

Installation of an additional stage of HEPA filters woul reduce the amount of particulate emission and uranium-234 and uranium-238 that bypasses the present ECT system, if the pres ductwork can be adapted or expanded to allow incorporation of more HEPA filters downstream of the existing filter system.

HEPA filters are estimated to remove at least 99.95 perc of particulate materials in a single pass. It has been shown however, that uncollected materials have a lower collection efficiency when passed through a second HEPA filter stage. Collection efficiency estimates for such a second stage may vary due to the size distribution of the original particulates. It is postulated that a second HEPA filter installed in series will remove 99 percent of the remaining particulates and reduce the amount of uranium-234 from 0.154 Ci/y to 0.093 Ci/y, and redu the amount of uranium-238 from 2.8E-2 Ci/y to 3.0E-3 Ci/y.

The cost of the control devices presently installed in t uranium product recovery facility is $55,000. The estimated for installation of backup HEPA filtration within the existin system is an additional $20,000. The present annual operatin cost is $14,640. Based upon the assumption that the air capacityof the system can be maintained by the existing fan system, additional power and HEPA changeout requirements woul increase the operating cost about 20 percent.

If significant structural additions or modifications are necessary for proper operation and maintenance of the expande air control system, then significant cost increases can be anticipated. In addition to the HEPA filter cost, modificati

that include ductwork, blowers, dampers, instrumentation, and electrical work would increase the cost to about $455,000. Engineering costs of about $115,000 and a 35 percent contingency would raise the total project cost to over $800,000. Major structural additions will further increase the cost. Operating costs are expected to double with the implementation of this modified system.

2.5.4.4 Additional Emission Control Technology for the Y-12 Plant (Uranium Product Preparation)

Replacing the existing scrubber with a high-energy venturi scrubber and adding a backup stage of HEPA filtration would reduce the emission of uranium-234 from this facility, if the present ductwork can be modified or expanded to allow incorporation of these changes.

Based upon the arguments presented in Section 2.5.4.3, about 99 percent of the particulate emission would be removed by the addition of a second HEPA filter stage. In addition, the use of a high-energy venturi scrubber would improve the collection efficiency of the scrubber system by 20 percent and would provide higher efficiency (98-99 percent) for removal of particulates below 1 micron. By implementing the additional ECT, the emission of uranium-234 would be reduced from 2.98E-2 Ci/y to less than 2.38E-4 Ci/y.

The cost of the control devices already installed in the uranium product preparations C-I wing building is $46,300. The estimated additional cost for adding a high-efficiency scrubber, including demisters, is $15,000 ($11,000 capital plus $4,000 installation). The estimated additional cost for backup HEPA filtration is $9,000. These estimates are based upon the assumption that the existing fan system is capable of maintaining the necessary pressures and flows with the added ECT.

The present annual operating cost of $6,880 is expected to increase 30 percent due to the power necessary to maintain high differential pressures in the venturi and provide flow through both HEPA filters.

If significant structural additions or modifications and other equipment such as special nitric acid scrubbers are necessary for proper operation and maintenance of the expanded air control system, then significant cost increases can be anticipated. In addition to the HEPA filter cost, modifications that include ductwork, blowers, dampers, instrumentation, and electrical work would increase the cost to about $200,000. Engineering costs of about $80,000 and a 35 percent contingency would raise the total project cost to about $400,000. Major structural additions would further increase the cost. Operating costs are expected to double with the implementation of this modified system.

2.5.4.5 Additional Emission Control Technology for the Y-12 Plant (Uranium Fuel Element Fabrication)

The fabrication process is located in the C Wing of Building 9212. Installation of HEPA filters would significantly reduce the amount of particulate uranium-234 emitted from this facility, if the present ventilation system can be modified or expanded to allow installation of HEPA filters downstream of the roughing filters.

HEPA filters collect almost 100 percent of the airborne particulate materials from airstreams containing typical size distributions of suspended materials. It is estimated that 99.95 percent of the materials that pass the roughing filters will be removed by a single pass through HEPA filtration. Based upon this assumption, the installation of HEPA filters would reduce the annual emission of uranium-234 from 1.73E-2 Ci to less than 8.7E-6 Ci.

The uranium fuel element fabrication facility is now served by a large ventilation system which exhausts air at the rate of 23.6 m^3/sec. A similarly sized system which includes the addition of HEPA filters is installed at the Y-12 plant uranium denitrator. The difference in cost between these facilities is $41,000, which is postulated as the cost to add HEPA filters to the fabrication facility. This is based upon the assumption that the air capacity of the system can be maintained by the existing fan system. The cost of additional power requirements and the cost of HEPA filter replacement will double operating costs to about $50,000 per year.

If significant structural additions or modifications such as air coolers are necessary for proper operation and maintenance of the expanded air control system, then significant cost increases can be anticipated. In addition to the HEPA filter cost, modifications that include ductwork, blowers, dampers, instrumentation, and electrical work would increase the costs to about $825,000. Engineering costs of about $200,000 and a 35 percent contingency would raise the total project cost to $1,450,000. Major structural additions would further increase the cost. Operating costs are expected to double with the implementation of this modified system.

2.5.4.6 Additional Emission Control Technology for the Oak Ridge Gaseous Diffusion Plant (Purge Cascade)

The Purge Cascade is part of the Oak Ridge Gaseous Diffusion Plant K-27 process area. All diffusion plant process buildings are three-story, steel frame with 6-mm transite side panels (preformed concrete). The Purge Cascade is intended to separate light gases from UF6 and vent them to the atmosphere through the emission control devices. Emissions from this building represent the largest hypothetical risk from the Oak Ridge Gaseous Diffusion Plant.

Radioactive emissions from the ORGDP Purge Cascade consist mainly of gaseous and particulate uranium and technetium fluorides that pass through existing abatement equipment. A new, low-energy venturi scrubber is planned for installation downstream of the existing spray scrubber to reduce mist carry-over and thus help mitigate equipment corrosion problems. This new scrubber should also reduce airborne emissions somewhat by removing more airborne particulate and droplet materials; however, quantification of the scrubbing action is not precise. It is dependent upon the gaseous solubility and upon the effectiveness of the mixing and impinging action. Addition of this device is estimated to remove about 50 percent of the remaining radioactive emissions.

The cost of the emission control devices now installed at the Purge Cascade is $1.25 million. The estimated additional cost for purchase of a low-energy venturi is $13,000. The added annual operating cost for this installation is estimated to be minor ($1,300) compared to the present annual operating cost of $300,000. Installation costs, which are sensitive to the amount of modification necessary to incorporate the added device, were not estimated.

Table 2.5-6. Summary of capital and operating costs for supplementary controls at the Oak Ridge Reservation.

Facility	Plant	Nuclide	Control Technology	Capital Cost($K)	Operating Cost($K)
ORNL	CRGDF	H-3	Tritiated water/ sieve dryer system	$1,660	$ 0
Y-12	U Prod. Recovery	U-234 U-238	Additional stage HEPA filters and high-energy venturi scrubbers	$800	$29
Y-12	U Prod. Prepara-tion	U-234 U-238	Additional stage HEPA filters and high-energy venturi scrubbers	$400	$13
Y-12	U Fuel Element Fabrication	U-234 U-238	Additional stage HEPA filters and high-energy venturi scrubbers	$1,450	$50
			TOTALS:	$4,310	$92

2.6 SAVANNAH RIVER PLANT

2.6.1 Site Description

The facilities at the Savannah River Plant are used primarily to produce plutonium and tritium, the basic materials required for nuclear weapons. Materials for medical and space applications are also manufactured here, however. The Savannah River Plant is situated along the Savannah River at a site 35 km southeast of Augusta, Georgia. The site covers about 770 km^2.

Operations are grouped into five major areas (designated the 100, 200, 300, 400, and 700 Areas) according to their operational function in the plutonium manufacture/recovery process.

2.6.1.1 100 Area - Nuclear Production Reactors

Three production reactors were in operation. The three reactors produce plutonium and tritium by irradiation of uranium and lithium. Heavy water is used both as a neutron moderator and as a primary coolant. All three reactors have been subsequently shut-down pending the resolution of safety issues and other oporational problems.

2.6.1.2 200 Area - Separations and Waste Management Facilities

Nuclear fuel reprocessing occurs in this area. Plutonium is recovered from irradiated uranium by the PUREX solvent-extraction process. Enriched uranium and plutonium-238 are recovered from other irradiated materials by a solvent-extraction procedure similar to the PUREX process.

2.6.1.3 300 Area - Fuel and Target Fabrication

Tubular fuel and target elements are produced by cladding depleted uranium fuel in aluminum or aluminum/lithium shells. A low-power reactor and a subcritical test reactor are then used to test for assembly defects.

2.6.1.4 400 Area - Heavy Water Production and Recovery

Heavy water is produced from river water by distillation and extraction. Heavy water is also recovered from contaminated reactor coolant. Heavy water is transported from this area to the 100 Area for use in the production reactors.

2.6.1.5 700 Area - The Savannah River Laboratory

Research and process development work is performed at the Savannah River Laboratory. Major activities in this area include fabrication of fuel element and target prototypes; fabrication of radioisotopic sources for medical, space, and industrial applications; thermal and safety studies of reactor operations; and applied research in physics and the environmental sciences.

2.6.2 Major Release Points and Existing Emission Control Technology

Radionuclides are released into the atmosphere from a number of facilities on the SRP site (Ze87, Mo84). Each operating area has one or more discharge stacks that have emission control equipment installed. Monitoring systems record data on a real-time or a near real-time basis. All stack release data are reported annually. The largest quantities of radionuclides are released from the fuel reprocessing areas (F and H Areas). The three production reactor stacks (C, K, and P) release the next largest quantities, followed in descending order of quantities of radionuclide emissions by the heavy water rework plant, the Savannah River Laboratory, and the fuel and target fabrication plant.

Tritium is released from six facilities, with the tritium facilities (232-H, 234-H, 238-H) contributing about 66 percent of the total tritium dose; the reactor areas (105-C, 105-K, and 105-P) contribute about 10 percent, 16 percent, and 7 percent, respectively; the Moderator Rework Unit (420-D) contributes about 0.6 percent; and the Savannah River Laboratory contributes less than 0.01 percent.

Argon-41 is released exclusively at the operating reactors in roughly equal proportions.

Carbon-14 is released from the three operating reactors and from the separations plants in F and H Areas in approximately equal proportions.

In terms of radiation dose to the offsite population, the principal sources are the H Area tritium facilities, followed in order of decreasing contribution by 105-K, 105-C, 105-P, and the F and H Areas separations plants. The contributions from other source locations are negligible (less than 1 percent).

2.6.2.1 200-H Area Tritium Facility Stacks

Releases of tritium from the four stacks associated with the tritium facilities in the 200-H Area constitute the principal sources of radioactive emissions at SRP.

The emission control system uses a long transit volume (the "Serpentine") as a means to capture and hold air flows from process hoods that contain accidental releases of tritium, so that the contained tritium can be removed from the air before discharge to the stack. A nominal air flow continually passes through the Serpentine to the stack line. Air from the process hoods also normally flows to the stack line. When an in-line ion chamber detects a preset level of tritium in the hood outflow, the Serpentine inlet from the process hood is opened, and the hood flow is diverted to the Serpentine. The volume of the holdup line is sufficient to prevent loss of the tritium burst to

the stack. An ionization chamber near the end of the Serpentine detects the tritium concentration as it exits the Serpentine. If the concentration is greater than a preset limit, the volume that exceeds the limit is subsequently diverted and processed through the Hopcalite stripper and zeolite beds to remove the tritium. If the concentration is less than the preset limit, the trapped air volume is discharged to the stack.

The system uses a holdup tank into which batches of inert gases or air from various operational activities are placed for eventual processing through a Hopcalite stripper and two zeolite beds.

The efficiency of the Hopcalite stripper varies with operating conditions (oxidizer bed temperature, oxygen and hydrogen content in the gases to be treated) and can range from a few percent to nearly 100 percent. The actual average efficiency of the strippers at SRP is classified information and cannot be reported here.

2.6.2.2 Production Reactor Area Stacks

Releases of radioactivity into the atmosphere at the three production reactors are the next largest contributors to the offsite population dose resulting from operations at the SRP. Actual releases will vary from reactor to reactor, year by year, depending upon activities.

A ventilation system typical of the production reactors is described below. The filter system consists of inlet prefilters to remove particulates from incoming air, moisture separators to remove entrained moisture droplets from the outgoing air stream, particulate (HEPA) filters to remove particulate material, and charcoal filters to remove iodines. There are no provisions for reducing the emission of tritium, noble gases, or carbon-14.

Monitoring equipment at the 61-m reactor stacks includes continuous Kanne chambers and dehumidifier samplers for monitoring tritium emission, a continuous noble gas monitor utilizing a Ge-Li detector/multichannel analyzer system, a continuous charcoal filter for monitoring radioiodines, and a continuous filter paper sampler for particulate monitoring.

2.6.2.3 200-F and 200-H Area Separation Plants

Releases of radioactivity to the 291-F and 291-H and associated stacks (221-F and 221-H facilities) are principally carbon-14, noble gases, and small amounts of iodine.

Effluent control equipment on the 200-F Area ventilation systems consists principally of particulate filters: fiberglass, HEPA, and sand filters. Silver nitrate beds are used for scrubbing iodine from the dissolver offgas stream.

2.6.3 Basis for the Dose and Risk Assessment

2.6.3.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed below in Table 2.6-1.

Table 2.6-1. Radionuclides released to air during 1986 from Savannah River Plant.

Nuclide	Release Rate (Ci/y)
Am-241	1.9E-4
Ar-41	8.3E+4
C-14	5.6E+1
Ce-141	1.9E-5
Ce-144	1.1E-2
Cm-244	2.8E-5
Co-60	8.0E-6
Cs-134	6.9E-4
Cs-137	3.0E-3
H-3	4.2E+5
I-129	8.8E-2
I-131	2.6E-2
Kr-85	7.1E+5
Kr-85m	2.0E+3
Kr-87	1.4E+3
Kr-88	2.4E+3
Nb-95	9.2E-3
Os-185	1.4E-4
Pu-238	2.0E-3
Pu-239	2.9E-4
Ru-103	3.5E-3
Ru-106	5.9E-2
Se-75	2.1E-5
Sr-89	9.2E-4
Sr-90	1.4E-3
U-234	1.6E-3
U-238	1.6E-3
Xe-131m	3.0E-1
Xe-133	1.1E+4
Xe-135	2.6E+3
Zr-95	4.4E-3

In modeling the site, all releases were assumed to be made from the F-separations area. The releases were aggregated to five stacks: Stack 1 is the 100 Area (60 m): all nuclear production reactors; Stack 2 is the 200 Area (61 m): plutonium and uranium separation; Stack 3 is the 300 Area (10 m): Fuel and Target Fabrication; Stack 4 is the 400 Area (10 m): Heavy Water

Recovery and Production; Stack 5 is not used; and Stack 6 is the 700 Area (50 m): Laboratory. Default particle sizes (1.00 AMAD) and solubility classes were assumed.

2.6.3.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Augusta/Bush, Georgia. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 15,000 m from the assumed release point (Ze87). Food consumption rates appropriate to a rural location were used.

2.6.4 Results of the Dose and Risk Assessment

The major contributors to exposure are tritium (77 percent) and argon-41 (18 percent). The predominant exposure pathways are inhalation, ingestion, and air immersion.

The results of the dose and risk assessment are presented in Tables 2.6-2 through 2.6-4. Table 2.6-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.6-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure as well as estimated deaths per year in the regional population. Table 2.6-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.6-2. Estimated radiation dose rates from the Savannah River Plant.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Remainder	3.2E+0	6.7E+2
Gonads	2.6E+0	5.5E+2
Breast	2.6E+0	5.5E+2
Lungs	2.7E+0	5.6E+2
Red marrow	2.6E+0	5.5E+2

Table 2.6-3. Estimated fatal cancer risks from the Savannah River Plant.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
8E-5	2E-1

Table 2.6-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Savannah River Plant.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	550,000	2E-1
1E-6 - 1E-5	0	0
< 1E-6	0	0
Totals	550,000	2E-1

2.6.5 Supplementary Controls

This section examines specific sources of radionuclide emissions, and existing control systems, discusses current discharge rates, suggests additional control equipment and anticipated reduction in emissions, and estimates costs of the suggested additional equipment (Mo86).

2.6.5.1 Additional Emission Control Technology for the 200-H Area Tritium Facility Stacks

Releases of tritium from the four stacks associated with the Tritium Facilities in the 200-H Area constitute the principal sources of radioactive emissions at SRP. They resulted in a radiation dose to the offsite population of about 67 man-rem during 1981. This dose represents about 57 percent of the total population dose from SRP emissions.

The efficiency of the catalytic oxidizer system might be improved by replacing the Hopcalite (80 percent MnO_2 - 20 percent CuO) beds with a palladium catalyst. Recycling the effluent gases through the stripper combined with hydrogen swapping will also improve the efficiency of the stripper. The SRP staff has estimated that recycling could reduce normal tritium emissions by 25 percent. The cost of the system improvements is estimated to be about $65 million. The system lifetime is estimated to be about 15 years.

2.6.5.2 Additional Emission Control Technology for the Production Reactor Area Stacks

Releases of radioactivity into the atmosphere at the three production reactors are the next largest contributors to the offsite population dose resulting from operations at the SRP. Actual releases vary from reactor to reactor, year by year, depending upon activities.

Tritium emissions from the heavy water moderated reactors could be reduced by (1) replacing tritiated moderator with fresh moderator, (2) minimizing evaporation losses from the moderator, and (3) removing tritium from the existing moderator. While none of these approaches is classified as emission control technology, they are operational in that they attempt to prevent tritium in the ventilation system rather than attempting to remove the tritium from the effluent air stream.

The first approach is not particularly viable. The effect would be only temporary since the tritium levels in the moderator build up with each year of reactor operation.

The second approach is normal operating practice and is already carried out to the extent feasible.

The third approach would use either vapor phase catalytic exchange with cryogenic distillation (CE-CD) or a thermal cycle absorption process (TCAP). These processes have the potential for reducing tritium emissions at the production reactors by about 90 percent once steady-state operation is achieved after about 6 years. SRP staff estimate capital costs for a CE-CD system are to be in the $20-40 million range. Estimated annual operating cost would be in the $1.5 to $2 million range, with an estimated operating life of 30 years. No estimates are currently available for the cost of a TCAP system.

Releases of argon-41 at the production reactors could be reduced by installing a holdup volume into which the air containing the argon-41 (from the annular cavity around the reactor tank) could be routed, thus allowing the radioactivity to decay to insignificant levels. A possible system would use an existing 1,893-m^3 tank in the emergency core cooling system. An air flow of 1.4 to 4.3 m^3/minute into an effective storage volume of 707 m^3 is expected to reduce argon-41 emissions by about 60 percent. The feasibility of utilizing the 1,893-m^3 tank for this purpose is being actively investigated. The capital cost of this proposed system is small, since mostly existing systems and equipment would be used.

No other systems for reducing emissions from the production reactors are presently under consideration.

2.6.5.3 Additional Emission Control Technology for the 200-F and 200-H Area Separation Plants

Releases of radioactivity to the 291-F and 291-H and associated stacks (221-F and 221-H facilities) are principally carbon-14, noble gases, and small amounts of iodine.

Carbon-14, the noble gases, and iodine contribute nearly all of the radiation dose from the separations plants. An absorber system utilizing flaked barium hydroxide octahydrate to form barium carbonate, thus capturing the carbon-14, could be installed. In addition, one of several techniques for capturing the noble gases (particularly krypton-85) could also be installed. These techniques, cryogenic distillation, fluorocarbon absorption, and absorption on mordenite beds, all have decontamination factors of about 100. The iodine removal capability of the existing iodine absorber beds utilizing silver nitrate could be improved if the beds were converted to silver mordenite, moved from the dissolver off-gas system, and installed in the vessel vent system.

SRP staff estimates that an integrated off-gas treatment system utilizing the above techniques would cost about $50 million per plant and would have annual operating costs of about $3 million.

2.7 FEED MATERIALS PRODUCTION CENTER

2.7.1 Description and Existing Controls

2.7.1.1 Site Description

The Feed Material Production Center, located 32 km northwest of Cincinnati, Ohio, produces uranium metal and other materials for DOE facilities. The uranium may be natural, depleted, or enriched with respect to uranium-235.

Raw materials are processed in the following manner. The material is first dissolved in nitric acid and separated by liquid organic extraction. The recovered uranium is reconverted to uranyl nitrate, heated to form uranium trioxide, reduced to uranium dioxide with hydrogen, and reacted with hydrogen fluoride to form uranium tetrafluoride. Purified metal is made by reacting the uranium tetrafluoride with metallic magnesium in a refractory-lined vessel.

The U.S. DOE Effluent Information System Nuclide Database Master List for 1986 reports emissions in 1986 from eight plants at the FMPC (EIS86). These emissions are listed in Table 2.7-1. The emissions are identified as natural uranium in the form of particulates. Each plant at the FMPC has several stacks.

DOE forecasts indicate increased use of the FMPC in support of increased work at other DOE sites (We87, Mo84). The actual magnitude of this increased FMPC production depends on the needs of other DOE sites but could reasonably be expected to double the 1981 production. A corresponding increase in total uranium emissions would therefore be expected, assuming no change in emission control technology.

2.7.1.2 Major Release Points and Existing Emission Control Technology

Emission control technology at the FMPC differs from that of other sites in two major aspects: (1) emissions are essentially all particulates, with natural uranium being the predominant radionuclide; and (2) each plant at the FMPC has multiple stacks, each with its own emission control device and each providing ventilation to a specific area or specific equipment within a given plant.

Chemical and radioactive emissions at the FMPC are controlled by wet scrubbers, bag-type dust collectors, and electrostatic precipitators. The radioactive emissions from the various plants are essentially all particulate emissions. Emissions from Plants 4, 5, and 8 are controlled by the bag-type dust collectors or wet scrubbers.

Bag-type dust collectors are installed on many of the stacks. The dust collectors for these particular stacks have been shown to have total system efficiencies of >99.9 percent over a 2-year period. Most of the material losses occur because of cloth bag ruptures or other malfunctions that allow the dust to bypass the filter.

Stack emissions are constantly sampled using a permanently installed in-stack sampling system. These systems require the collection of about 1 g of material before the collection filters are removed for analysis. A continuous stack monitoring device that will be used in addition to the existing stack samplers has been installed on selected stacks. The results to date indicate that the new stack monitoring device is very sensitive to small quantities of material loss; it has detected minor leaks in dust collection bags that, prior to its installation, had gone undetected until a buildup of material on the stack sampler was found.

2.7.2 Basis for the Dose and Risk Assessment

2.7.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed below in Table 2.7-1.

Table 2.7-1. Radionuclides released to air during 1986 from FMPC.

Nuclide	Release Rate (Ci/y)
U-234	2.0E-2
U-238	2.0E-2

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 Amad) were assumed. The uranium-234 and uranium-238 emissions were assumed to be 1/3 Class D, 1/3 Class W, and 1/3 Class Y.

2.7.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Covington/GTR Cincinnati, Ohio. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 800 m from the assumed release point (We87). Food consumption rates appropriate to an urban location were used.

2.7.3 Results of the Dose and Risk Assessment

The major contributors to exposure are uranium-234 (53 percent) and uranium-238 (48 percent). The predominant exposure pathway is inhalation for uranium-234 and uranium-238.

The results of the dose and risk assessment are presented in Tables 2.7-2 through 2.7-4. Table 2.7-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.7-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.7-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.7-2. Estimated radiation dose rates from FMPC.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	1.9E+1	1.1E+2

Table 2.7-3. Estimated fatal cancer risks from FMPC.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
3E-5	3E-3

Table 2.7-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from FMPC.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	85	2E-5
1E-6 - 1E-5	4,100	1E-4
< 1E-6	3,300,000	3E-3
Totals	3,300,000	3E-3

2.7.4 Supplementary Controls

The U.S. DOE Effluent Information System Nuclide Database Master List for 1986 reports emissions in 1986 from eight plants at the FMPC. Although the major emission sources (stacks) differ each year, Plants 4, 5, and 8 are consistently the greatest source of emissions. The emissions are identified as natural uranium in the form of particulates (EIS86, Mo84).

As mentioned, DOE forecasts indicate increased FMPC production, perhaps as much as double the 1981 production. A corresponding increase in total uranium emissions would therefore be expected, assuming no changes are made in the existing emission controls.

2.7.4.1 Emission Control Technology

The FMPC has over 50 dust collection stacks in either full- or part-time operation. The operating stacks already use very efficient dust collection systems. Additional improvement in reducing operational releases is expected by using Goretex fabric bags rather than wool bags and by using administrative controls in conjunction with the continuous stack monitor. Approximately 20 additional stacks have either been abandoned or placed on standby status. Extensive repair and refurbishment would be needed to return the abandoned and standby dust collection stacks to operation.

However, neither the use of improved fabric bags in the existing baghouses, nor installation of continuous radionuclide stack monitors will insure reductions in uranium particulate emissions at the FMPC. Reductions in emissions to lower levels will require the installation of secondary air cleaning systems on the primary emission sources located in Plants 4, 5, and 8.

2.7.4.1.1 Proposed Emission Control Equipment

It is proposed that HEPA filter systems be installed, in addition to the existing emission control technology, on each of the emission sources from Plants 4, 5, and 8 to reduce their particulate emissions. By definition, each individual HEPA filter must have a minimum particle removal efficiency >99.97 percent for particles 0.3 um diameter.

It has been assumed each system will use redundant HEPAs, each sized for the stated airflow. Filter housings and ductwork are stainless steel. Inlets to the HEPA systems are from existing baghouses or scrubbers.

Placement of the proposed HEPA filter systems depends on: (1) available existing space in Plants 4, 5, and 8; (2) space that could be made available by removal of obsolete and unneeded existing emission controls; and (3) allowable floor or roof live loads at the locations proposed for installation of the HEPA filter systems. The floor loading attributed to the proposed systems is very light and for most of the filter systems would require the addition of only minor secondary steel for support. However, the Plant 5 perimeter appears heavily loaded and may require the additional filter systems to be located outside the existing structure, i.e., a new structure or structures may be required for the filter systems installed in Plant 5.

2.7.4.1.2 Existing and Proposed Stack Monitoring Systems

Radionuclide emissions at the FMPC are essentially all natural uranium in the form of particulates. Emission particle sizes and particle densities have not been reported.

Each stack at the FMPC has an in-stack sampler to determine the quantity of particulates emitted. The sampler collects particulates on a filter paper which is periodically removed and the quantity of uranium collected determined by chemical analyses. Each stack sampler is operated under isokinetic conditions so that total stack emissions can be determined from the quantity of material collected by the stack sampler.

The FMPC has installed new, continuous stack monitors on the following stacks: Plant 4, Stacks G4-2, G4-12, and G4-14; Plant 5, Stacks G5-250, G5-260, and G5-261; Plant 8, Stack G43-27; and Plant 9, Stack G9N1-1039. The continuous stack monitors are pancake-type Geiger-Muller probes installed to monitor the back side of the filter paper used in the in-stack particulate sampler. The continuous stack monitors provide information in real-time on stack emissions. The new monitors can be alarmed for rate-of-rise of radioactivity detected and coupled to automatic shutoff of the process equipment. The rate-of-rise alarm on the continuous stack monitor indicates the failure of the existing primary emission control device (baghouse or

scrubber) to control emissions adequately. The usual cause of alarms for existing baghouses is a break or tear in a bag. The new continuous stack monitors have shown they can detect small leaks in bags that would have gone unnoticed until a buildup of material on the in-stack sampler was observed.

Thus, engineered controls to shut down a given process as a result of using the continuous stack monitor are possible. The FMPC already has administrative controls to shut down processes in order to replace leaking bags in the existing baghouses. However, the reliability of coupling process shut down to the continuous stack monitors is presently unknown. In addition, the FMPC has stated that some processes cannot be shut down during certain operational phases.

The use of the continuous stack monitor is highly recommended as a method to detect leaks in bags or excessive emissions from either the baghouses or scrubbers. However, installation of the continuous stack monitor cannot insure reductions in emissions; secondary particulate emission control devices are also required.

The continuous stack monitors are best used in their existing configuration, i.e., real-time detection of emissions prior to the secondary particulate emission control devices. This configuration allows rapid detection and repair of deficiencies in the primary emission control devices and should reduce the rate of particulate loading on the HEPA filter systems proposed as the secondary emission controls.

A second in-stack sampler (filter paper collector) downstream of the final emission control device is also recommended for uranium inventory control and determination of actual emissions to the environment. If possible, this in-stack sampler should be analyzed to correlate with annual reporting requirements.

2.7.4.2 Estimated Cost for Emission Control Technology

The FMPC has plans to obtain and install 14 additional continuous stack monitors at an estimated cost of $105K ($7.5K per continuous stack monitor). The acquisition of 14 additional continuous stack monitors would allow installation of a continuous stack monitor on each of the stacks that currently do not have one, plus on other selected stacks.

A summary of the cost estimates for the acquisition and installation of the conceptual design HEPA filter systems for each of the stacks is given in Table 2.7-5.

2.7.4.2.1 Effect of Proposed Equipment

Reductions in emissions from the existing emission control devices based on the installation of continuous stack monitors

Table 2.7-5. Cost estimates for acquisition and installation of HEPA filter systems.

Stack No.	HEPA Filter Installation Cost ($ Thousands)	Total Cost[a] ($ Thousands)
G4-2	129.9	324.8
G4-5[a]	57.2	143.0
G4-7	131.0	327.5
G4-14[a]	101.1	252.8
G5-249	131.0	327.5
G5-254	102.1	255.3
G5-256	131.0	327.5
G5-260[a]	102.4	256.0
G5-261[a]	332.2	830.5
Plant 55	76.3	190.8
G43-27[a]	178.7	446.8
8024	73.4	183.5
Rotary Kiln[a]	73.5	183.8
Oxidation #I[a]	57.2	143.0
Total	1677.0	4192.8

(a) Includes A-E fee, allowance for removal of existing systems, and allowance for additional structural supports.

and coupled to either engineered and/or administrative controls are not known at present. The effectiveness of these measures in reducing emissions depends both on the increased sensitivity of detection and the implementation of both effective engineered and administrative controls.

Installation of HEPA filter systems as secondary air cleaning systems is estimated to achieve at least a 90 percent reduction in emissions. The total emissions from the FMPC will vary as a function of its utilization. As stated previously, DOE forecasts increased use of the FMPC in the future. Consequently, the reduced stack emissions that would result from the installation of additional emission control technologies are not absolute values but will reflect the usage of the FMPC.

Some uncertainty results in designating only the scrubbers for the rotary kiln and oxidation furnace #1 in Plant 8 as needing secondary emission control technology because there are a total of four scrubbers in Plant 8. No data were available for the other two scrubbers, and the mass of material emitted from the scrubbers is the sum of the four units.

Architect-Engineer services are typically about 25 percent of all other costs. Thus, total costs for the proposed secondary emission controls may be expected to be about 2-1/2 times greater than the costs shown.

A total secondary emission control cost estimate for the seven stacks is approximately $2.3 million. This estimate is less than half the estimate provided by the FMPC for the six stacks having the greatest emissions in 1986. Direct comparison of the present cost estimates for a specific stack to those of the FMPC is not possible because FMPC provided no details for its estimates. The FMPC has estimated a cost of approximately $14M to install secondary emission controls on all presently operating stacks (Mo86). In either case, the cost estimates are approximate values, subject to revision based on additional information.

2.7.4.2.2 Operation and Maintenance Costs

Addition of continuous stack monitors, as planned by the FMPC, will result in the need for their periodic maintenance. These maintenance needs are not expected to be excessive, although the addition of one full-time-equivalent instrument technician may be required. Regular operations personnel are expected to be responsible for standard operation of the monitors. No unusual operating or maintenance costs are predicted as a result of the installation of additional continuous stack monitors.

HEPA filter replacement costs have been estimated to be $94,000 per year for the seven stacks having the greatest emissions and $111,000 per year for all fourteen stacks. The filter replacement cost estimate is based on an average cost of $350 per filter (stainless steel housing) and the total number of filters to be replaced per year (Mo86).

The FMPC currently has no facilities to process uranium-loaded HEPA filters of the size and quantity proposed in order to recover the uranium. Additional costs for this operation have not been estimated.

If the HEPA filters are discarded, they would have to be disposed of as low specific-activity radioactive waste, i.e., sent to a low-level radioactive waste burial ground. Costs for the packaging, transport, and burial of discarded HEPA filters have not been estimated.

2.8 BROOKHAVEN NATIONAL LABORATORY

2.8.1 Description and Existing Controls

2.8.1.1 Site Description

Studies conducted at Brookhaven Laboratories pertain to the use, environmental effects, and transport of both nuclear and nonnuclear energy materials. Other research programs include applied nuclear studies involving various radioisotopes and investigations of the physical, chemical, and biological effects of radiation. Brookhaven Laboratory is located in the center of Long Island, about 113 km from New York City.

The equipment and facilities used to support the research projects conducted at Brookhaven include several reactors, particle accelerators, and laboratories. Point and area sources of radionuclide releases at Brookhaven include:

- The 40-MW High-Flux Beam Reactor (HFBR)
- The Alternating Gradient Syncrotron, a proton accelerator used in ultra-high energy particle physics research
- The Brookhaven Linac Isotope Production Facility (BLIP)
- The Chemistry Linac Irradiation Facility (CLIF)
- The Brookhaven Medical Research Reactor
- The Van de Graaff accelerator
- Various chemistry and medical research laboratories

Most of the airborne radionuclide emissions from Brookhaven originate from the High-Flux Beam Reactor, the Brookhaven Linac Isotope Production Facility, and the Van de Graaff research generator. Lesser emissions are from the chemistry and medical research centers.

Because very small quantities of radionuclides are released from the Hazardous Waste Management Area, the assessments of exposure and health risk at the Brookhaven site are based on airborne releases from the remaining six effluent stacks. Process descriptions, effluent data, and site information were obtained from reports prepared by Brookhaven Laboratories and DOE studies (Mo84, Mi87b).

2.8.1.2 Major Release Points and Existing Emission Control Technology

In this section, the points of discharge that contribute most to the airborne radionuclide emissions at the BNL site are discussed.

Table 2.8-1. Radionuclide emission points stacks at Brookhaven National Laboratories.

Location	Stack Height (m)
Brookhaven Linac Isotope Production Facil., Bldg-931	18
High-Flux Beam Reactor Hot Laboratory	98
Hazardous Waste Management Area	10
Medical Research Reactor Building-491	45
Chemistry Building-555	Unknown
Medical Research Center	Unknown
Van de Graaff Accelerator Building-901	18

2.8.1.2.1 HFBR Stack

The principal radionuclides discharged from the HFBR stack are tritium (from the HFBR) and xenon-127 and small amounts of unidentified radionuclides that emit beta and gamma radiation (from the Hot Laboratory). Tritium is the most prevalent radionuclide discharged.

The HFBR facility (Building 750) is ventilated by about 566 m^3/min of air, all of which is filtered through absolute HEPA filters to remove particulates and radioactivity before being discharged from the 98-m stack. In addition, procedural and administrative controls have been implemented to detect tritium, prevent its leakage, and reduce the release of tritiated water vapor from the HFBR stack. Since 1977, yearly replacement of a portion of the heavy water (moderator and coolant) has reduced the annual tritiated water vapor released from the HFBR by approximately 50 percent.

The hot area of the Hot Laboratory (Building 801) consists of five semihot cells, three chemical processing hot cells, and three high-level hot cells for handling multicurie amounts of radioactive materials. Each cell is equipped with its own roughing exhaust air filter, as well as a backup HEPA filter in the exhaust line leading to the stack. The three chemical process cells have a separate exhaust air system that uses a NaOH scrubber and charcoal filter to remove radioiodines. The small amount of xenon-127 released is diluted after release from the stack. All effluents from the Hot Laboratory are exhausted to the 98-m HFBR stack.

2.8.1.2.2 Brookhaven LINAC Isotope Production Facility

The targets used for the production of desired radionuclides in the BLIP facility are sealed so that no radioactivity can escape from them during normal operation. However, oxygen-15 and tritium are formed by the incident protons in the target cooling

water. Larger release rates of oxygen-15 in relation to the other gases result because it is swept out with absorbed oxygen in the cooling water. The absorbed oxygen is formed by the radiolytic formation of stable oxygen. The airborne effluents from the BLIP facility undergo HEPA filtration to remove any particulates prior to monitoring and release from an 18-m stack. The oxygen-15 and tritium currently receive no treatment prior to discharge from the stack (Mo86).

2.8.1.2.3 Brookhaven Medical Research Reactor

The principal radioactive gas discharged during routine operations of the BMRR is 110-minute half-life argon-41, which is produced in the cooling air in the reactor's graphite reflectors. At a full power level of 3 MW, a release rate of about 3 Ci/hr has been established by direct measurements. The operation of the BMRR is administratively controlled to a daily limit of 24 MWhr. Currently, it is operated intermittently for short-lived activation irradiation. The BMRR is enclosed in a containment building that is maintained under negative pressure to prevent inadvertent releases to the outside. Air flow from the building is passed through HEPA and charcoal filters to remove particulates before being vented to the atmosphere via a 45-m stack.

2.8.1.2.4 Research Van de Graaff Accelerator

The principal radionuclide discharged to the atmosphere from the Research Van de Graaff Accelerator is tritium. Currently, about 95 percent of the release is in gaseous form and about 5 percent is tritiated water vapor. The air control system in this facility is designed to function as a closed system. During normal operation, a low-pressure pump is used to maintain negative pressure on the system. The output of this pump is routed through a catalytic recombiner where the tritium gas is converted to tritiated water vapor which is passed through a dessicant for removal. Spent dessicants are periodically removed and transported offsite for disposal with other low-level solid waste. When the accelerator is shut down for maintenance, the negative pressure is removed and air at atmospheric pressure is allowed to fill the system. Upon completion of maintenance, the system is pumped down to a negative pressure. During these times, the flow exceeds the capacity of the recombiner and the excess flows are routed directly to the stack via a by-pass line. When tritium ions are being accelerated, about 200 Ci/month of tritium gas is used. Of the total tritium used, about 50 percent is trapped by the dessicant and about 50 percent is released from the 18-m stack attached to Building 901.

2.8.2 Basis for the Dose and Risk Assessment

2.8.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.8-2.

Table 2.8-2. Radionuclides released to air during 1986 from Brookhaven National Laboratory.

Nuclide	Release Rate (Ci/y)
Ar-41	1.2E+3
Ba-133	2.7E-6
Be-7	1.8E-6
Br-82	7.8E-3
C-14	7.7E-4
Co-57	2.2E-5
Cr-51	1.1E-4
Fe-55	5.1E-3
H-3	1.6E+2
Hg-203	1.2E-6
I-125	5.2E-4
I-126	3.2E-4
I-131	5.1E-4
I-133	1.8E-4
Mn-54	1.0E-5
O-15	1.5E+2
P-32	2.5E-4
Ru-103	1.2E-5
S-35	5.7E-4
Sb-122	3.0E-7
Se-75	2.0E-5
Sn-113	2.0E-4
Sn-117m	4.2E-5
Tc-99	1.0E-4
Tc-99m	2.0E-4
Tl-201	2.1E-5
Xe-125	8.8E-5
Xe-127	5.7E-4
Xe-131m	6.8E-6
Zn-65	1.3E-6

In modeling the site, all releases were aggregated to six stacks: Stack 1 is Chemistry Building #555, with a stack height of 17 m; Stack 2 is the Van De Graaff Building 901, with a stack height of 18 m; Stack 3 is the HFBR Hot Lab, with a stack height of 98 m; Stack 4 is the Hazardous Waste Management Area, with a stack height of 10 m; Stack 5 is the MRC Buildings 490 and 491, with a stack height of 14 m; and Stack 6 is the BLIP Building

931, with a stack height of 18 m. Default particle sizes (1.00 AMAD) and solubility classes were assumed.

2.8.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Lawrence, New York. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 750 m from the assumed release point. Food consumption rates appropriate to an urban location were used.

2.8.3 Results of the Dose and Risk Assessment

The major contributor to exposure is argon-41 (94 percent). The predominant exposure pathway is air immersion.

The results of the dose and risk assessment are presented in Tables 2.8-3 through 2.8-5. Table 2.8-3 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.8-4 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.8-5 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.8-3. Estimated radiation dose rates from the Brookhaven National Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Gonads	8.0E-1	3.8E+0
Remainder	6.2E-1	3.0E+0
Breast	7.2E-1	3.4E+0
Red marrow	6.2E-1	2.9E+0
Lungs	6.1E-1	2.9E+0

Table 2.8-4. Estimated fatal cancer risks from the Brookhaven National Laboratories.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
2E-5	1E-3

Table 2.8-5. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Brookhaven National Laboratories.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	800	2E-4
1E-6 - 1E-5	1,800	6E-5
< 1E-6	5,200,000	9E-4
Totals	5,200,000	1E-3

2.8.4 Supplementary Controls

Ninety-four percent of the risk estimated for BNL results from the release of Argon-41 from the BMRR. Argon-41 emissions could be reduced by the addition of a hold-up tank to allow the argon-41 to decay.

2.9 MOUND FACILITY

2.9.1 Description and Existing Controls

2.9.1.1 Site and Release Point Description

The Mound Facility, located in Miamisburg, Ohio, about 16 km southwest of Dayton, Ohio, has a variety of active programs. These include research and development, processing of solid wastes for tritium recovery, fabrication and testing of weapons components, production of stable isotopes for the market, and manufacture of radioisotopic heat sources for military and aerospace applications.

The principal emissions of tritium and plutonium emanate from nine buildings, designated as HH, SW, H, PP, R, SM, WD, WDA, and 41. Buildings HH and SW, which contain the tritium recovery and reprocessing facilities, are the sole release points of tritium. Plutonium is released from the other facilities as a result of heat source production and waste disposal operations.

2.9.2 Basis for the Dose and Risk Assessment

2.9.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.9-1.

Table 2.9-1. Radionuclides released to air during 1986 from Mound Facility.

Nuclide	Release Rate (Ci/y)
H-3	3.6E+3
Pu-238	5.8E-6
Pu-239	1.4E-7
U-234	7.5E-8
U-238	8.4E-8

In modeling the site, all releases were assumed to be made from a single 61-m stack. Default particle sizes (1.00 AMAD) and solubility classes were assumed.

2.9.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Dayton, Ohio. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 1,500 m from the assumed release point (Mi87b). Food consumption rates appropriate to an urban location were used.

2.9.3 Results of the Dose and Risk Assessment

The major contributor to exposure is tritium (98 percent). The predominant exposure pathway is inhalation.

The results of the dose and risk assessment are presented in Tables 2.9-2 through 2.9-4. Table 2.9-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.9-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.9-4 presents the estimated distribution of fatal cancer risk to the regional population.

2.10 IDAHO NATIONAL ENGINEERING LABORATORY

2.10.1 Description and Existing Controls

2.10.1.1 Site Description

The Idaho National Engineering Laboratory is a reactor testing facility in southeastern Idaho, about 56 km west of Idaho Falls. The following four contractors operate facilities here: EGho, Inc.; Allied Chemical Corporation; Argonne West Laboratory; and Westinghouse Electric Corporation.

Table 2.9-2. Estimated radiation dose rates from the Mound Facility.

Organ	Nearby Individuals (mrem/y)	Regional Population person-rem/y)
Remainder	4.1E-2	3.3E+0
Gonads	3.7E-2	3.0E+0
Breast	3.7E-2	3.0E+0
Lungs	3.8E-2	3.0E+0
Red marrow	3.7E-2	3.0E+0

Table 2.9-3. Estimated fatal cancer risks from the Mound Facility.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
1E-6	3E-3

Table 2.9-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Mound Facility.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	1,000	2E-5
< 1E-6	2,900,000	3E-3
Totals	2,900,000	3E-3

EGc., operates several test reactors. These reactors provide operating information for the development of reactor safety programs, for determination of the performance of reactor materials and equipment, and occasionally, for use in research performed by private organizations. Other activities include disassembly and reassembly of large radioactive reactor components, preparation of test specimens for use in various operating reactors, and waste handling.

Fuel processing is the major operation that Allied Chemical conducts at this site. Its Idaho Chemical Processing Plant stores irradiated fuel and reprocessed fuel and converts high-level radioactive liquid waste to solid form.

Westinghouse operates the Naval Reactor Facility at the Idaho Laboratory. This facility serves as a testing area for prototype naval reactors and as a disassembly and inspection area for expended reactor cores.

Argonne West operates the experimental Breeder Reactor, the transient Reactor Test Facility, and the Zero Power Physics Reactor.

2.10.1.2 Major Release Points and Existing Emission Control Technology

2.10.1.2.1 Advanced Test Reactor (ATR)

The ATR has an operational thermal-power level rating of 150 MW. It is designed for use in developing advanced cores and fuel system materials for commercial power programs. The ATR is a light-water-moderated and cooled system that employs the flux concentration principle (flux traps) to achieve higher neutron flux levels.

Ventilation air from the ATR is discharged from a 76-m stack with no waste treatment system employed. The stack is monitored on a continuous basis for particle and gaseous activity. Noble gases, such as argon, krypton, and xenon, are released. The airflow rate of the stack is 1,275 m^3/min.

2.10.1.2.2 Idaho Chemical Processing Plant (ICPP)

The ICPP is used to process highly enriched-irradiated nuclear reactor fuel elements in order to recover uranium. Fuel elements from INEL reactors (test and research), other research reactors (domestic and foreign), and U.S. Navy ship propulsion reactors have been reprocessed. Airborne emissions from the ICPP are largely attributable to off-gases from the process dissolvers, process vessels, analytical facilities, sample stations, waste solvent burner, New Waste Calcining Facility (NWCF), and ventilation air. The New Waste Calcining Facility is used to convert radioactive liquid waste from the ICPP to a solid, using a fluidized bed calcination process.

The atmospheric protection system (APS) serves as a final cleanup facility for most ventilation systems and the process off-gas systems within the ICPP. The APS is divided into three treatment sections: (1) ventilation air treatment, (2) nitrogen oxide-bearing off-gas treatment, and (3) hydrogen-rich off-gas treatment.

The vessel off-gas treatment section of the APS facilitates treatment of the process off-gases from: (1) continuous process modification dissolver off-gas (CPMDOG), (2) vessel off-gas, and (3) the New Waste Calcining Facility. This section of the APS consists of a condenser, demister, superheater, prefilter, final filter, and blowers. The system is constructed of stainless steel for acid resistance.

A single-story 15.8 x 6.1 m building attached to the southeast corner of the HEPA building, CPP-649, contains the APS cleanup system and blowers for the VOG process off-gases. The cleanup portion of the system (condenser, demister, superheater, and prefilter) is in the east part of the building. Some valves that may require opening or closing during operation are equipped with reach rods that penetrate the shielding wall.

The demister consists of two 10-cm-thick stainless steel mesh elements contained in a stainless steel chamber.

The prefilter is constructed of five separate fiberglass beds supported on stainless steel screens. Contained in a 3.7 x 2.1 x 4.0 m stainless steel housing, the prefilter has a water line for flushing the filter medium. The prefilter can be bypassed during flushing. The flush water drains to the process equipment waste (PEW) evaporator feed tank. The three HEPA filters are housed in caissons equipped with dampers for individual filter isolation. The HEPA filters are made of acid- and moisture-resistant materials. The HEPA filters are equipped with knife-edge seals to prevent leakage.

Two stainless steel blowers exhaust the VOG streams to the main stack. Only one blower is required for normal operation. The operating blower is switched automatically to emergency power during commercial power outages; the standby blower starts automatically on failure of the operating blower to maintain necessary vacuum. The blowers are provided with automatic air operated valves to isolate the unit not in operation.

The ventilation exhaust filter system, a portion of the APS, consists of a deep-bed fiberglass prefilter in series with standard HEPA filters. The prefilter is located in an underground reinforced concrete vault (CPP-756), measuring 12.2 x 27.4 x 4.3 m. The vault includes a system for backwashing the prefilter medium. Over-temperature protection for the filters is provided by a fog-spray system located upstream of the prefilter. This system actuates on high-gas temperature in the duct and cools the gas and protects the filters from an in-cell fire. A bypass duct is provided around the prefilter for use during washing of the filter medium.

The ventilation air ducts from the various buildings join before entering the prefilter distribution plenum. The distribution plenum extends the full length of the west side of the vault and distributes air, via flow slots, into each of four bays.

The floor of the underground vault is sloped to the north; four troughs drain condensate or flush water to the north edge of the vault. From there, another trough carries the water to a 1,893-l capacity collection sump located in the northeast corner of the vault. The sump is equipped with a high-level alarm and a sampler. From the sump, the liquid and associated solids are jetted to the PEW evaporator feed tank, WL-102.

The south wall of the vault has six viewing ports for inspection of the vault and filters. No lights are provided in the vault; portable lighting is used when needed.

The roof of the vault is 0.3 m below grade and covered with about 0.6 m of earth for radiation shielding. The roof and earth cover are sloped to allow proper drainage, and the vault is of leaktight construction. The cracks between the removable interlocking concrete blocks are caulked, and a butyl rubber membrane covers the entire roof of the vault. Insulation board overlays the membrane to prevent damage by the soil.

The prefilter has an area of 279 m^2 and has a maximum flow rate of 4,245 m^3/min. The prefilter is designed for gas upflow through five layers of varying density, separately supported, packed fiberglass. The five individual layers are separated and supported by stainless steel wire screens. The screens are mounted on Amercoat-painted carbon steel frames and wired to support pipes spaced at 0.9-m intervals. The prefilter frame is attached to Unistrut embedded in the concrete walls; voids in the Unistrut and other openings are caulked with fiberglass to prevent bypassing of the filter medium.

Water spray systems are provided to flush particulates from the fiberglass deep-bed prefilters if the pressure drop becomes excessive. There are three spray lines, located at different elevations, to provide thorough washing of the filter medium.

Each of the three spray lines consists of five 1.2-cm diameter Type 304 stainless steel pipes; the bottom line is equipped with spray nozzles directed upward and the two upper lines have holes drilled in the lower portion of the pipes to supply flush water to the filter. To reduce water supply and removal requirements for flushing the ventilation air prefilter, flushing is done in sections. The spray system piping is stubbed off outside the ventilation air prefilter vault for later connection to a water supply, if required. The fiberglass deep-bed prefilters will not require replacement during the design lifetime of 20 years (from 1975). However, with the estimated dust loading in the ventilation air, the prefilter should last about 75 years without flushing or replacement.

Ventilation air from the prefilter is discharged through a concrete duct to the HEPA filters located in a building adjacent to the prefilter vault. The two-story reinforced concrete structure measures 23.5 x 10.1 and is 7.9 m high. The first story of the structure begins 2.4 m below grade.

The HEPA filters are 9.4 x 9.4 x 4.3 cm units, each rated at 42.5 m^3/min, with an initial pressure drop of 2.5 cm of water. The filters are housed in caissons for ease of maintenance and filter replacement.

From the HEPA filters, the ventilation air flows through three ventilation fans and is exhausted to the stack. The ventilation fans are direct drive and installed in parallel to provide the motive force for discharging the ventilation air to the stack. The fans are housed in a 6.6 x 14.6 m addition on the east side of the existing fan building (CPP-605). The fans are of carbon steel construction with backward airfoil blades. During normal operation, one or two of the three fans is operated on commercial power. If the operating fan fails during normal operation, the second and third fans can be started manually on commercial power. Automatic switching of an operating fan to emergency power, during commercial power outages, is provided by manual preselection. Each fan is provided with a damper that closes automatically if the fan stops. The dampers can be opened either with a wrench or via a pressurized N2 system if the need arises.

2.10.2 Basis for the Dose and Risk Assessment

2.10.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.10-1.

In modeling the site, all releases were assumed to be made from the ICPP, since this is the major source of uranium. The releases were assumed from a m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class W for antimony-125) were assumed.

2.10.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Pocatello, Idaho. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 15,000 m from the assumed release point (Ho87). Food consumption rates appropriate to a rural location were used.

2.10.3 Results of the Dose and Risk Assessment

The major contributors to exposure are argon-41 (51 percent), antimony-125 (32 percent), and krypton-88 (8 percent). The predominant exposure pathways are air immersion for argon-41 and ground surface for antimony-125 and krypton-88.

Table 2.10-1. Radionuclides released to air during 1986 from all Idaho Facilities.

Nuclide	Release Rate (Ci/y)
Ar-41	1.9E+3
Ba-139	7.5E+0
Ba-140	1.8E-6
Br-82	1.0E-2
C-14	3.3E-1
Co-60	4.4E-4
Cs-134	1.0E-4
Cs-137	2.4E-3
Cs-138	9.4E-1
Gd-153	9.8E-6
H-3	3.6E+1
Hg-203	1.4E-4
I-129	1.8E-1
I-131	7.4E-4
Kr-85	1.1E+4
Kr-85m	7.1E+1
Kr-87	1.5E+2
Kr-88	1.6E+2
La-140	1.8E-6
Mn-54	8.7E-5
Nb-95	5.2E-7
Pu-238	1.6E-5
Ru-103	2.0E-7
Sb-125	9.3E-1
Se-75	1.1E-4
Sr-85	3.2E-8
Sr-90	1.9E-6
Te-132	6.0E-8
Xe-133	5.2E+2
Xe-135	4.1E+2
Xe-135m	3.2E+0
Xe-138	4.1E+2
Y-90	3.1E-8

The results of the dose and risk assessment are presented in Tables 2.10-2 through 2.10-4. Table 2.10-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.10-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.10-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.10-2. Estimated radiation dose rates from the Idaho National Engineering Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Gonads	2.9E-2	7.3E-2
Remainder	2.3E-2	6.3E-2
Breast	2.7E-2	6.8E-2
Lungs	2.4E-2	6.1E-2
Red marrow	2.3E-2	5.7E-2

Table 2.10-3. Estimated fatal cancer risks from the Idaho National Engineering Laboratory.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
6E-7	2E-5

Table 2.10-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from INEL facilities.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	100,000	2E-5
Totals	100,000	2E-5

2.11 LAWRENCE BERKELEY LABORATORY

2.11.1 Description and Existing Controls

2.11.1.1 Site Description

Lawrence Berkeley Laboratory (LBL) is situated upon a hillside above the main campus of the University of California, Berkeley. The 130-acre site is located on the west-facing slope

of the Berkeley Hills, at elevations ranging from 500 to 1,500 feet above sea level. LBL is located in an urban environment on land owned by the University. The LBL site is bordered on the north by predominately single family homes and on the west by multiunit dwellings, student residence halls, and commercial districts. The population within an 80-km radius of the Laboratory is approximately 5.2 million (1980 census).

The Laboratory's activities are located both onsite and offsite. There are 67 buildings on the LBL hillside site, plus additional facilities located on the University campus, notably the Donner Laboratory of Biology and Medicine and the Melvin Calvin Laboratory. The onsite space consists of 1,350,000 gross square feet (gsf) in about 60 buildings: 1,307,000 in DOE buildings and trailers and 43,000 in University-owned buildings.

These facilities include four large accelerators, several small accelerators, several radiochemical laboratories, and the Tritium Labeling Laboratory. The large accelerators include the Bevatron, the Super HILAC, the 224-cm Sector-Focused Cyclotron, and the 467-cm Cyclotron.

The tritium facility was designed to accommodate kilocurie quantities of tritium as a labeling agent for chemical and biomedical research. Radiochemical and radiobiological studies in many laboratories typically use millicurie quantities of various radionuclides.

2.11.1.2 Major Release Points and Existing Emission Control Technology

Each laboratory box exhaust system includes a group of HEPA filters and/or gas traps. The tritium facility has a tritium recovery system in which unused tritium gas is circulated over hot copper oxide and the resultant water is trapped in a liquid nitrogen dewar, drained from the system, and packaged for disposal. This recovery system can be isolated from the labeling and storage system, and the tritium can be circulated continuously in a closed loop until the tritium concentration has dropped to an acceptable level for discharge to the atmosphere via the laboratory exhaust manifold. Silica gel traps are used to reduce the level of tritium discharged.

The purge ventilation system of the LBL tritium facility consists of an air evacuation system that draws air through inside filters into a vent pipe to the outside of the facility where it then undergoes mechanical forcing. This forcing vents the air through a vertical exhaust stack elevated 9 m above a hill directly behind the facility, giving an effective stack height of 18.3 m.

2.11.2 Basis for the Dose and Risk Assessment

2.11.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.11-1.

Table 2.11-1. Radionuclides released to air during 1986 from Lawrence Berkeley Laboratory.

Nuclide	Release Rate (Ci/y)
H-3	7.6E+1
I-125	3.7E-3
I-131	1.2E-3
Pu-239	7.4E-9
Sr-90	5.8E-5

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes were assumed.

2.11.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Oakland, California. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 250 m from the assumed release point (Sc87). Food consumption rates appropriate to an urban location were used.

2.11.3 Results of the Dose and Risk Assessment

The major contributor to exposure is tritium (90 percent). The predominant exposure pathway is inhalation.

The results of the dose and risk assessment are presented in Tables 2.11-2 through 2.11-4. Table 2.11-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.11-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.11-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.11-2. Estimated radiation dose rates from the Lawrence Berkeley Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Remainder	1.9E-2	7.8E-1
Gonads	1.8E-2	7.0E-1
Red marrow	2.5E-2	1.0E+0
Breast	1.8E-2	7.0E-1
Lungs	1.8E-2	7.0E-1

Table 2.11-3. Estimated fatal cancer risks from the Lawrence Berkeley Laboratory.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/year
5E-7	3E-4

Table 2.11-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Lawrence Berkeley Laboratory.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	5,000,000	3E-4
Totals	5,000,000	3E-4

2.12 PADUCAH GASEOUS DIFFUSION PLANT

2.12.1 Site Description

The DOE operation at the Paducah Gaseous Diffusion Plant consists of a uranium enrichment facility and a uranium hexafluoride manufacturing complex. The plant is located 6 km south of the Ohio River in McCrasken County, Kentucky.

The primary activity at this site is the diffusion cascade for the enrichment of uranium in fissionable uranium-235 content. All stages of the enrichment cascade take place in five buildings on the site. The manufacturing facility produces uranium hexafluoride from uranium oxide feedstocks.

2.12.2 Basis for the Dose and Risk Assessment

2.12.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.12-1.

Table 2.12-1. Radionuclides released to air during 1986 from Paducah Gaseous Diffusion Plant.

Nuclide	Release Rate (Ci/y)
Tc-99	8.8E -3
U-234	1.8E -4
U-238	1.8E -4

In modeling the site, all releases were assumed to be made from a 10-m stack, with a flow of 200 cfm. Default particle sizes (1.00 AMAD) and solubility classes (Class Y for uranium-234 and uranium-238) were assumed.

2.12.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Paducah/Barkley, Kentucky. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 1,500 m from the assumed release point (Mo86). Rural food consumption rates were used.

2.12.3 Results of the Dose and Risk Assessment

The major contributors to exposure are uranium-234 and uranium-238 (99 percent). The predominant exposure pathway for both is inhalation.

The results of the dose and risk assessment are presented in Tables 2.12-2 through 2.12-4. Table 2.12-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.12-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.12-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.12-2. Estimated radiation dose rates from the Paducah Gaseous Diffusion Plant.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	2.5E-1	3.1E-1

Table 2.12-3. Estimated fatal cancer risks from the Paducah Gaseous Diffusion Plant.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
4E-7	1E-5

Table 2.12-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Paducah Gaseous Diffusion Plant.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	500,000	1E-5
Totals	500,000	1E-5

2.13 LAWRENCE LIVERMORE LABORATORY

2.13.1 Site Description

The Lawrence Livermore National Laboratory, situated 64 km east of San Francisco, California, is primarily a nuclear weapons research and development center. Other activities, however, include research programs in laser isotope separation, laser fusion, magnetic fusion, biomedical studies, and nonnuclear energy.

Two accelerators, the Insulated Core Transfer Accelerator and the Electron Positron Linear Accelerator, are used in support

of the fusion and neutron physics research programs. The Light Isotope Handling Facility supports research in the area of light isotopes. The remaining facilities at this site deal with equipment decontamination and waste disposal.

2.13.2 Basis for the Dose and Risk Assessment

2.13.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.13-1.

Table 2.13-1. Radionuclides released to air during 1986 from Lawrence Livermore Laboratory/Sandia Livermore.

Nuclide	Release Rate (Ci/y)
H-3	1.8E+3
N-13	9.0E+1
O-15	9.0E+1
Pu-239	7.0E-9
Sr-90	1.3E-7

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes were assumed.

2.13.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Fairfield/Travis, California. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 3,500 m from the assumed release point (Mo86). Food consumption rates appropriate to a rural location were used.

2.13.3 Results of the Dose and Risk Assessment

The major contributor to exposure is tritium (98 percent). The predominant exposure pathway is inhalation.

The results of the dose and risk assessment are presented in Tables 2.13-2 through 2.13-4. Table 2.13-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.13-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.13-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.13-2. Estimated radiation dose rates from Lawrence Livermore Laboratory/Sandia Livermore.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Remainder	1.1E-2	4.2E+0
Gonads	1.1E-2	3.7E+0
Breast	1.1E-2	3.7E+0
Lungs	1.1E-2	3.8E+0
Red marrow	1.1E-2	3.7E+0

Table 2.13-3. Estimated fatal cancer risks from Lawrence Livermore Laboratory/Sandia Livermore.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
3E-7	1E-3

Table 2.13-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Lawrence Livermore Laboratory/Sandia Livermore.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	5,300,000	1E-3
Totals	5,300,000	1E-3

2.14 PORTSMOUTH GASEOUS DIFFUSION PLANT

2.14.1 Description and Existing Controls

2.14.1.1 Site and Release Point Description

The Portsmouth Gaseous Diffusion Plant, situated in Pike County, Ohio, about 1.6 km east of the Scioto River, is operated

by Goodyear Atomic Corporation. The primary activity at this site is the diffusion cascade for the enrichment of uranium in fissionable uranium-235 content. All stages of the enrichment cascade take place in five buildings on the site. The manufacturing facility produces uranium hexafluoride from uranium oxide feedstocks.

The most significant release point, which accounts for about 84 percent of total emissions, is the X-326 Top Purge Vent.

The DOE Effluent Information System Report for 1986 identifies the following major specific sources for the Portsmouth Plant: the X-326 Building Top and Side Purge Vent, the X-330 Building Cold Recovery Facility, and the X-333 Building Cold Recovery Facility (EIS86).

The radioisotopes in these releases are uranium and its daughters plus technetium-99, a long-lived fission product. The technetium-99 results from introducing uranium feed from reprocessed irradiated nuclear reactor fuel.

2.14.1.2 Emission Control Technology

The main control technologies presently used at Portsmouth are:

- o Cold trapping (the UF6 is removed by freezing)
- o Sodium fluoride absorption
- o Activated alumina absorption

These methods are primarily useful in preventing the release of uranium. They are also effective on uranium decay daughters and on the fission-product isotope technetium-99.

The X-326 Purge Vent is the major source of radionuclide emissions to the atmosphere at Portsmouth. The existing control device is the purge cascade itself, which removes the bulk of the UF_6. The remaining light gases are sent through an alumina trap and diluted with an air jet exhauster before venting.

There are four purge vents. Each vent is 23 m high, 47 cm apart. The diameter of each vent is 10 cm. Each vent has a flow rate of 4.72 x 10-2 m^3/s at ambient temperature.

2.14.2 Basis for the Dose and Risk Assessment

2.14.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.14-1.

Table 2.14-1. Radionuclides released to air during 1986 from the Portsmouth Gaseous Diffusion Plant.

Nuclide	Release Rate (Ci/y)
Pa-234m	1.4E-2
Tc-99	1.2E-1
Th-234	1.4E-2
U-234	2.3E-2
U-235	1.2E-3
U-236	3.4E-5
U-238	1.4E-2

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) were assumed, and the uranium was assumed to have a D solubility class.

2.14.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Huntington, West Virginia. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 1,500 m from the assumed release point (Oa87a). Food consumption rates appropriate to a rural location were used.

2.14.3 Results of the Dose and Risk Assessment

The major contributors to exposure are uranium-234 and uranium-238 (96 percent). The predominant exposure pathway for both is inhalation.

The results of the dose and risk assessment are presented in Tables 2.14-2 through 2.14-4. Table 2.14-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.14-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.14-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.14-2. Estimated radiation dose rates from the Portsmouth Gaseous Diffusion Plant.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Endosteum	3.4E-1	5.7E+0
Remainder	3.0E-2	7.7E-1
Red marrow	2.3E-2	4.0E-1

Table 2.14-3. Estimated fatal cancer risks from the Portsmouth Gaseous Diffusion Plant.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
2E-7	9E-5

Table 2.14-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Portsmouth Gaseous Diffusion Plant.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	620,000	9E-5
Totals	620,000	9E-5

2.15 ARGONNE NATIONAL LABORATORY

2.15.1 Site Description

Argonne National Laboratory is an energy research and development center that performs investigations in basic physics, chemistry, materials science, the environmental sciences, and biomedicine. Argonne also plays an important role as a nuclear and nonnuclear engineering center. The laboratory complex is located in Dupage County, Illinois, 43 km southwest of Chicago.

Argonne National Laboratory has the following principal nuclear facilities:

(1) 10- and 200-kW research reactors
(2) A critical assembly reactor
(3) A 60-inch cyclotron
(4) A prototype, superconducting, heavy ion linear accelerator
(5) Van de Graaff and Dynamitron-type charged-particle accelerators
(6) A high-energy neutron source
(7) Cobalt-60 irradiation sources
(8) Laboratories engaged in work with multicurie quantities of the actinide elements

The 200-kW JANUS research reactor and the laboratory handling area (hot cells) are the main sources of radionuclide releases from the Argonne complex.

Specific details of the site activities and emissions are available from annual emission reports prepared by the laboratory, the DOE Effluent Information System, and environmental monitoring studies conducted by DOE (Mo84, EPA84, EIS86).

2.15.2 Basis for the Dose and Risk Assessment

2.15.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.15-1.

Table 2.15-1. Radionuclides released to air during 1986 from Argonne National Laboratory.

Nuclide	Release Rate (Ci/y)
Ar-41	1.4E+0
C-11	9.0E+1
Cs-134	2.0E-7
Cs-137	4.9E-7
H-3	5.0E+1
I-129	1.6E-5
I-131	1.5E-6
Kr-85	1.7E+0
Nb-95	1.5E-8
Pu-239	5.6E-9
Rn-220	7.0E+3
Sb-125	3.4E-5
Zr-95	7.5E-9

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class D for carbon-11) were assumed.

2.15.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Midway Airport, Illinois. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data.Nearby individuals were located 750 m from the assumed release point. Urban food consumption rates were used.

2.15.3 Results of the Dose and Risk Assessment

The major contributors to exposure are carbon-11 and tritium. The predominant exposure pathway is inhalation for carbon-11 and air immersion for tritium.

The results of the dose and risk assessment are presented in Tables 2.15-2 through 2.15-4. Table 2.15-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.15-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.15-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.15-2. Estimated radiation dose rates from the Argonne National Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	3.1E-2	2.5E-1
Remainder	2.7E-3	2.1E-1

Table 2.15-3. Estimated fatal cancer risks from the Argonne National Laboratory.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
1E-7	8E-5

Table 2.15-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Argonne National Laboratory.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	7,900,000	8E-5
Totals	7,900,000	8E-5

2.16 PINELLAS PLANT

2.16.1 Site Description

The Pinellas Plant, located 10 km northwest of St. Petersburg, Florida, is a major facility engaged in the production of nuclear weapons. Although descriptions of the principal operations resulting in atmospheric releases of radioactive materials could not be found in the literature, they are neutron generator development and production, testing, and laboratory operations. Small, sealed plutonium capsules are used as heat sources in the manufacture of radioisotopic thermoelectric generators. The heat sources are triple-encapsulated to prevent release of plutonium to the atmosphere.

Emissions of radionuclides were identified from three sources: the main stack, laboratory stack, and building stack.

2.16.2 Basis for the Dose and Risk Assessment

2.16.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.16-1.

Table 2.16-1. Radionuclides released to air during 1986 from Pinellas Plant.

Nuclide	Release Rate (Ci/y)
H-3	1.9E+2
Kr-85	4.6E+0

In modeling the site, all releases were assumed to be made from a 10-m stack.

2.16.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Tampa, Florida. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 1,500 m from the assumed release point. Food consumption rates appropriate to a rural location were used.

2.16.3 Results of the Dose and Risk Assessment

The major contributor to exposure is tritium (100 percent). The predominant exposure pathway is inhalation.

The results of the dose and risk assessment are presented in Tables 2.16-2 through 2.16-4. Table 2.16-2 presents the dosesreceived by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.16-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.16-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.16-2. Estimated radiation dose rates from the Pinellas Plant.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Remainder	4.7E-3	5.3E-1
Gonads	4.4E-3	4.7E-1
Breast	4.4E-3	4.7E-1
Lungs	4.4E-3	4.7E-1
Red marrow	4.3E-3	4.7E-1

Table 2.16-3. Estimated fatal cancer risks from the Pinellas Plant.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
1E-7	2E-4

Table 2.16-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Pinellas Plant.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	1,900,000	2E-4
Totals	1,900,000	2E-4

2.17 NEVADA TEST SITE

2.17.1 Site Description

The Nevada Test Site lies about 100 km northwest of Las Vegas, Nevada, in Nye County. This facility, which is part of DOE'sweapons research and development complex, is responsible for design, maintenance, and testing of nuclear weapons. Other activities at this site include development of new nuclear energy technologies and radioactive waste disposal.

Radionuclide emissions result primarily from underground tests of nuclear weapons. Sources of these releases include drill-back operations, tunnel ventilation, leakage of gases from underground test sites, and resuspension of contaminated soils.

2.17.2 Basis for the Dose and Risk Assessment

2.17.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.17-1.

In modeling the site, all releases were assumed to be made from a single point source, since the nearest individual is 70 km from the site (Mo86). The releases were assumed from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes were assumed.

Table 2.17-1. Radionuclides released to air during 1986 from the Nevada Test Site.

Nuclide	Release Rate (Ci/y)
H-3	1.2E+2
I-131	2.4E+0
I-133	9.6E-6
Kr-85	4.3E+0
Xe-133	3.6E+4
Xe-133M	5.8E-2
Xe-135	4.1E-2

2.17.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Yucca Flats, Nevada. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 70,000 m from the assumed release point. Food consumption rates appropriate to a rural location were used.

2.17.3 Results of the Dose and Risk Assessment

The major contributors to exposure are xenon-133 (81 percent) and tritium (10 percent). The predominant exposure pathways are air immersion and ingestion.

The results of the dose and risk assessment are presented in Tables 2.17-2 through 2.17-4. Table 2.17-2 presents the dosesreceived by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.17-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure as well as estimated deaths per year in the regional population. Table 2.17-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.17-2. Estimated radiation dose rates from the Nevada Test Site.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Gonads	5.3E-3	1.2E-2
Remainder	3.5E-3	8.1E-3
Breast	6.5E-3	1.5E-2
Thyroid	1.9E-2	5.7E-2

Table 2.17-3. Estimated fatal cancer risks from the Nevada Test Site.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
1E-7	3E-6

Table 2.17-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Nevada Test Site.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	3,500	3E-6
Totals	3,500	3E-6

2.18 KNOLLS LABORATORY - KESSELRING

2.18.1 Site Description

The Kesselring site, occupying a 1,579-ha site, is located near West Milton, New York, approximately 27 km north of Schenectady. The surrounding area is rural and sparsely populated; about 1.08 million people live within 80 km.

The Kesselring site has four pressurized water reactor plants and associated support facilities used for training. Particulate and gaseous activity contained in the primary coolant may become airborne from reactor coolant discharges and sampling operations and during laboratory operations.

At the Kesselring site, exhaust air from reactor coolant discharges, sampling, and laboratory operations is passed through HEPA filters, monitored, and released from elevated stacks.

2.18.2 Basis for the Dose and Risk Assessment

2.18.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.18-1.

Table 2.18-1. Radionuclides released to air during 1986 from Knolls Atomic Power Lab-Kesselring.

Nuclide	Release Rate (Ci/y)
Ar-41	1.6E-1
C-14	3.4E-1
Co-60	3.4E-6
H-3	8.0E-2
Kr-83m	7.0E-4
Kr-85	2.0E-6
Kr-85m	2.0E-3
Kr-87	1.9E-3
Kr-88	4.0E-3
Xe-131m	9.2E-4
Xe-133	2.2E-2
Xe-135	2.3E-2

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class Y for cobalt-60) were assumed.

2.18.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Albany/CO, New York. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 250 m from the assumed release point (Mo86). Food consumption rates appropriate to an urban location were used.

2.18.3 Results of the Dose and Risk Assessment

The major contributors to exposure are argon-41 (69 percent), cobalt-60 (12 percent), and carbon-14 (7 percent). The predominant exposure pathways are air immersion for argon-41 and cobalt-60, and ground surface for carbon-14.

The results of the dose and risk assessment are presented in Tables 2.18-2 through 2.18-4. Table 2.18-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.18-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.18-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.18-2. Estimated radiation dose rates from the Knolls Lab-Kesselring.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Gonads	2.5E-3	1.5E-2
Remainder	3.8E-3	3.2E-2
Breast	4.4E-3	3.7E-2
Red marrow	6.9E-3	6.5E-2
Lungs	2.5E-3	1.8E-2

Table 2.18-3. Estimated fatal cancer risks from the Knolls Lab-Kesselring.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
1E-7	2E-5

Table 2.18-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Knolls Atomic Power Lab-Kesselring.

Risk Interval	Number of Persons	Deaths y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	1,200,000	2E-5
Totals	1,200,000	2E-5

2.19 BATTELLE COLUMBUS LABORATORY

2.19.1 Site Description

Battelle Columbus Laboratory (BCL) conducts various NRC-licensed activities, as well as activities under Department of Energy contracts (Sw87).

BCL operates two complexes in the Columbus Ohio, area. The first site is the King Avenue Site, which consists of 4 ha near a residential area in Columbus. The Ohio State University intramural sports practice field borders the site to the north.

The second site is the Nuclear Sciences Area of the West Jefferson site, which is located about 27 km west of the King Avenue laboratories. This site occupies about 5 ha on a 405-ha tract of land. Approximately 1.5 million people live within 80 km of the laboratory.

The King Avenue site has a uranium-235 processing facility located within Building 3. This building also houses the melting facility and powder metallurgy laboratory. The uranium processing facility manages all transactions involving nuclear material at the King Avenue site. However, handling of contract and licensed material has been very limited since 1977, and monitoring of airborne emissions was discontinued in 1975.

At the West Jefferson site, activities at the Nuclear Sciences Area include operations in the JN-1 hot cell (where irradiated reactor fuel elements are studied) and materials accountability and storage operations, conducted at the JN-2 vault. The JN-4 plutonium laboratory, where research was conducted on uranium-235/plutonium-239 nitride reactor fuel, is being decommissioned.

2.19.2 Basis for the Dose and Risk Assessment

2.19.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.19-1.

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class D for K-40, Class Y for uranium-235 and plutonium-239) were assumed.

2.19.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Columbus, Ohio. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 750 m from the assumed release point (Mo86). Food consumption rates appropriate to an urban location were used.

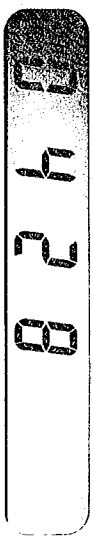

Table 2.19-1. Radionuclides released to air during 1986 from Battelle Columbus.

Nuclide	Release Rate (Ci/y)
Ac-228	1.0E-5
Be-7	1.2E-5
Bi-214	2.4E-5
Co-57	1.4E-6
Co-60	3.7E-6
Cs-134	1.5E-6
Cs-137	3.0E-6
I-131	8.7E-7
K-40	3.0E-4
Kr-85	7.6E+0
Pb-212	3.0E-6
Pb-214	1.5E-5
Pu-239	4.0E-7
Sb-125	5.1E-6
Sr-90	5.8E-7
Tl-208	2.6E-6
U-235	2.6E-6

2.19.3 Results of the Dose and Risk Assessment

The major contributors to exposure are potassium-40 (61 percent), uranium-235 (24 percent), and plutonium-239 (10 percent). The predominant exposure pathways are ground surface for potassium-40 and inhalation for uranium-235 and plutonium-239.

The results of the dose and risk assessment are presented in Tables 2.19-2 through 2.19-4. Table 2.19-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.19-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.19-4 presents the estimated distribution of fatal cancer risk to the regional population.

2.20 FERMI NATIONAL LABORATORY

2.20.1 Site Description

The Fermi National Accelerator Laboratory is principally involved with basic research in high-energy physics. Another important activity involves the treatment of cancer patients with neutrons released by the second stage of the accelerator. The Fermi complex is located east of Batavia, Illinois, in the greater Chicago area.

Table 2.19-2. Estimated radiation dose rates from the Battelle Columbus Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	3.1E-3	1.5E-2
Gonads	8.7E-4	6.2E-3
Remainder	7.2E-4	5.2E-3
Breast	7.8E-4	5.7E-3

Table 2.19-3. Estimated fatal cancer risks from the Battelle Columbus Laboratory.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
2E-8	3E-6

Table 2.19-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Battelle Columbus.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	1,900,000	3E-6
Totals	1,900,000	3E-6

The accelerator at the Fermi Laboratory, a proton synchrotron, routinely operates at energies up to 400 GeV (billion electron volts). The proton beams produced in the accelerator are used in three different onsite experimental facilities: (1) the Meson area, (2) the Neutrino area, and (3) the Proton area. Radionuclides are produced in these areas and by the accelerator when either the proton beam itself or secondary particles interact with air.

Another source of radionuclides at Fermi Laboratory is a magnet-debonding oven, where failed magnets for the accelerator

are baked at high temperatures to break down the adhesives that help form the magnets.

2.20.2 Basis for the Dose and Risk Assessment

2.20.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.20-1.

Table 2.20-1. Radionuclides released to air during 1986 from Fermi National Accelerator Laboratory.

Nuclide	Release Rate (Ci/y)
C-11	3.4E+0
H-3	3.0E-3

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class D for C-11) were assumed.

2.20.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Midway Airport, Illinois. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 1,500 m from the assumed release point (Ba87, Mo84). Food consumption rates appropriate to a rural location were used.

2.20.3 Results of the Dose and Risk Assessment

The major contributor to exposure is carbon-11 (100 percent). The predominant exposure pathway is air immersion.

The results of the dose and risk assessment are presented in Tables 2.20-2 through 2.20-4. Table 2.20-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.20-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.20-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.20-2. Estimated radiation dose rates from the Fermi National Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Gonads	9.2E-4	4.1E-3
Remainder	7.1E-4	3.2E-3
Breast	8.6E-4	3.9E-3
Lungs	9.1E-4	4.1E-3
Red marrow	7.0E-4	3.2E-3

Table 2.20-3. Estimated fatal cancer risks from the Fermi National Laboratory.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
2E-8	1E-6

Table 2.20-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Fermi National Laboratory.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	7,700,000	1E-6
Totals	7,700,000	1E-6

2.21 SANDIA NATIONAL LABORATORY

2.21.1 Site Description

The operations at Sandia National Laboratories near Albuquerque, New Mexico, include weapons testing, arming and fusing nuclear weapons, and developing modifications to delivery systems (De87, Mo84). The major facilities include the Sandia Pulsed Reactor and the Annular Core Pulsed Reactor (both of which

are used to irradiate test materials) and the Relativistic Electron Beam Accelerator. Support facilities include the Neutron Generator Facility, the Tube Loading Facility, the Fusion Target Loading Facility, the Tritium Laboratory, and the Nondestructive Test Facility, all of which are located in Technical Areas (TA) I and V. TA-I, in the northwest corner of the site, also houses research and design laboratories. TA-III is the site of the Sandia low-level radioactive waste dump.

2.21.2 Basis for the Dose and Risk Assessment

2.21.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/yr, from all sources during 1986 are listed in Table 2.21-1.

Table 2.21-1. Radionuclides released to air during 1986 from Sandia National Laboratory/Lovelace Research Institute.

Nuclide	Release Rate (Ci/y)
Ar-41	5.5E+0
H-3	1.3E-1
Pb-212	8.5E-3

In modeling the site, all releases were assumed to be from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class D for lead-212) were assumed.

2.21.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Albuquerque/Sunpt, New Mexico. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 3,500 m from the assumed release point. Urban food consumption rates were used.

2.21.3 Results of the Dose and Risk Assessment

The major contributors to exposure are argon-41 (74 percent) and lead-212 (26 percent). The predominant exposure pathways are air immersion for argon-41 and inhalation for lead-212.

The results of the dose and risk assessment are presented in Tables 2.21-2 through 2.21-4. Table 2.21-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.21-3 presents the estimated lifetime fatal

cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.21-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.21-2. Estimated radiation dose rates from the Sandia National Laboratory/Lovelace Research Institute.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Remainder	5.3E-4	1.9E-2
Gonads	5.9E-4	2.1E-2
Lungs	1.2E-3	4.9E-2
Breast	5.4E-4	1.9E-2
Red marrow	5.6E-4	2.1E-2

Table 2.21-3. Estimated fatal cancer risks from the Sandia National Laboratory/Lovelace Research Institute.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
1E-8	8E-6

Table 2.21-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Sandia National Laboratory/Lovelace Research Institute.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	500,000	8E-6
Totals	500,000	8E-6

2.22 BETTIS ATOMIC POWER LABORATORY

2.22.1 Site Description

The Bettis Atomic Power Laboratory is situated on an 0.8 km^2 tract in West Mifflin, Pennsylvania, approximately 12 km south of Pittsburgh. This facility designs and develops nuclear power reactors.

2.22.2 Basis for the Dose and Risk Assessment

2.22.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.22-1.

Table 2.22-1. Radionuclides released to air during 1986 from Bettis Atomic Power Laboratory.

Nuclide	Release Rate (Ci/y)
Co-60	1.7E-6
Cs-137	1.7E-6
I-129	1.8E-6
I-131	6.9E-6
Kr-85	9.4E-1
Rn-220	6.3E-2
Sb-125	3.1E-5
Sr-90	1.7E-6
U-234	6.0E-7
U-238	6.0E-7
Xe-131m	1.5E-4
Xe-133	3.8E-7

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class Y for uranium-234 and uranium-238, Class W for antimony-125) were assumed.

2.22.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Pittsburgh, Pennsylvania. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 250 m from the assumed release point. Food consumption rates appropriate to a rural location were used.

2.22.3 Results of the Dose and Risk Assessment

The major contributors to exposure are uranium-234 and uranium-238 (69 percent) and antimony-125 (10 percent). The predominant exposure pathways are inhalation for uranium-234 and uranium-238, and ground surface for antimony-125.

The results of the dose and risk assessment are presented in Tables 2.22-2 through 2.22-4. Table 2.22-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.22-5 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.22-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.22-2. Estimated radiation dose rates from the Bettis Atomic Power Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	4.3E-3	3.5E-2

Table 2.22-3. Estimated fatal cancer risks from the Bettis Atomic Power Laboratory.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
1E-8	1E-6

Table 2.22-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Bettis Atomic Power Laboratory.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	3,100,000	1E-6
TOTALS	3,100,000	1E-6

2.23 KNOLLS LAB - WINDSOR

2.23.1 Site Description

The Windsor site consists of only 4 ha near Windsor, Connecticut, about 8 km north of the city of Hartford. The area is a rural far ing and industrial region along the Farmington River. Approximately 3.1 million people live within 80 km.

2.23.2 Basis for the Dose and Risk Assessment

2.23.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.23-1.

Table 2.23-1. Radionuclides released to air during 1986 from Knolls Atomic Power Lab-Windsor.

Nuclide	Release Rate (Ci/y)
Ar-41	7.8E-2
C-14	4.7E-2
Co-60	2.6E-7
H-3	1.1E-2
Kr-83M	5.1E-5
Kr-85	2.3E-7
Kr-85M	1.9E-4
Kr-87	1.4E-4
Kr-88	3.6E-4
Xe-131M	1.0E-5
Xe-133	1.9E-3
Xe-133M	6.6E-5
Xe-135	1.8E-3

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes were assumed.

2.23.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Hartford/Bradley, Connecticut. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 250 m from the assumed release point. Food consumption rates appropriate to a rural location were used.

2.23.3 Results of the Dose and Risk Assessment

The major contributor to exposure is argon-41 (93 percent). The predominant exposure pathway is air immersion.

The results of the dose and risk assessment are presented in Tables 2.23-2 through 2.23-4. Table 2.23-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.23-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.23-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.23-2. Estimated radiation dose rates from the Knolls Lab-Windsor.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Gonads	3.8E-4	2.3E-3
Remainder	3.0E-4	4.2E-3
Breast	3.5E-4	4.9E-3
Red marrow	3.0E-4	8.1E-3
Lungs	2.9E-4	2.5E-3

Table 2.23-3. Estimated fatal cancer risks from the Knolls Lab-Windsor.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
8E-9	2E-6

Table 2.23-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Knolls Atomic Power Lab-Windsor.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	3,200,000	2E-6
Totals	3,200,000	2E-6

2.24 ROCKY FLATS PLANT

2.24.1 Site Description

Activities at the Rocky Flats Plant, located in Jefferson County, Colorado, about 26 km from Denver, are restricted to fabrication and assembly of components for nuclear weapons and the support of these operations (Se88).

Fabrication operations include reduction rolling, blanking, forming, and heat treating. Assembly operations include cleaning, brazing, marking, welding, weighing, matching, sampling, heating, and monitoring. Solid residue generated during plutonium-related operations is recycled through one of two plutonium-recovery processes. Process selection depends on the purity and plutonium content of the residue. Both processes produce a plutonium nitrate solution from which the metal can be extracted. The recovered plutonium is returned to the storage vault for use in foundry operations. A secondary objective of the process is the recovery of americium-241.

Radionuclides are released from short stacks and building vents at this plant. Building 771, Main Plenum, was selected for comparison purposes and calculations. This point releases 54 percent of the plutonium-239 and -240 and 3 percent of the uranium-233, -234, and -235 emitted at Rocky Flats. The most significant release point for uranium is from a single duct in Building 883, which releases approximately 19 percent of the total uranium emissions from the plant.

2.24.2 Basis for the Dose and Risk Assessment

2.24.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.24-1.

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class Y for uranium-238, Class W for americium-241) were assumed.

Table 2.24-1. Radionuclides released to air during 1986 from Rocky Flats Plant.

Nuclide	Release Rate (Ci/y)
Am-241	4.8E-6
H-3	2.2E-1
Pu-233	1.7E-8
Pu-234	1.7E-8
Pu-238	9.8E-7
Pu-239	1.5E-5
Pu-240	1.5E-5
U-233	4.3E-6
U-234	4.3E-6
U-238	1.7E-5

2.24.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Denver/Stapleton, Colorado. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 750 m from the assumed release point (Se88). Food consumption rates appropriate to a rural location were used.

2.24.3 Results of the Dose and Risk Assessment

The major contributors to exposure are uranium-238 (35 percent) and americium-241 (45 percent). The predominant exposure pathway is inhalation.

The results of the dose and risk assessment are presented in Tables 2.24-2 through 2.24-4. Table 2.24-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.24-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.24-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.24-2. Estimated radiation dose rates from the Rocky Flats Plant.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	6.3E-3	1.2E-1
Endosteum	1.6E-2	2.0E-1
Remainder	7.5E-4	9.3E-3

Table 2.24-3. Estimated fatal cancer risks from the Rocky Flats Plant.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
1E-8	9E-6

Table 2.24-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Rocky Flats Plant.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	1,900,000	9E-6
Totals	1,900,000	9E-6

2.25 PANTEX PLANT

2.25.1 Site Description

The Pantex Plant, located 30 km northeast of Amarillo, Texas, is a nuclear weapons assembly and disassembly plant. Because most radioactive materials handled during the assembly of nuclear weapons are contained in sealed vessels, normal operations involving these materials do not result in major releases of radionuclides (La88).

2.25.2 Basis for the Dose and Risk Assessment

2.25.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.25-1.

Table 2.25-1. Radionuclides released to air during 1986 from Pantex Plant.

Nuclide	Release Rate (Ci/y)
H-3	1.3E-1
U-238	1.0E-5

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class Y for uranium-238) were assumed.

2.25.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Amarillo, TX. The 0-80 km population distribution was produced using the computer code SECPOP and the 1980 Census Bureau data. Nearby individuals were located 1,500 m from the assumed release point (La88). Food consumption rates appropriate to a rural location were used.

2.25.3 Results of the Dose and Risk Assessment

The major contributor to exposure is uranium-238 (94 percent). The predominant exposure pathway is inhalation.

The results of the dose and risk assessment are presented in Tables 2.25-2 through 2.25-4. Table 2.25-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.25-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.25-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.25-2. Estimated radiation dose rates from the Pantex Plant.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	2.2E-3	3.5E-3

Table 2.25-3. Estimated fatal cancer risks from the Pantex Plant.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
4E-9	7E-8

Table 2.25-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Pantex Plant.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	260,000	7E-8
Totals	260,000	7E-8

2.26 KNOLLS LAB - KNOLLS

2.26.1 Site Description

Knolls Atomic Power Laboratory has facilities at three separate sites: Knolls, Kesselring, and Windsor. Development ofnuclear reactors and training of operating personnel are the major efforts at the Knolls Laboratory. The Knolls and Kesselring complexes are located near Schenectady, NY, and the Windsor site is near Windsor, Connecticut.

Operations at the Knolls site involving radioactive materials are serviced by controlled exhaust systems that

discharge through elevated stacks. Exhaust air is passed through HEPA and carbon filters and is continuously sampled prior to release. Small amounts of krypton-85 generated by examination of irradiated fuel are released in the exhaust stacks. Generation of argon-41 is minimized by controlling air leakage into the low-power critical assembly.

2.26.2 Basis for the Dose and Risk Assessment

2.26.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.26-1.

Table 2.26-1. Radionuclides released to air during 1986 from Knolls Atomic Power Lab-Knolls.

Nuclide	Release Rate (Ci/y)
Co-60	1.0E-6
I-131	3.7E-6
Kr-85	7.9E-1
Kr-85m	4.1E-3
Kr-87	5.8E-3
Kr-88	1.2E-2
Pu-238	1.3E-7
Sb-125	2.8E-5
Sn-113	1.3E-6
Sr-90	2.5E-5
U-234	3.3E-6
U-235	1.0E-7
U-236	6.6E-9
U-238	9.1E-10
Xe-131m	5.7E-7
Xe-133	1.4E-3
Xe-135	1.3E-2

In modeling the site, all releases were assumed to be made from a 10-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class Y for uranium-234) were assumed.

2.26.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Albany/CO, New York. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 250 m from the assumed release point. Food consumption rates appropriate to an urban location were used.

2.26.3 Results of the Dose and Risk Assessment

The major contributor to exposure is uranium-234 (79 percent). The predominant exposure pathway is inhalation.

The results of the dose and risk assessment are presented in Tables 2.26-2 through 2.26-4. Table 2.26-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.26-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.26-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.26-2. Estimated radiation dose rates from the Knolls Lab-Knolls.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Lungs	1.7E-3	3.1E-2

Table 2.26-3. Estimated fatal cancer risks from the Knolls Lab-Knolls.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
3E-9	1E-6

Table 2.26-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Knolls Atomic Power Lab-Knolls.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	1,200,000	1E-6
Totals	1,200,000	1E-6

2.27 AMES LABORATORY

2.27.1 Site Description

Until 1978, the Ames Laboratory, which is operated by Iowa State University, was used as a neutron source for the production of byproduct materials and the neutron irradiation of various materials for research. The reactor was fueled with enriched uranium, moderated and cooled by heavy water, and operated continuously at 5,000 watts thermal. Operation of the Ames Laboratory Research Reactor was terminated on December 1, 1977. Decommissioning began January 3, 1978, and was completed on October 31, 1981. A waste processing and disposal facility still located at the site serves the campus reactor and research laboratories.

Prior to its decommissioning, the major airborne releases from the research reactor were tritium and argon-41. Tritium, the major radionuclide released during the 1981 decommissioning activities, was emitted from the 30-m reactor stack, which is 215 m from the nearest property boundary. Monitoring has indicated that no airborne emissions from the research laboratories have reached the main campus.

2.27.2 Basis for the Dose and Risk Assessment

2.27.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.27-1.

Table 2.27-1. Radionuclides released to air during 1986 from Ames Laboratory.

Nuclide	Release Rate (Ci/y)
H-3	7.6E-2

In modeling the site, all releases were assumed to be made from a 10-m stack.

2.27.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Waterloo, Iowa. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 750 m from the assumed release point. Rural food consumption rates were used.

2.27.3 Results of the Dose and Risk Assessment

The major contributor to exposure is tritium (100 percent). The predominant exposure pathways are ingestion and inhalation.

The results of the dose and risk assessment are presented in Tables 2.27-2 through 2.27-4. Table 2.27-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.27-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.27-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.27-2. Estimated radiation dose rates from the Ames Laboratory.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Remainder	1.6E-5	2.3E-4
Gonads	1.3E-5	1.8E-4
Breast	1.3E-5	1.8E-4
Lungs	1.3E-5	1.8E-4
Red marrow	1.3E-5	1.8E-4

Table 2.27-3. Estimated fatal cancer risks from the Ames Laboratory.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
4E-10	9E-8

Table 2.27-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from the Ames Laboratory.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	680,000	9E-8
Totals	680,000	9E-8

2.28 ROCKETDYNE ROCKWELL

2.28.1 Site Description

Rockwell International operates two facilities, one near Los Angeles and one near Santa Susana, Calafornia. These facilities conduct research and development and also manufacture nuclear reactor components. The Los Angeles facility performs uranium fuel processing operations and conducts research involving gamma radiation. The Santa Susana facility uses neutron radiography to inspect nuclear reactor components. This facility also serves as a materials handling laboratory and waste processing operation for other DOE facilities.

Radionuclide emissions originate from the materials handling laboratory and the waste processing facilities at the Santa Susana site.

2.28.2 Basis for the Dose and Risk Assessment

2.28.2.1 Source Terms and Release Point Characterization

The total airborne releases, in Ci/y, from all sources during 1986 are listed in Table 2.28-1.

Table 2.28-1. Radionuclides released to air during 1986 from Rocketdyne Division, Rockwell International.

Nuclide	Release Rate (Ci/y)
Sr-90	1.3E-5

In modeling the site, all releases were assumed to be made from a 30-m stack. Default particle sizes (1.00 AMAD) and solubility classes (Class D for strontium-90) were assumed.

2.28.2.2 Other Parameters Used in the Assessment

Meteorological data used in the assessment are from Burbank, California. The 0-80 km population distribution was produced using the computer code SECPOP and 1980 Census Bureau data. Nearby individuals were located 250 m from the assumed release point. Food consumption rates appropriate to an urban location were used.

2.28.3 Results of the Dose and Risk Assessment

The major contributor to exposure is strontium-90 (100 percent). The predominant exposure pathway is inhalation.

The results of the dose and risk assessment are presented in Tables 2.28-2 through 2.28-4. Table 2.28-2 presents the doses received by nearby individuals and the regional population. Doses to organs accounting for 10 percent or more of the risk are presented. Table 2.28-3 presents the estimated lifetime fatal cancer risk to nearby individuals with maximum exposure, as well as estimated deaths per year in the regional population. Table 2.28-4 presents the estimated distribution of fatal cancer risk to the regional population.

Table 2.28-2. Estimated radiation dose rates from Rocketdyne Division, Rockwell International.

Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Red marrow	7.0E-6	1.4E-3
Endosteum	1.5E-5	3.2E-3

Table 2.28-3. Estimated fatal cancer risks from Rocketdyne Division, Rockwell International.

Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
2E-11	7E-8

Table 2.28-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) population from Rocketdyne Division, Rockwell International.

Risk Interval	Number of Persons	Deaths/y
1E-1 - 1E+0	0	0
1E-2 - 1E-1	0	0
1E-3 - 1E-2	0	0
1E-4 - 1E-3	0	0
1E-5 - 1E-4	0	0
1E-6 - 1E-5	0	0
< 1E-6	8,800,000	7E-8
Totals	8,800,000	7E-8

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3. NRC-LICENSED AND NON-DOE FEDERAL FACILITIES

3.1 INTRODUCTION AND BACKGROUND

The Nuclear Regulatory Commission (NRC) and the Agreement States issue licenses for the use of radionuclides. This chapter deals with all of these licensed facilities that are not involved in nuclear power generation and with Federal facilities other than those owned by the Department of Energy (DOE). The facilities that are part of the light-water uranium fuel cycle are discussed in Chapter 4 of this report, and DOE facilities are examined in Chapter 2. Facilities licensed only for the possession of sealed sources are not considered, since sealed sources do not release radionuclides to air.

NRC and Agreement State licensees are divided into by-product, source material, and special nuclear material categories. By-product licensees are further divided into hospitals, radiopharmaceutical manufacturers, research laboratories, sealed source manufacturers, and low-level waste incinerators. Special nuclear material licensees are divided into research reactors and non-light-water reactor fuel fabricators.

Most non-DOE Federal facilities are included in the above categories. For example, Veterans Administration hospitals are included in the hospital category. Federal facilities not included in any other category are discussed separately. Thus, this source category is divided into nine sub-categories:

- Hospitals
- Radiopharmaceutical Manufacturers
- Research Laboratories
- Research Reactors
- Sealed Source Manufacturers
- Non-LWR Fuel Fabricators
- Source Material Licensees
- Low-Level Waste Incinerators
- Non-DOE Federal Facilities.

There are approximately 6,000 such facilities, and they are found in all 50 states. The largest groups are the 3,680 licensed hospitals and the 1,500 research laboratories. The smallest group is the four non-light-water reactor fuel fabricators. These facilities emit radionuclides over a wide spectrum, usually in small amounts. Typically, effluent controls are activated charcoal filters to delay the release of iodine and noble gases and high efficiency particulate air (HEPA) filters to capture particulates. Controls and information pertaining to each category are discussed separately.

The information presented in this chapter was obtained from sources identified by a literature search and direct contact with licensees and regulators. Whenever possible, current (1988) data

were used in the assessment. To determine which facilities are likely to have the highest levels of emissions, Radiation Safety Officers at licensed facilities and staff at the NRC and Agreement States were contacted. The facilities identified were then contacted to obtain effluent release data and additional site-specific information. Since it was not possible to survey all 6,000 licensees, facilities with high or unusual emissions may have been missed.

The raw data from a Conference of Radiation Control Program Directors' (CRCPD) survey of waste production and effluents were also used (CRC87). While this survey does not identify specific facilities or their exact locations, it does provide data on the number of facilities and emissions. Additional data were obtained from the American Hospital Association (AHA86) and from survey results presented in Cook (Co81) and Corbit (Co83).

Based on the emissions identified for each facility, the radiation doses and risks to nearby individuals and to the regional population were asssessed. The methodology discussed in Volume I of this Background Information Document was used in all of the assessments.

3.2 HOSPITALS

3.2.1 General Description

Over half of the hospitals in the United States handle radiopharmaceuticals. Most use them for radionuclide imaging, in which a compound labeled with a nuclide such as technetium-99m is traced through the patient's body using an elaborate radiation detection system. Hospitals also administer large, therapeutic amounts of nuclides such as I-131. Radiopharmaceuticals are mostly in liquid form but can also be gaseous or solid.

Radiogases, such as Xe-133, are used for in-vivo lung studies. The gas is inhaled by the patient, then exhaled into a collection or ventilation system. The gas is either released directly to air, charcoal filtered, or held for decay. Liquids are stored and handled in fume hoods, which may have effluent filters. They can be volatilized during administration to the patient, which normally occurs in a room at negative pressure but without effluent controls.

Data from the American Hospital Association indicate that there are 3,680 hospitals in the United States that handle diagnostic radiopharmaceuticals (AHA86). About a third of these (1,371) also handle therapeutic amounts of these drugs. Two-thirds of these hospitals are located in urban areas; the rest are in rural locations. States with the largest number of such hospitals are California (317), Texas (270), and New York (197).

3.2.2 Basis for Risk Assessment

The doses and risks caused by release of radionuclides to air were assessed by constructing one model facility to represent typical hospitals and a second model facility to represent a very large hospital with larger emissions.

3.2.2.1 Emissions

Effluent data from over 100 hospitals, obtained from the CRCPD survey (CRC87), were used to construct the model facilities. Nearly all hospitals reported releases of xenon-133; the highest release was 31.4 Ci/y and the average was 1 Ci/y. Eight hospitals reported releases of iodine-125; the highest release, 0.039 Ci/y, is about four times the average value of 0.01 Ci/y. Six reported releases of iodine-131; these also averaged 0.01 Ci/y. Other nuclides were reported by one or two hospitals. The absence of reported radioiodine releases is common, due to the lack of effluent monitoring at hospitals. Facilities with no reported emissions were omitted from the computation of the average release rates.

The average emissions for xenon-133, iodine-125, and iodine-131 were used to construct the typical model facility. These average values are consistent with the release rates reported by Corbit (Co83) and SC&A (SCA84). The large model hospital was created using the maximum release reported in the CRCPD survey. The estimated emissions for the model hospitals are shown in Table 3-1.

Table 3-1. Estimated emissions from model hospitals.

Facility	Radionuclide	Release Rate (Ci/y)
Typical Hospital	Xe-133	1.0E+0
	I-125	1.0E-2
	I-131	1.0E-2
Large Hospital	Xe-133	3.1E+1
	I-125	3.9E-2
	I-131	2.4E-2

3.2.2.2 Site Characteristics

The model representing typical hospitals was assessed at two different locations. To represent the doses and risks in urban areas, an assessment was made using demographic and meteorological data for Boston, MA. Data for Columbia, MO, were used to estimate the doses and risks for rural areas. The two

assessments used urban and rural food supply assumptions respectively. In both assessments, the stack height was set at 1 meter and the nearest individuals were assumed to be 150 meters downwind from the release point. The large model hospital was again assessed using a 1 meter release height, an urban location, and assuming the nearby individuals are 100 meters downwind.

Detailed information on the values input to the assessment codes for these models is presented in Appendix A.

3.2.3 Results of the Dose and Risk Assessment

The results of the dose and risk assessment of the model hospital facilities are presented in Tables 3-2 and 3-3. The highest doses and risks are estimated for the large model hospital. The highest doses to both nearby individuals and the regional population are to the thyroid, 5.1 mrem/y and 12 person-rem/year, respectively. These doses are caused by the iodine-125, predominately via the ingestion pathway. The risks predicted for the model large hospital indicated that nearby individuals have a lifetime fatal cancer risk of approximately 2 in one million, and that there will be 7E-5 deaths/year in the regional population.

The results for the model hospitals representing typical urban and rural hospitals show lower doses and risks. For the nearby individuals and the regional population at the model urban hospital the highest doses are also to the thyroid, 0.2 mrem/y, and 1.4 person-rem/year, respectively. Iodine-125 and iodine-131 are the significant radionuclides, and inhalation is the predominant pathway. The releases from the urban hospital are estimated to result in lifetime fatal cancer risks to nearby individuals much less than 1 in one million and to cause approximately 1E-5 deaths/year in the regional population.

The highest doses received by nearby individuals (28 mrem/y) and the regional populations (7.0 person-rem/year) at the model rural hospital are also to the thyroid, due to emissions of radioiodines. These doses are higher than those at the urban hospital due to the greater significance of the ingestion pathway.

The estimated distribution of the fatal cancer risk in the exposed populations is presented in Table 3-4. An estimated 6E-2 deaths/year are caused by emissions from all hospitals. These estimates were made by scaling the results obtained for the typical urban and rural model hospitals by the number of urban and rural hospitals, 2,467 and 1,213, respectively. The number of persons at risk was constrained to the population of the United States.

Table 3-2. Estimated radiation dose rates from model hospitals.

Facility	Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Urban Hospital	Gonads	1.2E-2	4.6E-2
	Breast	1.4E-2	5.6E-2
	Thyroid	2.0E-1	1.4E+0
	Remainder	7.1E-3	2.8E-2
Rural Hospital	Thyroid	2.8E+1	7.0E+0
Large Hospital	Thyroid	5.1E+0	1.2E+1

Table 3-3. Estimated fatal cancer risks from model hospitals.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
Urban Hospital	2E-7	1E-5
Rural Hospital	5E-6	2E-5
Large Hospital	2E-6	7E-5

Table 3-4. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all hospitals.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	0	0
1E-6 to 1E-5	*	*
< 1E-6	240,000,000	6E-2
Totals	240,000,000	6E-2

*Results from the large model hospital indicate there may be some individuals at this risk level, but insufficient information is available to quantify either the number of persons or the deaths/year.

3.2.4 Supplementary Control Options and Costs

Emissions from facilities in this segment of the NRC-licensed source category do not result in exposures or risks high enough to warrant a full evaluation of supplementary control options and costs. Well-proven control technologies such as charcoal for iodine or decay traps for noble gases could be employed. Costs for any such system cannot be accurately determined due to the number of facilities and the lack of information on ventilation rates and on the extent of current use of controls.

3.3 RADIOPHARMACEUTICAL MANUFACTURERS

3.3.1 General Description

Radiopharmaceutical suppliers, distributors, and nuclear pharmacies number approximately 120 (Ce81). These are broken down into 15 large firms, 70 small to medium-sized firms, and 35 nuclear pharmacy operators. The analysis focused on the large firms that manufacture the radionuclides. These firms handle large amounts of radionuclides in hot cells, which are equipped with air cleaning systems (typically HEPA filters and charcoal). The smaller firms change the chemical form of the nuclides, while the pharmacies repackage the material into convenient amounts.

Information obtained on small firms and pharmacies suggests that radionuclides are handled in fume hoods, which are equipped with very efficient air cleaning filters. The most common filters are charcoal beds, which trap radioiodines and noble gases. Airborne effluents of these facilities are consequently very much lower than those of the large manufacturers.

3.3.2 Basis for Risk Assessment

The assessment of radiopharmaceutical manufacturers is based on the results obtained for four reference facilities. The reference facilities are actual manufacturers that are among the largest producers.

3.3.2.1 Emissions

Emissions data for three of the reference facilities were obtained from the manufacturers themselves. The fourth facility operates a nuclear reactor and is thus required to file effluent reports with the NRC. The dose and risk assessments are based on 1987 effluent data. Emissions data were also available from the CRCPD survey (1987) for seven unidentified facilities. These data were used for comparative purposes only. Emissions for the reference facilities are shown in Table 3-5.

Table 3-5. Effluent release rates (Ci/y) for radiopharmaceutical manufacturers.

Radionuclide	Reference Facility A	B	C	D
P-32	-	1.6E-2	-	-
S-35	1.9E-2	1.6E-2	3.8E-1	-
I-125	1.3E-2	2.0E-2	-	2.5E+0
I-131	-	2.5E-3	-	3.9E+0
H-3	-	-	9.8E+1	-
C-14	-	-	8.5E+0	-
Xe-135	-	-	-	8.1E+3
Xe-135m	-	-	-	2.9E+3
Xe-133	-	2.8E+0	-	1.4E+4
Xe-133m	-	-	-	4.5E+2
Kr-88	-	-	-	1.7E+3
Kr-87	-	-	-	1.2E+2
Kr-85	-	9.5E-1	-	1.7E+0
Kr-85m	-	-	-	1.3E+3
Kr-83m	-	-	-	4.6E+2
Ar-41	-	-	-	1.1E+3

3.3.2.2 Site Characteristics

Actual site data, where available, were used for the risk assessments. Meteorological data were taken from the nearest airports: Chicago, IL (A); Boston, MA (B&C); and Newburgh, NY (D). Stack heights used were all 15 m. Distances to the nearby individuals are 430 m, 200 m, 150 m, and 480 m. Food fractions typical of urban areas were assumed in all cases except Reference Facility D where rural food fractions were used.

3.3.3 Results of the Dose and Risk Assessment

The doses and risks estimated for the four reference facilities are presented in Tables 3-6 and 3-7. The highest estimated doses and risks are at Reference Facility D, where nearby individuals and the regional population are predicted to receive doses to the thyroid of 9.5E+1 mrem/y and 6.0E+2 person-rem/year respectively. The lifetime fatal cancer risk to nearby individuals is estimated to be 2E-4; the releases cause 2E-2 deaths/year in the regional population.

The total risk from radiopharmaceutical manufacturers is estimated to be 2E-2 deaths/year. This is the sum of the estimates for Reference Facilities A through C, multiplied by 5 and added to the estimate for Reference Facility D. The factor of 5 is used to expand the three reference facilities to cover all 15 actual facilities. Facility D is treated individually because it is the only facility that operates a nuclear reactor. Table 3-8 presents the collective risks, and number of people at risk, as a function of individual risk level.

Table 3-6. Estimated radiation dose rates from radiopharmaceutical manufacturers.

Reference Facility	Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
A	Gonads	8.9E-4	8.9E-3
	Thyroid	5.4E-2	2.4E+0
B	Gonads	4.4E-2	1.4E-2
	Breast	5.2E-2	1.7E-2
	Thyroid	7.3E-1	1.6E+0
	Remainder	2.8E-2	1.3E-2
C	Gonads	7.1E-3	7.2E-1
	Breast	7.5E-3	9.9E-1
	Red Marrow	7.9E-3	1.3E+0
	Lungs	7.2E-3	7.7E-1
	Remainder	7.9E-3	9.9E-1
D	Gonads	7.6E+0	7.4E+1
	Breast	7.5E+0	7.6E+1
	Thyroid	9.5E+1	6.0E+2
	Remainder	5.7E+0	5.4E+1

Table 3-7. Estimated fatal cancer risks from reference radiopharmaceutical manufacturers.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
A	2E-8	7E-6
B	1E-6	9E-6
C	2E-7	4E-4
D	2E-4	2E-2

Table 3-8. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all radiopharmaceutical manufacturers.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	3,100	2E-3
1E-6 to 1E-5	140,000	3E-3
< 1E-6	110,000,000	2E-2
Totals	110,000,000	2E-2

3.3.4 Supplementary Control Options and Costs

Supplemental controls are examined for Reference Facility D, which has the highest estimated doses and risks. The nuclides contributing the most to dose are iodine-125 and iodine-131. Control of these nuclides is typically by adsorption on activated charcoal. However, Reference Facility D already employs this control method.

Nevertheless, it is possible to increase the efficiency of the existing charcoal adsorption system. Factors that influence efficiency are the impregnant used, flow rate, humidity, and temperature (Mo83). The first supplemental control examined is drying the exhaust air before it enters the charcoal adsorbers. Because the retention efficiency of charcoal is degraded by high humidity conditions, drying the exhaust air will boost efficiency.

The second option is chilling the charcoal beds. At lower temperatures, iodine is retained on the charcoal for longer periods. With a short half-life nuclide, such as iodine-131 (8 days), the activity decaying on the beds can be greatly increased.

The cost of employing these enhancements is difficult to determine, because they are dependent upon the configuration of the existing system. If the original installation allowed for the addition of these options at a later date, then their installation would not be difficult. However, this is probably not the case.

Lacking the data needed to perform an engineering study, the cost of these modifications can only be estimated grossly. At 50 percent of the cost of a new system, this is estimated to be $350,000 (DM80).

The effectiveness of these modifications can only be estimated. A reduction in radioiodine emissions of 99 percent and noble gas emissions of 75 percent can be assumed. Such a reduction would lower the calculated risks from this facility to 5E-3 deaths/year, reducing the predicted fatalities caused by releases from all radiopharmaceutical manufacturers to 7E-3 deaths/year.

3.4 LABORATORIES

3.4.1 General Description

The NRC and Agreement States license approximately 1,500 laboratories that use radionuclides in unsealed forms. This number is obtained by taking the total number of NRC-licensed laboratories to be approximately 800 (NRC87) and adding it to a previous count of 700 facilities licensed by the Agreement States (Co83). These laboratories are estimated to be 57 percent academic and the remainder either government or private research facilities. This estimate assumes that the number of academic laboratories is a more stable figure and has remained relatively unchanged from previous estimates (Ce81).

Academic laboratories generally encompass a large number of sites in one area and use small amounts of a large number of radionuclides. Twenty-nine radionuclides were identified in use at various laboratories. Private and government laboratories use millicurie to curie amounts of particular radioisotopes, depending upon the actual procedures used. One of the more important applications is the use of radioactively labeled chemicals (i.e., radioiodine labeled proteins) to trace dynamic processes.

The most pervasive form of effluent control is one or more high efficiency particulate air (HEPA) filters in series connected to a fume hood, hot cell, or glove box containing the radioactive material. Often charcoal filters are used alone or in series with HEPA filters to control the release of iodine and noble gases. Exhaust alarms are typically set to sound if the concentration at the release point reaches 10 percent of the maximum permissible concentration (MPC) limit established by the licensing authority. Quality assurance is maintained by periodic wipe testing of the exhaust system either before the last filter, if the filters are in a series, or at the point of release.

3.4.2 Basis for Risk Assessment

3.4.2.1 Emissions

Emissions data were gathered from 46 facilities. The results from the CRCPD survey of effluents were also used. This was a confidential survey, with the laboratories separated into academic, private, and government facilities. The results from

Corbit (Co83) were used, but only on a limited basis, because data were separated by isotope and not by facility.

Approximately 41 percent of all laboratories have emissions that are either zero or below the lower limits of detection of their monitoring equipment. The majority of the laboratories that do emit detectable quantities have exhaust concentrations between 1 and 5 percent of the applicable MPC. The largest emissions are estimated to be less than 10 percent of the MPC, but for the purpose of this study were conservatively assumed to be 10 percent of the MPC. Emissions are usually not monitored continuously; instead, surveys are conducted monthly or bi-monthly, and the emissions are estimated from these measurements.

A weighted average of all the information, omitting zero responses, was used to estimate emissions for the model facility. These are given in Table 3-9. The emission data were weighted by segment composition (private/government = 43 percent, academic = 57 percent) and sample size (primary = 45, CRCPD = 140, and Corbit = 44). The Corbit study (Co83) was given a weight equivalent to one-half of its actual weight because it was not separated into academic and private facilities. Finally, the large number of nuclides was reduced by screening out those nuclides making a negligible contribution to dose.

Table 3-9. Effluent release rates (Ci/y) for laboratories.

Radionuclide	Model Facility	Reference Facility A
H-3	1.1E+0	-
C-14	3.9E-3	-
S-35	4.7E-4	-
Co-60	3.8E-5	2.1E-4
Kr-85	1.8E-1	-
I-125	2.4E-3	-
I-131	5.1E-4	8.1E-3
Xe-133	2.2E-1	-
Cs-137	-	1.5E-4
Pu-239	3.7E-9	-
Am-241	7.6E-10	-

3.4.2.2 Site Characteristics

The model facility was placed in an urban area for purposes of the risk assessment. Meteorological data were taken from an actual airport. The release point was characterized as a 6 m stack, 350 m from the closest resident. Facility A is an actual laboratory with a 10 m release height. Meteorological data from the nearest airport were used in the analysis. The closest resident was in an urban area, 100 m from the stack.

3.4.3 Results of the Dose and Risk Assessment

The results of the dose and risk assessment of the largest and model facilities are presented in Tables 3-10 and 3-11. The estimated organ doses are all below 1 mrem/y for nearby individuals, and the maximum lifetime fatal cancer risk is estimated to be 3E-7.

The estimated distribution of the fatal cancer risk in the exposed population is presented in Table 3-12. The total collective risk (deaths/year) from research laboratories is obtained by scaling the model facility risk by 622, the estimated number of laboratories that have non-zero emissions. The result is an estimated 8E-3 deaths/year. The number of persons at risk is constrained to the population of the United States.

Table 3-10. Estimated radiation dose rates from laboratories.

Facility	Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Model Laboratory	Gonads	1.2E-2	3.4E-2
	Breast	1.3E-2	3.5E-2
	Thyroid	1.5E-1	4.3E-1
	Remainder	8.8E-3	3.1E-2
Reference Laboratory A	Gonads	6.6E-3	9.9E-2
	Breast	6.0E-3	9.0E-2
	Thyroid	2.7E-2	5.2E-1
	Remainder	5.0E-3	7.6E-2

Table 3-11. Estimated fatal cancer risks from laboratories.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
Model Laboratory	3E-7	1E-5
Reference Laboratory A	1E-7	3E-5

Table 3-12 Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all laboratories.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	0	0
1E-6 to 1E-5	0	0
< 1E-6	240,000,000	8E-3
Totals	240,000,000	8E-3

3.4.4 Supplementary Control Options and Costs

Emissions from facilities in this segment of the NRC-licensed source category do not result in doses or risks high enough to warrant a full evaluation of supplementary control options and costs.

3.5 RESEARCH AND TEST REACTORS

3.5.1 General Description

There were 70 research and test reactors operating as of December 1987 (NRC87). These reactors range in power level from zero (three critical experiment facilities) to 10,000 kilowatts. Most are located at universities and are used for teaching and research. Of the many different designs and manufacturers, the most common is General Atomics' TRIGA reactor.

There are two additional unlicensed reactors operated by the U.S. Army in Maryland and New Mexico. They are discussed in Section 3.10 of this chapter.

Most facilities ventilate the reactor building directly to the atmosphere through tall stacks or roof vents. The larger facilities employ particulate filters. Nearly all of the facilities monitor their effluents.

3.5.2 Basis for Risk Assessment

Doses and risks resulting from test and research reactors are evaluated on the basis of four actual reactors with the largest emissions.

3.5.2.1 Emissions

Emission data, shown in Table 3-13, were collected for the four largest emitters identified by Corbit (Co83). These include three university research reactors and one government research reactor. Emissions data from Corbit were supplemented by information presented in the facilities' annual operating reports (e.g., MIT87). The principal nuclide emitted is argon-41. Tritium is also emitted, although in lesser amounts.

Table 3-13. Effluent release rates (Ci/y) for research reactors.

Facility	Radionuclide H-3	Ar-41
Reference Reactor A	1.6E+1	2.5E+3
Reference Reactor B	1.6E+2	4.7E+2
Reference Reactor C	-	4.2E+3
Reference Reactor D	-	2.5E+2

3.5.2.2 Site Characteristics

Actual site data were used for the four risk assessments. Meteorological data were taken from airports near the four facilities (Columbia, MO; Ft. Meade, MD; Boston, MA; Providence, RI). The stack heights are 33 m, 33 m, 50 m, and 34 m, respectively. Rural food supply assumptions were used for all cases except Boston. The distances to the nearest individuals are 750 m, 1,500 m, 750 m, and 1,500 m, respectively.

3.5.3 Results of the Dose and Risk Assessment

Doses and risks were calculated for each of the four reference reactors. The results are presented in Tables 3-14 and 3-15. The highest exposures received by nearby individuals are estimated to be 0.8 mrem/y to the gonads, and the individuals at highest risk are estimated to have a lifetime fatal cancer risk of 2E-5.

The fatal cancer risk estimated from these four reactors was extrapolated to obtain the total collective risk (deaths/year) from all research and test reactors. The extrapolation is based on the ratio of the argon-41 released by the four largest emitters (7,416 Ci/y) to the argon-41 released by all 70 research reactors (12,557 Ci/y). This ratio, 0.59, was used to scale up the risk from the four reactors to the total population risk of 4E-2 deaths/year from all research and test reactors. Table 3-16 presents the estimated collective risk, and the number of people at risk, as a function of individual risk level.

Table 3-14. Estimated radiation dose rates from research reactors.

Facility	Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Reference Reactor A	Gonads	7.8E-1	8.1E+0
	Breast	7.0E-1	7.3E+0
	Red Marrow	6.0E-1	6.3E+0
	Lungs	6.0E-1	6.2E+0
	Remainder	6.0E-1	6.3E+0
Reference Reactor B	Gonads	2.6E-1	7.3E+0
	Breast	2.3E-1	6.9E+0
	Red Marrow	2.0E-1	6.2E+0
	Lungs	2.0E-1	6.2E+0
	Remainder	2.0E-1	6.8E+0
Reference Reactor C	Gonads	2.7E-1	6.8E+1
	Breast	2.4E-1	6.1E+1
	Red Marrow	2.1E-1	5.2E+1
	Lungs	2.1E-1	5.2E+1
	Remainder	2.1E-1	5.2E+1
Reference Reactor D	Gonads	3.6E-2	4.4E-1
	Breast	3.3E-2	3.9E-1
	Red Marrow	2.8E-2	3.4E-1
	Lungs	2.8E-2	3.3E-1
	Remainder	2.8E-2	3.4E-1

Table 3-15. Estimated fatal cancer risks from research reactors.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
Reference Reactor A	2E-5	2E-3
Reference Reactor B	5E-6	2E-3
Reference Reactor C	6E-6	2E-2
Reference Reactor D	7E-7	1E-4

Table 3-16. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from research and and test reactors.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	1,300	2E-4
1E-6 to 1E-5	630,000	2E-2
< 1E-6	23,000,000	2E-2
Totals	24,000,000	4E-2

3.5.4 Supplementary Control Options and Costs

Emissions from facilities in this segment of the NRC-licensed source category do not result in exposures or risks high enough to warrant a full evaluation of supplementary control options and costs.

3.6 SEALED SOURCE MANUFACTURERS

3.6.1 General Description

Sealed source manufacturers take radionuclides in an unsealed form and put them into a permanently sealed container. Two categories of sealed source manufacturers contribute to airborne emissions. The first category consists of manufacturers that produce sealed radiation sources other than tritium (such as Am-241). There are eight known manufacturers of this type. An additional six manufacturers of this type (e.g., The Nucleus, Oak Ridge, TN) use only exempt quantities of radionuclides and produce negligible emissions.

The other category of sealed source manufacturer seals tritium gas into self-luminous lights. There are three known firms that perform this type of work. All of these facilities are located in industrial areas. They rely heavily on engineered safeguards to prevent releases of radionuclides.

The radiation source manufacturers use high efficiency particulate air (HEPA) filters singly or in series to remove radionuclides from their effluent streams. The lighting manufacturers use desiccant columns, sometimes combined with catalytic recombiners, to remove tritium from their effluents. The only part of the process that results in emissions is the loading of radionuclides into containers which are subsequently sealed. All of the work is done in controlled areas, with

radiation monitors in operation to detect any leaks. The sealed containers are stored and shipped without emissions.

3.6.2 Basis for Risk Assessment

The doses and risks resulting from the operations of sealed source manufacturers are assessed using the actual emissions and site characteristics for the three manufacturers of self-luminous lights (Reference Facilities A, B, and C) and a model facility to represent the non-tritium source manufacturing facilities.

3.6.2.1 Emissions

The source term for the model radiation source facility is based on the arithmetic average of the emissions from four facilities that provided data. The model facility emits krypton-85, cobalt-60, americium-241, iridium-192, and californium-252, as shown in Table 3-17. The tritium lighting producers all provided effluent data for 1984, so no model facility is needed. Their emissions are also shown in Table 3-17. Since 1984, Reference Facility C has installed a catalytic recombiner system; therefore, current emissions are lower than the 1984 values.

Table 3-17. Effluent release rates (Ci/y) for sealed source manufacturers.

Radionuclide	Model Facility	Reference Facility A	Reference Facility B	Reference Facility C
H-3	-	3.4E+2	1.5E+3	2.2E+3
Co-60	3.2E-7	-	-	-
Ni-63	-	8.0E-6	-	-
Kr-85	2.4E-1	-	-	-
Ir-192	3.3E-6	-	-	-
Po-210	-	1.4E-4	-	-
Am-241	1.4E-7	6.1E-5	-	-
Cf-252	3.0E-9	-	-	-

3.6.2.2 Site Characteristics

The model facility was placed in an urban area. It was assumed to have a 6 m stack, 250 m away from the nearest resident. The tritium lighting manufacturers were assessed using actual site data. Meteorology was taken from nearby airports (Buffalo, NY; White Plains, NY; and Harrisburg, PA). Stack heights were set at 10 m. Nearby individuals are located 7,500 m, 400 m, and 150m, respectively, from the facilities. The New York sites were treated as urban sites; the Pennsylvania site, as rural.

3.6.3 Results of the Dose and Risk Assessment

Tables 3-18 and 3-19 show the results of the assessment for the model radiation source facility and all of the tritium lighting facilities. The highest estimated doses from non-tritium sealed source manufacturers are estimated to be to the endosteum and red marrow, both less than 1 mrem/y. The lifetime risk to nearby individuals is 8E-10. For the tritium lighting manufacturers, nearby individuals are estimated to receive doses on the order of 6 mrem/y and to have a lifetime fatal cancer risk of 2E-4.

To estimate the collective risk (deaths/year) from all sealed source manufacturers, the risk from the model was multiplied by 8 and added to the sum of the risks from the three tritium lighting facilities. This yields the total risk from this category of 2E-2 deaths/year. Table 3-20 presents this collective risk, and the number of people at risk, as a function of individual risk level.

Table 3-18. Estimated radiation dose rates from sealed source manufacturers.

Facility	Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Model Facility	Red Marrow	1.8E-4	1.3E-3
	Endosteum	2.2E-3	1.5E-2
	Remainder	1.0E-4	7.1E-4
Reference Facility A	Gonads	1.4E-1	1.0E+0
	Breast	1.3E-1	1.0E+0
	Red Marrow	1.7E-1	1.2E+0
	Lungs	1.4E-1	1.1E+0
	Endosteum	5.8E-1	3.4E+0
	Remainder	1.9E-1	1.4E+0
Reference Facility B	Gonads	5.6E-1	2.8E+1
	Breast	5.6E-1	2.8E+1
	Red Marrow	5.5E-1	2.8E+1
	Lungs	5.6E-1	2.8E+1
	Remainder	6.0E-1	3.3E+1
Reference Facility C	Gonads	5.4E+0	9.2E+0
	Breast	5.4E+0	9.2E+0
	Red Marrow	5.4E+0	9.1E+0
	Lungs	5.5E+0	9.2E+0
	Remainder	6.7E+0	1.1E+1

Table 3-19. Estimated fatal cancer risks from sealed source manufacturers.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
Model Facility	8E-10	8E-8
Reference Facility A	4E-6	4E-4
Reference Facility B	2E-5	1E-2
Reference Facility C	2E-4	4E-3

Table 3-20. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from sealed source manufacturers.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	550	4E-4
1E-6 to 1E-5	13,000	8E-4
< 1E-6	63,000,000	1E-2
Totals	63,000,000	2E-2

3.6.4 Supplemental Control Options and Costs

One of the sealed source manufacturers (Reference Facility C) is estimated to cause doses to nearby individuals in excess of 5 mrem/y. This exposure is due to emissions of kilocuries of tritium. Additional treatment of this effluent is possible.

In general, tritium is emitted as either tritiated water or tritium gas. Tritiated water can be removed from an effluent stream by using desiccant columns. These types of systems are very efficient. To remove tritium gas, however, requires that some type of catalytic recombiner be installed to transform the tritium gas into tritiated water. The costs of removal depend on exhaust flow rate. At low flow rates (approximately <40 m^3/min), it is estimated that the costs would be approximately $1.66 million to $7 million (Mo83). These costs are relatively high,

because this technology is not widely applied. There are only a handful of such installations, and each one is custom engineered.

Applying this supplemental control to Reference Facility C would cost approximately $1.7 to $7.0 million. The effectiveness of this system can only be estimated. Assuming a 99 percent reduction in emissions from Reference Facility C, the risk from this category would be reduced by half, to 1E-2 deaths/year.

3.7 NON-LWR FUEL FABRICATORS

3.7.1 General Description

Facilities in this category fabricate uranium fuel for research reactors and naval propulsion reactors. Three facilities making naval fuel were identified. One other facility manufactures only research reactor fuel. The process is similar to fabrication of power reactor fuel, where enriched UO_2 is formed into pellets, which are stacked inside tubes, and then bundled into fuel assemblies or cores. Fabrication procedures for naval fuel are classified.

Effluents to air are controlled using HEPA filters and/or gas scrubbers. The scrubbers are used to neutralize and remove the nitrogen oxides formed during HNO_3 pickling (chemical milling) operations at some facilities.

3.7.2 Basis for Risk Assessment

The doses and risks associated with this segment of the NRC-licensed source category are evaluated using actual emissions data and site characteristics for three of the four facilities.

3.7.2.1 Emissions

Recent (1987) data were obtained from operating reports for three facilities. The nuclides released that contribute the most to dose are uranium-234 and uranium-235. Release quantities of these and other isotopes are shown in Table 3-21.

3.7.2.2 Site Characteristics

Actual site and facility data were used in the risk assessment. Meteorology data were taken from nearby airports (Providence, RI; Knoxville, TN; and North San Diego, CA). Urban food supply assumptions were used, except in the analysis of the first facility which is in a rural area. The first facility releases effluents from a roof vent and was treated as an area source (525 m^2). The second and third facilities release through 35 m and 6 m stacks, respectively. The distances to the nearest residents are respectively 425 m, 350 m, and 750 m.

Table 3-21. Effluent release rates (Ci/y) for non-LWR fuel fabricators.

Radionuclide	Facility		
	Naval A	Naval B	Research
U-234	4.3E-5	3.4E-3	3.3E-6
U-235	1.2E-6	8.1E-5	1.5E-5
U-236	7.8E-8	1.2E-6	-
U-238	2.1E-9	5.7E-5	3.6E-6
Am-241	-	2.6E-8	-
Pu-238	-	4.2E-8	-
Pu-239	-	2.2E-8	-
Pu-240	-	2.0E-8	-
Pu-241	-	2.8E-6	-
Pu-242	-	2.9E-11	-
Th-232	-	-	4.0E-8
Ar-41	-	-	1.2E+0
Co-60	-	-	4.0E-5
Sr-90	-	-	4.8E-7
Y-90	-	-	4.8E-7
Cs-137	-	-	1.4E-4
I-131	-	-	1.0E-6

3.7.3 Results of the Dose and Risk Assessment

Off-site dose and risk were calculated for the three facilities from which release data were obtained. The results are shown in Tables 3-22 and 3-23. None of these facilities are estimated to cause nearby individuals doses greater than 1 mrem/y, and the lifetime fatal cancer risks to nearby individuals are less than 1E-6.

The estimated distribution of the fatal cancer risk to the regional populations from all non-LWR fuel fabricators is presented in Table 3-24. The deaths/year from the naval fuel fabricators were added and scaled up by 50 percent to account for the other facility of this type. The risks from the single research reactor fuel fabricator were then added. The result is the total risk of 2E-4 deaths/year.

3.7.4 Supplemental Control Options and Costs

Emissions from facilities in this segment of the NRC-licensed source category do not result in exposures or risks high enough to warrant a full evaluation of supplementary control options and costs. The well-proven technology of additional HEPA filtration systems could be employed to reduce emissions further.

Table 3-22. Estimated radiation dose rates from non-LWR fuel fabricators.

Facility	Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Naval Fuel A	Lungs	1.5E-1	2.5E-1
Naval Fuel B	Lungs	4.2E-1	4.7E+0
Research Fuel	Gonads	1.1E-2	5.1E-2
	Lungs	1.1E-1	5.8E-1
	Remainder	8.4E-3	4.1E-2

Table 3-23. Estimated fatal cancer risks from non-LWR fuel fabricators.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
Naval Fuel A	2E-7	6E-6
Naval Fuel B	7E-7	1E-4
Research Fuel	4E-7	3E-5

Table 3-24. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all non-LWR fuel fabricators.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	0	0
1E-6 to 1E-5	0	0
< 1E-6	8,200,000	2E-4
Totals	8,200,000	2E-4

3.8 SOURCE MATERIAL LICENSEES

3.8.1 General Description

Source material licensees are companies that handle relatively large amounts of thorium or uranium (non-enriched) during the manufacture of a product. The NRC licenses 12 facilities for the use of thorium (Mo88). Nine of them are currently using thorium. It is assumed that a similar number of facilities are active in Agreement States. This assumption is probably conservative because after contacting half of the Agreement States, only one active license for the use of thorium was located. Only four facilities in the United States hold source material licenses for the processing of depleted uranium.

The processes used by these licensees are varied. The facilities that emit thorium process low-thorium-content alloys into wire for lighting purposes. Other uses of thorium include scrap collection, glass creation, and lens coating. The depleted uranium is universally extruded into projectiles. In all of these processes, HEPA filters are used in series to reduce effluent levels. During extrusion and machining, lubricants are sprayed on the material to prevent particles from becoming airborne. The lubricants are then collected and disposed of as solid waste.

3.8.2 Basis for the Risk Assessment

A reference thorium facility and a reference uranium facility were used to evaluate the doses and risks of source material manufacturers.

3.8.2.1 Emissions

The emissions from source material licensees are split between facilities that have no emissions and facilities that emit approximately 3E-4 Ci/y of thorium or uranium. The thorium facilities are modeled by an existing facility that emits at this level. The uranium plants emit depleted uranium in the hundreds of microcuries. These plants are likewise modeled by a reference facility. Release rates are shown in Table 3-25.

Table 3-25. Effluent release rates for source material licensees.

Facility	Radionuclide (Ci/y)			
	U-234	U-235	U-238	Th-232
Uranium	2.7E-4	7.0E-6	2.7E-4	-
Thorium	-	-	-	3.0E-4

3.8.2.2 Site Characteristics

The two reference facilities were assessed using actual site and facility data. Meteorology data came from nearby airports (Cleveland, OH, and Bristol, TN). Effluent release heights are 10 m and 6 m, respectively. The nearest residents are located 100 m and 200 m away from the respective facility. Both facilities were assessed using urban food assumptions.

3.8.3 Results of the Dose and Risk Assessment

Tables 3-26 and 3-27 present the results of the dose and risk estimates for nearby individuals and the regional population for the reference facilities. Nearby individuals are estimated to receive doses to the lungs or the endosteum on the order of 3 mrem/y and to have a lifetime fatal cancer risk of about 4E-6.

Table 3-28 presents the estimated distribution of the fatal cancer risk to the regional populations from all source material licensees. This estimate was obtained by scaling the results for the reference facilities by the number of actual facilities. The total collective risk is estimated to be 1E-3 deaths/year.

Table 3-26. Estimated radiation dose rates from source material licensees.

Facility	Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Uranium	Lungs	2.7E+0	3.4E+0
Thorium	Lungs	2.6E+0	1.0E+1
	Endosteum	4.1E+0	1.6E+1

Table 3-27. Estimated fatal cancer risks from source material licensees.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
Uranium	4E-6	8E-5
Thorium	3E-6	1E-4

Table 3-28. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all source material licensees.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	0	0
1E-6 to 1E-5	0	0
< 1E-6	24,000,000	1E-3
Totals	24,000,000	1E-3

3.8.4 Supplemental Control Options and Costs

Emissions from facilities in this segment of the NRC-licensed source category do not result in exposures or risks high enough to warrant an evaluation of supplementary control options and costs. The well-proven technology of additional HEPA filtration systems could be employed to reduce emissions further.

3.9 LOW-LEVEL WASTE INCINERATORS

3.9.1 General Description

Airborne effluents from low-level waste handling and disposal arise primarily from waste incineration. The practice of evaporating disposal site liquids has ceased, so this is no longer a source of releases to air. Incineration is done mainly by large research laboratories and hospitals. About 100 such incinerators are operating in the United States.

The older incinerators usually release directly to the atmosphere. The newer ones are designed with sophisticated effluent control systems, including afterburners, venturi scrubbers, and gas scrubbers (e.g., NaOH and water). Since the newer units have much higher capacities (e.g., 1,000 lb/hr), they are replacing the older units.

3.9.2 Basis for the Risk Assessment

The dose and risk assessment is based on a large reference facility to obtain doses and risks to nearby individuals and a model facility with average emissions to obtain collective doses and risks.

3.9.2.1 Emissions

Effluent data were obtained from the CRCPD survey (1987) for 35 incinerators. Nearly all reported releases of tritium and carbon-14. Nine or fewer facilities reported releases of sulfur-35, chromium-51, iodine-125, and phosphorus-32. A model facility was created using the average releases of these nuclides. An actual facility reporting the largest releases of the above nuclides was also modeled. Table 3-29 presents the source terms used in the assessment.

3.9.2.2 Site Characteristics

The model and large incinerator were both placed at a suburban site for the risk assessment. They both have a stack height of 35 m and a thermal release rate of 2.2E+5 cal/second. The nearest resident is located 300 m away. Both assessments used meteorological data from a nearby airport.

Table 3-29. Effluent release rates (Ci/y) for low-level waste disposal facilities.

Radionuclide	Model Facility	Reference Facility
H-3	1.0E-1	1.3E+0
C-14	5.0E-2	1.5E+0
P-32	7.0E-2	1.4E-1
S-35	1.0E-1	8.7E-1
Cr-51	1.0E-2	5.0E-2
Se-75	-	1.0E-3
I-125	1.5E-2	9.0E-2

3.9.3 Results of the Dose and Risk Assessment

Assessments for the model incinerator and the large reference facility indicate that nearby individuals receive doses less than 1 mrem/y and have lifetime fatal cancer risks of less than 1E-6. The results are shown in Tables 3-30 and 3-31.

Table 3-32 presents the estimated distribution of the fatal cancer risk to the regional populations from all low-level waste disposal facilities. This estimate was obtained by scaling up the risks from the model facility by a factor of 100. This gives a risk of 1E-3 deaths/year from all incinerators.

3.9.4 Supplemental Control Options and Costs

Emissions from low-level waste disposal facilities do not result in exposures or risks high enough to warrant an evaluation of supplementary control options and costs.

Table 3-30. Estimated radiation dose rates from low-level waste disposal facilities.

Facility	Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Model Facility	Red Marrow	8.0E-4	6.7E-2
	Lungs	4.6E-4	2.3E-2
	Endosteum	9.4E-4	7.7E-2
	Remainder	2.5E-4	2.5E-2
Reference Facility	Gonads	1.2E-2	6.0E-1
	Breast	1.4E-2	7.5E-1
	Thyroid	1.1E-1	1.1E+1
	Remainder	8.1E-3	5.5E-1

Table 3-31. Estimated fatal cancer risks from low-level waste disposal facilities.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
Model Facility	1E-8	1E-5
Reference Facility	3E-7	2E-4

Table 3-32. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all low-level waste disposal facilities.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	0	0
1E-6 to 1E-5	0	0
< 1E-6	240,000,000	1E-3
Totals	240,000,000	1E-3

3.10 NON-DOE FEDERAL FACILITIES

3.10.1 General Description

This category includes Department of Defense (DOD) facilities. Other non-DOE federal facilities, such as Veterans Administration hospitals and NASA research laboratories, are included in the evaluations presented in Sections 3.2, 3.4, and 3.5. Federal facilities operated by the DOE are discussed in Chapter 2.

This category is made up of two groups of DOD facilities. The first and largest group consists of nuclear shipyards and naval bases. The second consists of DOD research reactors. There are 13 active shipyards and bases. Seven are on the east coast, five are on the west coast, and one is in Hawaii. These facilities refuel and service the Navy's nuclear fleet. Most of the radioactive wastes are in solid form. According to the Navy, there are no significant discharges of airborne radioactivity (Ma88). Exhaust air from waste handling buildings is passed through HEPA filters to control emissions.

The DOD operates two unlicensed research reactors, at Aberdeen, MD, and White Sands, NM. Operations and effluent control are essentially the same as for the research reactors described in Section 3.5.

3.10.2 Basis for the Risk Assessment

A single model facility is used to estimate the doses and risks from this segment of the NRC-licensed source category, as the magnitudes of the releases from both the DOD reactors and the shipyards are comparable.

3.10.2.1 Emissions

Effluent monitoring at DOD shipyards and bases reveals few measurable nuclides (Ma88). However, the Navy has estimated maximum releases, based on many years of monitoring data. These releases are primarily noble gases and cobalt-60 (see Table 3-33). Since the magnitude of the releases from DOD research reactors (Co83) are comparable to the maximum releases estimated by the Navy, the emissions for the single model facility represent both types of actual DOD sites.

3.10.2.2 Site Characteristics

For purposes of the risk assessment, the model DOD facility was placed at the site of an actual west coast shipyard. Meteorological data came from that same shipyard. The release height was assumed to be 15 m, and the distance to the nearest residents is 1,500 m. Rural food supply assumptions were used.

Table 3-33. Effluent release rates (Ci/y) for DOD facilities.

Radionuclide	Model Facility
H-3	1.0E-3
C-14	1.0E-1
Co-60	1.0E-3
Kr-83m	2.0E-2
Kr-85m	2.4E-2
Kr-87	5.0E-2
Kr-88	2.0E-2
Xe-131m	5.0E-3
Xe-133m	1.0E-2
Xe-133	2.1E-1
Xe-135	2.5E-1
Ar-41	4.1E-1

3.10.3 Results of the Dose and Risk Assessment

The doses and risks from the model facility are shown in Tables 3-34 and 3-35. Table 3-36 presents the estimated distribution of the fatal cancer risk to the regional populations from all DOD facilities. This estimate was made by multiplying the risks estimated for the model facility by a factor of 12. This factor is obtained by considering the shipyards and bases that are in proximity (e.g., Newport News and Norfolk, VA) as single facilities. The collective population risk from all DOD facilities is estimated to be 1E-3 deaths/year.

3.10.4 Supplementary Control Options and Costs

Emissions from facilities in this segment of the NRC-licensed and non-DOE Federal source category do not result in exposures or risks high enough to warrant an evaluation of supplementary control options and costs.

Table 3-34. Estimated radiation dose rates from DOD facilities.

Facility	Organ	Nearby Individuals (mrem/y)	Regional Population (person-rem/y)
Model Facility	Gonads	1.1E-2	2.5E-1
	Breast	1.0E-2	2.4E-1
	Red Marrow	8.9E-3	2.3E-1
	Lungs	1.1E-2	2.3E-1
	Remainder	9.0E-3	2.1E-1

Table 3-35. Estimated fatal cancer risks from DOD facilities.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
Model Facility	2E-7	8E-5

Table 3-36. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all DOD facilities.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	0	0
1E-5 to 1E-4	0	0
1E-6 to 1E-5	0	0
< 1E-6	64,000,000	1E-3
Totals	64,000,000	1E-3

3.11 SUMMARY OF THE COLLECTIVE RISKS FROM ALL FACILITIES

The population risks calculated for each of the nine subcategories were combined to obtain an estimate of the total deaths/year resulting from emissions from all NRC-licensed facilities. The results are presented in Table 3-37. Because the regional population extends 80 km from each facility, individuals are exposed to emissions from more than a single facility. Thus, the combined regional population obtained by summing the results of the individual estimates exceeds the total population of the United States. The number of persons at risk shown in Table 3-37 is therefore limited to 240 million persons, the population of the United States. The total risk from this category, 2E-1 deaths/year, was not adjusted to account for this overlap, since virtually all the risk is incurred by individuals living close to each facility.

The largest contributors to the collective risk are research reactors and hospitals, estimated to cause 4E-2 and 6E-2 deaths/year, respectively. Although hospitals have relatively low emissions, there are many of them. The next highest contributors to collective risk are radiopharmaceutical manufacturers, estimated to cause 2E-2 deaths/year, and research laboratories (8E-3 deaths/year). Like hospitals, research

laboratories have low emissions, but their large number results in small risks to many persons.

Table 3-37. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from all NRC-licensed facilities.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	*	*
1E-5 to 1E-4	5,000	2E-3
1E-6 to 1E-5	780,000	3E-2
< 1E-6	239,000,000	1E-1
Totals	240,000,000	2E-1

*Results indicate there may be some individuals at this risk level, but insufficient information is available to quantify either the number of persons or the deaths/year.

With respect to individual risk, the maximum value of 2E-4 lifetime fatal cancer risk is estimated for both a radiopharmaceutical manufacturer and a sealed source manufacturer. A research reactor and another sealed source manufacturer account for the next highest individual risk, estimated to be 2E-5.

These estimates of deaths per year in the regional populations and maximum lifetime risks to nearby individuals must be viewed with caution. Only a limited number of the 6,000 facilities in this category could be evaluated, and the evaluations rest on unverified emissions data provided by the facilities. While the methodology attempted to evaluate the facilities with the greatest potential risk, the lack of emissions data for so many of the facilities makes it impossible to state with certainty that this goal was achieved. Thus, there may be NRC-licensed and non-DOE Federal facilities causing greater doses and risks than those that have been estimated in this evaluation.

3.12 REFERENCES

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4. URANIUM FUEL CYCLE FACILITIES

4.1 INTRODUCTION

The uranium fuel cycle includes uranium mills, uranium hexafluoride conversion facilities, uranium enrichment facilities, light-water reactor fuel fabricators, light-water power reactors, and fuel reprocessing plants. With the exception of the uranium enrichment facilities that are owned by the Federal government and operated by contractors under the supervision of the Department of Energy (DOE), these facilities are licensed by the Nuclear Regulatory Commission (NRC) or the Agreement States. Releases of radioactive materials from these facilities during normal operation are subject to the limits established by 40 CFR 190. 40 CFR 190 limits the exposure to any member of the general public from radionuclides released to air or water to 25 mrem/y to the whole body or to any organ except the thyroid, which is limited to 75 mrem/y. In addition, the NRC requires releases of radioactive materials to be as low as reasonably achievable (ALARA) below these regulatory limits.

As part of the current rulemaking, the EPA has performed a dose and risk assessment of current airborne emissions from uranium fuel-cycle facilities. The results of the dose and risk assessment indicate that airborne emissions from operating uranium mills cause greater doses and risks than those from the uranium conversion, fuel fabrication, and light-water reactor sectors of the fuel cycle.

4.1.1 Previous Evaluations

The potential public health impacts of the release of radioactive materials into ambient air from the uranium fuel cycle have been comprehensively evaluated. The EPA has prepared a series of reports describing this evaluation. These reports include:

U.S. Environmental Protection Agency, Environmental Analysis of the Uranium Fuel Cycle - Part I - Fuel Supply, EPA 520/9-73-003C, Office of Radiation Programs, Washington, D.C., 1973;

U.S. Environmental Protection Agency, Environmental Analysis of the Uranium Fuel Cycle - Part II, Nuclear Power Reactors, EPA 520/9-73-003C, Office of Radiation Programs, Washington, D.C., 1973;

U.S. Environmental Protection Agency, A Radiological Emissions Study at a Fuel Fabrication Facility, EPA 520/5-77-004, Office of Radiation Programs, Washington, D.C., 1978;

U.S. Environmental Protection Agency, Radiological Impact Caused by Emission of Radionuclides into Air in the United States, EPA 520/7-79-006, Washington, D.C., 1979;

U.S. Environmental Protection Agency, Final Environmental Impact Statement for Remedial Action Standards for Inactive Uranium Processing Sites, EPA 520/4-82-013-1, October 1982;

U.S. Environmental Protection Agency, Final Environmental Impact Statement for Standards for the Control of Byproduct Materials from Uranium Ore Processing, EPA 520/1-83-008-1, September 1983;

U.S Environmental Protection Agency, Radionuclides, Background Information Document for Final Rules, EPA 520/1-84-022, Office of Radiation Programs, October 1984; and

U.S. Environmental Protection Agency, Final Rule for Radon-222 Emissions from Licensed Uranium Mill Tailings, Background Information Document, EPA 520/1-86-009, August 1986.

4.1.2 Scope of the Evaluation

The segments of the uranium fuel cycle addressed in this chapter include:

1. Uranium mills and their associated tailings piles;
2. Uranium conversion facilities;
3. Fuel fabrication facilities; and
4. Nuclear power facilities.

Each of these categories is addressed in the following sections, which include a general description of each facility's characteristics, processes, emission controls, radionuclide emissions, and predicted radiation dose equivalent rates and health risks to nearby individuals and the populations within 80 kilometers of these facilities. In addition, for categories with the highest exposures, supplementary control options and costs are presented here.

The assessment of doses and risks shows that particulate releases from operating uranium mills cause some members of the general public to receive organ dose equivalents greater than 25 mrem/y; for nearby individuals, estimates of the dose equivalent to the lungs and the endosteum are as high as 120 and 85 mrem/y, respectively. The nearby individuals at greatest risk are estimated to have a lifetime fatal cancer risk of 2E-4. The

basis for these estimates and the detailed results are presented in the following sections.

The assessment of uranium mills addresses only particulate emissions. Radon emissions from the tailings are addressed in Chapter 9. The uranium enrichment plants are included in the assessment of DOE facilities (see Chapter 2). As there are no operable fuel reprocessing plants in the United States, and since reprocessing is prohibited under current policies, this segment of the uranium fuel cycle has not been evaluated. High-level waste disposal facilities are addressed in Chapter 5.

4.2 URANIUM MILLS

4.2.1 General Description

4.2.1.1 Uranium Mill Operations in the United States

Uranium mills extract uranium from ores which contain only 0.01 to 0.3 percent U_3O_8. Uranium mills, typically located near uranium mines in the western United States, are usually in areas of low population density. The product of the mills is shipped to conversion plants, where it is converted to volatile uranium hexafluoride (UF_6) which is used as feed to uranium enrichment plants.

As of December 1988, of 27 uranium mills in the United States licensed by the NRC or Agreement States, 4 were operating, 8 were on standby, 14 were being decommissioned, and 1 had been built but never operated. The 8 mills on standby could resume operations, but the 14 mills that are being decommissioned will never operate again. The status of each mill is presented in Table 4-1. The status descriptions used in this document are not necessarily the same as the license definitions. Umetco's Uravan mill is listed as on standby; however, since the mill's tailings impoundment is being reclaimed, the mill is considered to be decommissioned for the purpose of this assessment.

The operating mills have a capacity of 9,600 tons of ore per day. The number of operating mills is down considerably from 1981, when 21 mills were processing approximately 50,000 tons of ore per day.

4.2.1.2 Process Description

The mined ore is stored on pads prior to processing. Crushing and grinding and a chemical leaching process separate the uranium from the ore. The uranium product is recovered from the leach solution and then dried and packaged. The waste product (mill tailings) is piped as a slurry to a surface impoundment area (tailings pile).

Table 4-1. Uranium mills licensed by the U.S. Nuclear Regulatory Commission as of December 1988.

Licensee	Common Name	Location	Rated Capacity[a] (t ore/d)	Status[b]	Process[c]
American Nuclear Corp.	-	Gas Hills, WY	950	3	1,5
Anaconda	Bluewater	Bluewater, NM	6000	3	1,3
Atlas Minerals	Moab	Moab, UT	1400	3	2,3
Bear Creek Uranium Co.	Bear Creek	Converse Co. WY	2000	3	1,3
Bokum Resources	-	Marquez, NM	2000	4	1,3
Chevron Resources Co.	Panna Maria	Panna Maria, TX	2500	1	1,3
Conoco-Pioneer	-	Falls City, TX	3400	3	1,3
Cotter Corp.	Canon	Canon City, CO	1200	2	1,3
Dawn Mining Co.	Dawn	Ford, WA	450	3	1,3
Exxon	Ray Point	Ray Point, TX	-	3	-
Exxon Minerals	Highland	Converse Co., WY	3200	3	1,3
Homestake Mining Co.	Homestake	Grants, NM	3400	1	4,6
BP American	L-Bar	Seboyeta, NM	1600	3	-
Minerals Exploration	Sweetwater	Sweetwater Co., WY	3000	2	1,3
Pathfinder Mines	Lucky Mc	Gas Hills, WY	2500	2	1,3
Pathfinder Mines	Shirley Basin	Shirley Basin, WY	1700	1	1,3
Petrotomics	Petromics	Shirley Basin, WY	1500	3	1,3
Plateau Resources	Shootaring	Shootaring Cnyn, UT	750	2	1,3
Quivira	Ambrosia	Ambrosia Lake, NM	-	2	-
Rio Algom	La Sal	La Sal, UT	750	2	4,6
TVA	Edgemont	Edgemont, SD	-	3*	-
Umetco Minerals Corp.	Gas Hills	Gas Hills, WY	1400	3	1,5
Umetco Minerals Corp.	White Mesa	Blanding, UT	2000	1	1,7
Umetco Minerals Corp.	Uravan	Uravan, CO	1300	2**	1,3
UNC Mining & Milling	Church Rock	Church Rock, NM	3000	3	1,3
Western Nuclear Inc.	Split Rock	Jeffrey City, WY	1700	3	1,3
Western Nuclear Inc.	Sherwood	Wellpinit, WA	2000	2	1,3

Status Codes:
1 = Facility Operating
2 = Facility on Standby
3 = Facility Decommissioned or Being Decommissioned
4 = Facility Built, Never Operated

Process Codes:
1 = Acid leach
2 = Alkaline leach
3 = Solvent extraction
4 = Carbonate leach
5 = Eluex
6 = Caustic precipitation
7 = Column ion exchange

Data Sources:
(a) Tons of ore/day (Jo81).
(b) Personal communication with Dale Smith, USNRC, Denver, Colorado.
(c) From Ri81.

* Decommissioning and long-term stabilization complete.
** Per public comment by Umetco, the mill is being maintained on standby although the tailings impoundment is being reclaimed.

Radioactive materials released to the air during these operations include natural uranium and thorium and their respective decay products (e.g., radium, lead, radon). These radionuclides, with the exception of radon, are released as particulates.

4.2.1.2.1 Ore Storage

Ore is hauled from the mine in trucks. A minimum 10-day supply of ore is kept on storage pads, which are several hectares in area. The ore is transferred to the mill crushing unit via front-end loaders or bulldozers. Although the ore is usually moist upon receipt at the storage pad, it can become dry during storage. The transfer operations, as well as wind erosion, result in dust formation and release of radioactive material in particulate form.

4.2.1.2.2 Milling

The process of extracting uranium from ore starts with crushing and grinding. The ores are crushed dry, but water is added during the grinding process. Some of the newer mills use a one-step wet process called semi-autogenous grinding which eliminates the dry ore crushing step.

The next step consists of leaching uranium out of the ore and separating the uranium product from the leach solution. There are two basic leaching processes: acid leaching for ores with low lime content, and alkaline or carbonate leaching for ores with high lime content. The leach solution is then chemically treated to remove the uranium product. Most mills that use the acid leaching process follow with solvent extraction, a process where the uranium product is separated from the solution by an organic solvent and is then separated from the solvent by a stripping and precipitation operation. The mills that use the alkaline or carbonate leaching process add a caustic to the leach solution, resulting in the precipitation of sodium diuranate. In both cases, the product is dried in large ovens and packaged in 55-gallon drums.

The steps that generate significant radioactive emissions are the dry operations: crushing, drying, and packaging. The intermediate stages are carried out wet in enclosed vessels and do not produce significant amounts of airborne emissions.

4.2.1.2.3 Tailings

After the uranium product is separated from the ore in the leaching process, the residual ore is pumped as a slurry to a tailings impoundment area. A tailings pile, typically about 100 hectares in area, is surrounded by an embankment of impervious material. The liquid portion of the slurry is partially recovered and recycled by some mills and is allowed to evaporate at other mills. The solid tailings are made up of a

sand fraction (particles from 38 to 200 mesh) and a slime fraction (particles smaller than 200 mesh).

An active tailings pile contains wet and dry areas. The slurry feed pipe is moved around the impoundment area to keep the pile level; therefore, the pile has a pond area where the slurry is fed while the rest of the pile is drying out.

As sections of the pile dry, the tailings become a source of windblown dust. The slime component, the most likely to become airborne because of its small particle size, contains uranium concentrations twice as high as the sands (NRC79).

4.2.1.3 Existing Emission Controls

4.2.1.3.1 Ore Storage

Dust from ore storage pads can be controlled by the use of windbreaks and water sprays. Windbreaks are concrete or wood fences around the pile which reduce the amount of wind blowing across the pile. This reduces the drying effect of the wind, as well as reducing the tendency of the wind to pick up dust.

Ore piles with a moisture content of 4 percent or more do not cause dust problems (NRC79). Spraying the pile increases the moisture content of the ore. A tank truck with pumps and hoses can be used for spraying.

4.2.1.3.2 Milling

Dust is controlled during the crushing process by placing air exhaust hoods at the crusher, screens, and transfer points. The exhaust air passes through a dust collector before it is discharged to the atmosphere through a roof vent. As indicated earlier, if a semi-autogenous grinding process is used, then the dry crushing step is eliminated and essentially no dust is emitted.

The off-gas from the drying oven passes through a dust separation system before discharge to the roof vent. Air exhaust hoods are placed in the packaging area, and the exhaust is passed through a dust collector before being vented.

The primary method of removing dust from the exhaust gas is the wet scrubber. Wet scrubbers remove dust particles by impacting them with water droplets. The most common type of wet scrubber is the orifice scrubber, which has a removal efficiency of 93.6 percent. Also common is the impingement scrubber, which has a removal efficiency of 97.9 percent. The venturi scrubber, used infrequently, has a removal efficiency of 99.5 percent but requires more energy to operate than the other two scrubber types. The removal efficiencies presented are those cited by the NRC for these applications (NRC79).

Baghouses are frequently used to remove dust from the crushing and packaging area exhaust. The exhaust air is passed through bag filters made of woven or felted material. Baghouses have a rated removal efficiency of 99.9 percent. They are not suitable for cleaning the dryer off-gas because of the high temperature and moisture content.

4.2.1.3.3 Tailings

Control of dust from a tailings pile is similar to control of dust from the ore storage pad. The tailings pile can be kept wet by truck spraying or by discharging the slurry from multiple discharge points instead of one point.

An alternative method of dust control for tailings surfaces that are not being added to or disturbed is to put a chemical stabilizer on the surface of the pile. Some stabilizers mix with the tailings to form a crust. Other materials, such as asphalt sprays, form a thin film on the pile surface. Both methods are temporary and require annual maintenance.

4.2.2 Basis for the Dose and Risk Assessment of Uranium Mills

The following sections describe the basis for the site-specific and model facilities used to assess the airborne releases of radionuclides from uranium mills. Information on the source term, meteorological, and demographic data assumed are also presented. Detailed information on the parameters supplied to the AIRDOS/DARTAB/RADRISK computer codes is presented in Appendix A. Site-specific source term, meteorological, and demographic data were supplied as input to the assessment codes for the four operating mills and for six of the seven mills on standby. Cotter Corporation's Canon City mine, which is on standby, currently has no dry tailings piles and therefore was not included in the assessment. A generic model mill was used for the assessment of doses and risks from tailings piles of mills that are either decommissioned or undergoing decommissioning. Outputs of the codes include estimates of: dose equivalents to the most exposed individuals (mrem/y); lifetime fatal cancer risk to the most exposed individuals; dose equivalents to the regional (0-80 km) population (person-rem/year); and the number of cancer deaths in the regional population per year of operation (deaths/year).

4.2.2.1 Radionuclide Emissions

The magnitude of releases from uranium mills differs for operating and shutdown facilities. Therefore, in addition to measured process releases reported to the NRC, models were developed to represent windblown particles from active tailings and windblown releases from dry tailings piles where operations have ceased and final stabilization has not yet occurred.

4.2.2.1.1 Operational Experience and Projected Future Emissions

The drying area and the crushing area are the major sources of process releases at a typical plant. Ninety percent of the uranium-234 and uranium-238 released come from the dryer area at the end of the process. On the other hand, thorium-230 and radium-226 emissions result primarily from operations, such as crushing, that occur at the beginning of the process.

Although the number of operating uranium mills has decreased sharply over the last decade, the demand for yellowcake has been steadily increasing as more nuclear power plants have come on line. Yellowcake from foreign sources has supplied an increasing percentage of demand. However, the number of operating mills is expected to stabilize or perhaps even increase slightly in the near future. Radionuclide releases from uranium milling operations should be proportional to the quantity of uranium ore processed.

4.2.2.1.2 Development of Source Term for Assessments

The source terms for operating uranium mills and mills on standby include particulate radionuclides released to air from process exhausts and those blown from the dry areas of the tailings impoundments. The source terms used in the assessment for operating mills, mills on standby, and a generic inactive tailings impoundment, which was used to model decommissioned mills, are presented in Table 4-2.

The source terms presented here for the operating facilities differ from those presented in the draft document due to the use of more current information concerning the total area of wet and dry tailings and the concentration of radium-226 in the tailings at each of the facilities. Also, source terms for mills on standby are now presented wereas they were not originally included in the draft document.

The release rates (Ci/y) for process exhausts are based on measurements of natural uranium, thorium-230, and radium-226. These data were obtained for three of the four mills from the semi-annual environmental monitoring reports submitted by the mills to the Nuclear Regulatory Commission. Whereas Panna Maria was not included in the original assessment due to an inability to obtain measured process release rate data, the mill has now been included using, information obtained from Chevron Resources Company.

Tailings pile emissions are not measured by the mill operators, since the size of the tailings impoundments makes measurement of windblown releases impractical. Therefore, the release rates (Ci/y) from the tailings presented in Table 4-2 were calculated using the methodology presented in NRC's Regulatory Guide 3.59, and the areas of dried tailings and

average radium concentrations shown in Table 4-3, using dusting factors appropriate for the site meteorology and tailings pile characteristics presented in EPA86.

The analysis includes consideration of the predominant periods of tailings resuspension and dispersion during episodes of high wind speed. No data were found showing particle size distributions for process dusts. Particle size distributions for tailings dusts show that approximately 30 percent of the particles are in the respirable size range of 10 microns or less (NRC80). Only the respirable fraction of the total dusts was included in the assessment, and an activity median aerodynamic diameter (AMAD) of 3.0 microns, consistent with the data for tailings dusts, was assumed. Data on lung clearance classifications for windblown tailings could not be found. Therefore, the default values recommended by the ICRP were used for all radionuclides blown from the tailings.

Tailings pile release rates for the Canon City mill are not shown in Table 4-2 since the site currently has no dried tailings impoundments. Tailings release rates for Umetco Minerals Corporations's Uravan mill are also not included. Although the Uravan mill is on standby, the tailings impoundment is being reclaimed. Thus, for the purposes of this assessment, the Uravan mill is considered to be decommissioned and is therefore modeled using the model inactive tailings impoundment.

The lung clearance classifications for uranium from process exhausts are based on solubility studies of yellowcake in simulated lung fluid (Co74, De79, De82, and Ka80). The classifications used for thorium, radium, lead, and polonium are the default values recommended by the ICRP (ICRP66).

The NRC has calculated emissions from tailings piles from several specific mills. These values range from 2.0E-4 to 2.7E-3 Ci/y for uranium-238/uranium-234, 3.3E-3 to 5.2E-2 Ci/y for thorium-230, and 3.2E-3 to 5.5E-2 Ci/y for radium-226 (EPA79).

Annual radionuclide releases from tailings of the model inactive mill, for which permanent stabilization has not been performed, are also presented in Table 4-2. Methodology for calculating these emissions was the same as that for calculating emissions from tailings of active mills. The higher rate of emissions for the inactive tailings pile is attributable to the reduced moisture content of the inactive tailings and the increased pile size.

Table 4-2. Source terms for uranium milling.

Radionuclide	Lung Clearance	AMAD	Release Rate (Ci/y) Process Exhaust	Tailings
CHEVRON'S PANNA MARIA MILL[a]				
U-238	Y	3.0	1.9E-3	-[b][c]
U-238	D	3.0	1.9E-3	
U-235	Y	3.0	1.1E-5	-
U-235	D	3.0	1.1E-5	
U-234	Y	3.0	1.9E-3	-
U-234	D	3.0	1.9E-3	
Th-230	Y	3.0	9.6E-5	-
Ra-226	W	3.0	3.8E-6	-
Pb-210	D	3.0	3.8E-6	-
Po-210	W	3.0	3.8E-6	-
HOMESTAKE'S HOMESTAKE MILL				
U-238	Y	3.0	1.7E-1	1.0E-4[c]
U-238	D	3.0	1.7E-1	
U-235	Y	3.0	8.3E-4	7.1E-7[c]
U-235	D	3.0	8.3E-4	
U-234	Y	3.0	1.7E-1	1.0E-4[c]
U-234	D	3.0	1.7E-1	
Th-230	Y	3.0	4.3E-2	1.0E-3[c]
Ra-226	W	3.0	3.9E-2	1.0E-3[c]
Pb-210	D	3.0	3.9E-2	1.0E-3[c]
Po-210	W	3.0	3.9E-2	1.0E-3[c]
MINERALS EXPLORATION'S SWEETWATER MILL[a]				
U-238	Y	3.0	-[d]	4.3E-3
U-238	D	3.0	-	
U-235	Y	3.0	-	3.0E-5
U-235	D	3.0	-	
U-234	Y	3.0	-	4.3E-3
U-234	D	3.0	-	
Th-230	Y	3.0	-	4.3E-2
Ra-226	W	3.0	-	4.3E-2
Pb-210	D	3.0	-	4.3E-2
Po-210	W	3.0	-	4.3E-2

(a) Source term added to those originally included in draft document to reflect data obtained during comment period.
(b) Panna Maria currently has no dry tailings impoundments.
(c) Changes in source terms with respect to the draft document reflect information on tailings areas and radium-226 concentrations obtained during comment period.
(d) Mill is currently on standy.

Table 4-2. Source terms for uranium milling (continued).

Radionuclide	Lung Clearance	AMAD	Release Rate (Ci/y) Process Exhaust	Tailings
PATHFINDER'S LUCKY MC MILL[a]				
U-238	Y	3.0	-[b]	1.1E-3
U-238	D	3.0	-	
U-235	Y	3.0	-	8.0E-6
U-235	D	3.0	-	
U-234	Y	3.0	-	1.1E-3
U-234	D	3.0	-	
Th-230	Y	3.0	-	1.1E-2
Ra-226	W	3.0	-	1.1E-2
Pb-210	D	3.0	-	1.1E-2
Po-210	W	3.0	-	1.1E-2
PATHFINDER'S SHIRLEY BASIN MILL				
U-238	Y	3.0	1.1E-2	5.4E-3[c]
U-238	D	3.0	1.1E-2	
U-235	Y	3.0	8.0E-5	3.9E-5[c]
U-235	D	3.0	8.0E-5	
U-234	Y	3.0	1.1E-2	5.4E-3[c]
U-234	D	3.0	1.1E-2	
Th-230	Y	3.0	1.9E-4	5.4E-2[c]
Ra-226	W	3.0	5.9E-4	5.4E-2[c]
Pb-210	D	3.0	5.9E-4	5.4E-2[c]
Po-210	W	3.0	5.9E-4	5.4E-2[c]
PLATEAU RESOURCES' SHOOTARING MILL[a]				
U-238	Y	3.0	-[b]	2.0E-4
U-238	D	3.0	-	
U-235	Y	3.0	-	1.4E-6
U-235	D	3.0	-	
U-234	Y	3.0	-	2.0E-4
U-234	D	3.0	-	
Th-230	Y	3.0	-	2.0E-3
Ra-226	W	3.0	-	2.0E-3
Pb-210	D	3.0	-	2.0E-3
Po-210	W	3.0	-	2.0E-3

(a) Source term added to those originally included in draft document to reflect data obtained during comment period.
(b) Mill is currently on standby.
(c) Changes in source terms with respect to the draft document reflect information on tailings areas and radium-226 concentrations obtained during comment period.

Table 4-2. Source terms for uranium milling (continued).

Radionuclide	Lung Clearance	AMAD	Release Rate (Ci/y) Process Exhaust	Tailings
QUIVIRA'S AMBROSIA LAKE MILL[a]				
U-238	Y	3.0	-[b]	1.1E-3
U-238	D	3.0	-	
U-235	Y	3.0	-	7.5E-6
U-235	D	3.0	-	
U-234	Y	3.0	-	1.1E-3
U-234	D	3.0	-	
Th-230	Y	3.0	-	1.1E-2
Ra-226	W	3.0	-	1.1E-2
Pb-210	D	3.0	-	1.1E-2
Po-210	W	3.0	-	1.1E-2
RIO ALGOM'S LA SAL MILL				
U-238	Y	3.0	2.8E-2	-[c][d]
U-238	D	3.0	2.8E-2	
U-235	Y	3.0	2.1E-4	-
U-235	D	3.0	2.1E-4	
U-234	Y	3.0	2.8E-2	-
U-234	D	3.0	2.8E-2	
Th-230	Y	3.0	1.0E-4	-
Ra-226	W	3.0	2.8E-4	-
Pb-210	D	3.0	3.3E-4	-
Po-210	W	3.0	3.3E-4	-
UMETCO'S WHITE MESA MILL				
U-238	Y	3.0	2.1E-2	1.4E-4[c]
U-238	D	3.0	2.1E-2	
U-235	Y	3.0	1.5E-4	1.1E-6[c]
U-235	D	3.0	1.5E-4	
U-234	Y	3.0	2.1E-2	1.4E-4[c]
U-234	D	3.0	2.1E-2	
Th-230	Y	3.0	4.9E-4	1.4E-3[c]
Ra-226	W	3.0	4.8E-4	1.4E-3[c]
Pb-210	D	3.0	1.2E-3	1.4E-3[c]
Po-210	W	3.0	1.2E-3	1.4E-3[c]

(a) Source term added to those originally included in draft document to reflect data obtained during comment period.
(b) Mill is currently on standby.
(c) Changes in source terms with respect to the draft document reflect information on tailings areas and radium-226 concentrations obtained during comment period.
(d) La Sal currently has no dry tailings impoundments.

Table 4-2. Source terms for uranium milling (continued).

Radionuclide	Lung Clearance	AMAD	Release Rate (Ci/y) Process Exhaust	Tailings
WESTERN NUCLEAR INC.'S SHERWOOD MILL[a]				
U-238	Y	3.0	-[b]	1.0E-3
U-238	D	3.0	-	
U-235	Y	3.0	-	7.1E-6
U-235	D	3.0	-	
U-234	Y	3.0	-	1.0E-3
U-234	D	3.0	-	
Th-230	Y	3.0	-	1.0E-2
Ra-226	W	3.0	-	1.0E-2
Pb-210	D	3.0	-	1.0E-2
Po-210	W	3.0	-	1.0E-2
MODEL INACTIVE TAILINGS PILE[c]				
U-238	Y	3.0		8.0E-3
U-235	Y	3.0		5.8E-5
U-234	Y	3.0		8.0E-3
Th-230	Y	3.0		8.0E-2
Ra-226	W	3.0		8.0E-2
Pb-210	D	3.0		8.0E-2
Po-210	W	3.0		8.0E-2

(a) Source term added to those originally included in draft document to reflect data obtained during comment period.
(b) Mill is currently on standby.
(c) After closure, prior to stabilization.

Table 4-3. Areas of the tailings impoundments at uranium mills and average radium-226 concentrations.(a)

Mill	Total Area (acres/ha)	Wet Area (acres/ha)	Dry Area (acres/ha)	Radium-226 (pCi/g)
New Mexico				
Ambrosia Lake				
- Secondary	121/49	13/5	108/44	237
- Lined Ponds	280/113	162/66	118/47	22
Homestake	210/85	140/57	70/28	300
Texas				
Panna Maria	160/65	160/65	0/0	198
Utah				
La Sal	93/38	93/38	0/0	420
Shootaring	7/3	3/1	4/2	280
White Mesa	30/53	125/51	5/2	981
Washington				
Sherwood	80/32	40/16	40/16	200
Wyoming				
Lucky Mc				
- Piles 1,2, & 3	203/82	143/58	60/24	220
- Evap. Ponds	104/42	104/42	0/0	22
Shirley Basin	275/111	215/87	60/24	208
Sweetwater	37/15	30/12	7/3	280
Inactive Tailings	79/32	0/0	79/32	280

(a) The data in this table has changed with respect to the draft document in response to information recieved during the comment period.

4.2.2.2 Dispersion Parameters

In modeling the releases from the mills, both a stack source and an area source were used to represent the process and tailings releases respectively. A 12-meter stack with a 1.2-meter diameter and volumetric flow of 12.7 meters was used for process exhausts. The total area (wet and dry) of the tailings impoundments was used for the size of area sources.

Meteorological data from the nearest meteorological station with joint frequency data in the form required by the assessment codes were used for the active mills. For the inactive tailings, generic meteorological data presented in NRC80 were used. The sources of the meteorological data used for each assessment are presented in Table 4-4.

Table 4-4. Sources of meteorological data used in the assessment of uranium milling.

Mill	Location	Meteorological Station
New Mexico		
Ambrosia Lake	Ambrosia Lake, NM	Ambrosia Lake, NM
Homestake	Grants, NM	Ambrosia Lake, NM
Texas		
Panna Maria	Panna Maria, TX	San Antonio, TX
Utah		
La Sal	La Sal, UT	Grand Junction, CO
Shootaring	Hanksville, UT	
White Mesa	Blanding, UT	Farmington, NM
Washington		
Sherwood	Wellpinit, WA	Spokane, WA
Wyoming		
Lucky Mc	Riverton, WY	
Shirley Basin	Casper, WY	Casper, WY
Sweetwater	Rawlings, WY	
Inactive Tailings	Generic (see text)	-

4.2.2.3 Demographic Data

The actual populations living within 5 km of the operating mills were enumerated by sector segments during site visits made to each mill in 1983 (PNL84). The data for Canon City, Ambrosia Lake, Homestake, La Sal, and Sherwood were updated following site visits by SC&A in 1989. These distributions, presented in Table 4-5, were used in conjunction with the population distributions for 5 to 80 km generated by the computer code SECPOP from 1980 U.S. Census Bureau data. The population distribution for the generic tailings pile was taken from NRC80.

Actual data on food production in the vicinity of these mills were not obtained. Instead, generic food production rates (urban/low productivity) representative of the areas where these mills are located were used in the assessment.

Table 4-5. Estimated populations living within 0 to 5 km of active uranium milling facilities.(a)

Mill	0-0.5 km	0.5-1.0 km	1.0-2.0 km	2.0-3.0 km	3.0-4.0 km	4.0-5.0 km
New Mexico						
Ambrosia Lake*	0	0	0	0	0	0
Homestake*	0	0	187	104	42	57
Texas						
Panna Maria	0	12	42	33	81	285
Utah						
La Sal*	0	0	0	0	40	0
Shootaring	0	0	0	0	0	171
White Mesa	0	0	0	0	0	8
Washington						
Sherwood*	0	0	0	0	32	17
Wyoming						
Lucky Mc	0	0	0	0	0	0
Shirley Basin	0	0	0	0	0	0
Sweetwater	0	0	0	0	0	0

(a) The data source is PNL84 except where marked with an *. These data were updated following site visits by SC&A in 1989.

4.2.3 Results of the Dose and Risk Assessments of Uranium Mills

The AIRDOS-EPA/DARTAB/RADRISK assessment codes estimate the 50-year committed dose equivalents to organs from exposure via air immersion, ground-surface, inhalation, and ingestion pathways. Table 4-6 presents the results of the dose assessment to nearby individuals and to the regional (0-80 km) populations around uranium milling facilities. The organs listed in Table 4-6 are those where the dose is estimated to contribute 10 percent or more of the total fatal cancer risk.

Table 4-6. Estimated radiation dose rates from uranium mills.

Mill	Organ	Nearby Individuals (mrem/y)			Regional Population (person-rem/y)		
		Process	Tailings	Total	Process	Tailings	Total
New Mexico							
Ambrosia Lake	Lungs	-	8.2E-2	8.2E-2	-	7.4E-1	7.4E-1
	Endosteum	-	2.8E-1	2.8E-1	-	3.3E+0	3.3E+0
	Red Marrow	-	2.2E-2	2.2E-2	-	2.6E-1	2.6E-1
	Remainder	-	7.0E-3	7.0E-3	-	1.5E-1	1.5E-1
Homestake	Lungs	8.7E+1	3.6E-1	8.7E+1	9.7E+1	4.2E-1	9.7E+1
	Endosteum	4.9E+1	1.1E+0	5.0E+1	6.7E+1	1.4E+0	6.8E+1
	Red Marrow	-	8.9E-2	8.9E-2	-	1.1E-1	1.1E-1
	Remainder	-	-	-	-	-	-
Texas							
Panna Maria	Lungs	2.0E+0	-	2.0E+0	1.8E+0	-	1.8E+0
	Endosteum	-	-	NA	1.4E+0	-	1.4E+0
	Remainder	-	-	NA	1.0E-1	-	1.0E-1
Utah							
La Sal	Lungs	1.0E+0	-	1.0E+0	9.7E-1	-	9.7E-1
	Endosteum	-	-	-	1.1E+0	-	1.1E+0
	Red Marrow	-	-	-	-	-	-
	Remainder	-	-	-	9.8E-2	-	9.8E-2
Shootaring	Lungs	-	9.8E-2	9.8E-2	-	1.9E-2	1.9E-2
	Endosteum	-	3.1E-1	3.1E-1	-	7.0E-2	7.0E-2
	Red Marrow	-	2.5E-2	2.5E-2	-	5.5E-3	5.5E-3
	Remainder	-	6.2E-3	6.2E-3	-	2.1E-3	2.1E-3
White Mesa	Lungs	3.5E-1	1.5E-3	3.5E-1	7.1E-1	3.0E-2	7.4E-1
	Endosteum	-	5.0E-2	5.0E-2	7.7E-1	1.6E-1	9.3E-1
	Red Marrow	-	4.0E-3	4.0E-3	-	1.3E-2	1.3E-2
	Remainder	-	-	-	6.6E-2	9.0E-3	7.5E-2
Washington							
Sherwood	Lungs	-	4.2E-1	4.2E-1	-	1.0E+0	1.0E+0
	Endosteum	-	1.3E+0	1.3E+0	-	1.0E+1	1.0E+1
	Red Marrow	-	1.1E-1	1.1E-1	-	8.1E-1	8.1E-1
	Remainder	-	2.6E-2	2.6E-2	-	8.1E-1	8.1E-1
Wyoming							
Lucky Mc	Lungs	-	3.7E-2	3.7E-2	-	1.1E-1	1.1E-1
	Endosteum	-	1.4E-1	1.4E-1	-	9.3E-1	9.3E-1
	Red Marrow	-	1.1E-2	1.1E-2	-	7.2E-2	7.2E-2
	Remainder	-	4.9E-3	4.9E-3	-	6.7E-2	6.7E-2

Table 4-6. Estimated radiation dose rates from uranium mills (continued).

Mill	Organ	Nearby Individuals (mrem/y) Process	Tailings	Total	Regional Population (person-rem/y) Process	Tailings	Total
Wyoming (cont.)							
Shirley Basin	Lungs	7.4E-2	2.0E-1	2.7E-1	3.9E-1	1.1E+0	1.5E+0
	Endosteum	-	7.1E-1	7.1E-1	8.6E-1	1.0E+1	1.1E+1
	Red Marrow	-	5.6E-2	5.6E-2	-	7.9E-1	7.9E-1
	Remainder	-	2.0E-2	2.0E-2	7.6E-2	7.5E-1	8.3E-1
Sweetwater	Lungs	-	2.7E-1	2.7E-1	-	3.2E-1	3.2E-1
	Endosteum	-	9.2E-1	9.2E-1	-	2.4E+0	2.4E+0
	Red Marrow	-	7.3E-2	7.3E-2	-	1.9E-1	1.9E-1
	Remainder	-	2.5E-2	2.5E-2	-	1.7E-1	1.7E-1
Inactive Tailings	Lungs	-	9.8E+1	9.8E+1	-	2.2E+0	2.2E+0
	Endosteum	-	3.1E+2	3.1E+2	-	1.6E+1	1.6E+1
	Red Marrow	-	2.5E+1	2.5E+2	-	1.2E+0	1.2E+0
	Remainder	-	-	-	-	1.0E+0	1.0E+0

The lifetime fatal cancer risks to nearby individuals and the estimated deaths per year in the regional populations are shown in Table 4-7 for each mill. The estimated distribution of the total fatal cancer risk from all mills and the number of persons at each risk interval are presented in Table 4-8. The values of fatal cancer risk distribution from the model inactive tailings pile were multiplied by 15 to obtain an estimate of the distribution from all decommissioned mills. The results for the four operating mills and the seven mills on standby were added to obtain the distribution from all mills.

The only significant pathways for dose and risk are inhalation and ingestion. For nearby individuals, inhalation is generally predominant; for regional populations, ingestion is more important. For nearby individuals, the most significant nuclides released from tailings piles are thorium-230 and lead-210, while the most important plant emissions are uranium-238 and uranium-234. For regional populations, the most important nuclide released from tailings piles is lead-210, but thorium-230, polonium-210, and radium-226 are also emitted in significant quantities. Of nuclides emitted from process stacks, uranium-238 and uranium-234 contribute the most to population dose and risk with, in some cases, less important contributions from lead-210 and thorium-230.

Table 4-7. Estimated fatal cancer risks from uranium mills.

Facility	Nearby Individuals Lifetime Fatal Cancer Risk	Regional (0-80 km) Population Deaths/y
New Mexico		
Ambrosia	2E-7	3E-5
Homestake	2E-4	2E-3
Texas		
Panna Maria	3E-6	5E-5
Utah		
La Sal	2E-6	3E-5
Shootaring	2E-7	7E-7
White Mesa	6E-7	2E-5
Washington		
Sherwood	1E-6	8E-5
Wyoming		
Lucky Mc	1E-7	7E-6
Shirley Basin	6E-7	9E-5
Sweetwater	7E-7	2E-5
Model Inactive Tailings	2E-4	1E-4

Table 4-8. Estimated distribution of the fatal cancer risk to the regional (0-80 km) populations from uranium mills.

Risk Interval	Number of Persons	Deaths/y
1E-1 to 1E+0	0	0
1E-2 to 1E-1	0	0
1E-3 to 1E-2	0	0
1E-4 to 1E-3	84	2E-4
1E-5 to 1E-4	6,500	1E-3
1E-6 to 1E-5	32,000	2E-3
< 1E-6	2,200,000	2E-3
Totals	2,200,000	5E-3

4.2.4 Supplementary Control Options and Costs

4.2.4.1 Controls for Process Releases

The NRC has evaluated additional controls for the process operations that result in significant airborne emissions (NRC80). Several well-proven control technologies can be employed on the ore crushing and yellowcake drying and packaging exhausts. Table 4-9 presents the predicted efficiencies and costs of these technologies. The lifetime costs shown in the last column of the table are based on 15 years of operation.

Table 4-9. Effluent controls for process emissions.

Control	Efficiency, %	Costs (thousands of 1980 dollars) Capital	Annual	Lifetime
Ore Crushing Exhaust Dust-Removal Units				
Orifice	94	55	14	325
Wet Impingement	97.9	138	16.8	390
Low-Energy Venturi Scrubber	99.5	205	32.8	695
Fabric Filter	99.9	387	33.2	885
Fabric Filter & HEPA		407	91.3	1775
Yellowcake Drying and Packaging Exhaust Dust-Removal Units				
Wet Impingement	97.9	45.0	5.5	130
Low-Energy Venturi Scrubber	99.5	55.5	10.8	220
Medium-Energy Venturi Scrubber	99.7	66.1	15.9	305
High-Energy Venturi Scrubber	99.9	71.5	23.8	430
High-Energy Venturi & HEPA		108.2	29.4	550

4.2.4.2 Controls for Windblown Particulates

The solid portion of a dry tailings pile, particularly the slime, is a source of radioactive contamination. The slime contains uranium concentrations twice as high as the sand and, due to its small particle size, becomes easily airborne. Several alternatives have been identified to control potential contaminated dust problems from dry tailings: (a) wetting the tailings; b) leaching the tailings to remove residual radioactivity; c) fixation/solidification of the tailings; (d) application of stabilizers to the surface of the piles to form a crust; and e) covering of the tailings either above or below the ground surface. The method most commonly used at milling operations is wetting of the dry tailings by sprinkler trucks.

This section presents estimated capital, operating, and maintenance costs for each of the alternatives listed above. The